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NEA COMMITTEE ON REACTOR PHYSICS

REACTOR PHYSICS ACTIVITIES IN NEA MEMBER COUNTRIES

October 1986-September 1987

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

NUCLEAR ENERGY AGENCY
38, boulevard Suchet 75016 Paris

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This document is a compilation of national activity reports presented at the Thirtieth Meeting of the NEA Committee on Reactor Physics, held at the Technical Research Centre of Finland in Helsinki, from 14th to 18th September 1987.

REACTOR PHYSICS ACTIVITIES IN AUSTRALIA

(October 1986 - September 1987)

by

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1. RESEARCH REACTOR STUDIES

(i) Neutronic Modelling of HIFAR

A neutronic model of HIFAR which may be used to simulate reactor operations is currently under development. The calculations will be performed in the AUS system (Robinson, 1975) using the POW (Pollard, 1974) and CHAR (Robinson, 1986) modules for neutron diffusion and burnup respectively. A fuel management module HIFUME is currently being written to control the calculations. It is intended that the complexities of AUS and the models employed will be effectively hidden from the user, and that the model can be applied to routine fuel management calculations.

A 2-group coarse mesh (30*30) XY model of HIFAR giving results which compare well with those from the previous 5-group, finer-mesh model (Harrington, 1984) has been established. The cross-sections used have been condensed to 2 groups through bilinear weighting by fluxes and adjoints from an RZ calculation of HIFAR. Both microscopic cross-sections for individual nuclides and macroscopic materials cross-sections have been generated as a function of fuel element burnup. The set of nuclides is sufficient for calculating fission product transients.

The model includes an artificial set of absorbers on the boundary of the core to correct for the errors introduced by smearing the fuel elements and to account partially for the coarse control arms. A method of including the effect of rigs has been developed and a library of the required absorber concentrations for the past five years' operation has been generated. This was done in the course of applying the model, with macroscopic fuel cross-sections as a function of burnup only, to that period. The results were quite satisfactory.

(ii) Tailored Epithermal Neutron Beams in HIFAR for BNCT

For the boron neutron capture therapy (BNCT) of deep seated metastatic melanoma, an epithermal (up to a few keV energy) neutron beam from the HIFAR 10H horizontal facility could be useful if the inherent contamination from fast neutrons and gamma rays could be minimised.

Calculations for HIFAR showed that, even though a filter material such as AlF_3 attenuates the fast neutron dose, the beam quality improvement is counteracted by a relative increase in gamma flux arising from neutron captures in the filter materials, particularly the aluminium. Published studies of the effectiveness of epithermal neutron filters for BNCT applications have almost universally failed to consider this effect and its contribution to the general tissue dose.

Because the aluminium gammas are of high energy, they cannot be attenuated by lead or bismuth without comparable attenuation of the epithermal neutron flux. Most of them, however, arise from thermal neutron captures and the addition of an absorber such as ^6Li (which emits an alpha particle when a thermal neutron is captured) was investigated as a means of reducing the hard gamma source. While the calculations showed that some improvement in beam quality could be expected from the addition of ^6Li to the AlF_3, a conceptual tangential beam with no filter would still be superior.

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AUSTRIAN RESEARCH CENTER SEIBERSDORF (ÖFZS)
INSTITUTE OF REACTOR SAFETY:

In collaboration with KFA-Jülich methods are elaborated for the calculation of mixed HTR cores. These mixed HTR cores occur during the change from highly enriched uranium to low enriched uranium fuel cycles. Special emphasis is placed upon the treatment of the resonance integral since now not only Thorium but also Uranium 238 appear as resonance absorber materials. With these upgraded methods the measured Pu 239 amounts from burnt fuel elements will be predicted more accurately.

REACTOR PHYSICS ACTIVITIES IN BELGIUM

September 1986 - August 1987

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

THERMAL REACTORS

Critical experiments

The validation of calculation methods applied in pressure vessel surveillance programmes is the objective of measurements performed in the VENUS critical facility.

The reactor configuration studied in VENUS simulates the main features of a PWR core (fig. 1) : low enriched UO_2 or UO_2-PuO_2 fuel pins, staircase-shaped core boundary, core baffle, barrel, neutron pad. 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. Most important in a surveillance programme is the fast neutron fluence accumulated in the vessel, at the maximum of the azimuthal distribution. The experimental effort is therefore concentrated on this aspect of the computer code validation.

The VENUS-1 core previously studied contained UO_2 fuel exclusively. Concerning the fast neutron flux measurements outside the core, a general agreement was found with calculations although the C/E values decreased with the distance from the core boundary, mainly at high energy. The theoretical analysis was made with DOT 3.5 in (R,Θ) geometry, using the S_8P_3 approximation and 17-group cross-sections derived from the VITAMIN-C 171-group library.

The VENUS-2 core now under investigation has been defined in order to examine how far major conclusions from the VENUS-1 study can be affected by significant amounts of plutonium-239 in the outer regions of the core loading. This simulates the presence of high burn-up fuel assemblies placed at the core periphery in power reactors in order to decrease the neutronic exposure of the reactor vessel.

The major part of the measurements for the VENUS-2 campaign has been completed. Scanning of irradiated fuel rods from a 45° sector (see fig. 1) has provided the core power distribution; for an average value : 1.0, the fission density varies in the range 1.52 to 0.24 according to the location. Computed and measured results agree within better than 5 % in the UO_2 zones and C/E values in the range 1.03 to 1.09 are observed in the UO_2-PuO_2 zone.

For the neutron propagation outside the core, the selected measurement locations were concentrated on two radial and one azimuthal traverses. Several sensors were used : $^{238}U(n,f)$, $^{237}Np(n,f)$, $^{103}Rh(n,n')$, $^{115}In(n,n')$, as in the VENUS-1 campaign, but also $^{64}Zn(n,p)$, $^{27}Al(n,\alpha)$, $^{58}Ni(n,p)$ in order to clarify the discrepancies in the high neutron energy range. In particular, axial buckling measurements in the outer structures are examined, to study the space-energy dependence of the axial leakage which should be taken into account in the two-dimensional (R,Θ) calculations.
Results of measurements and their analysis can be found in [1][2][3].

It is planned to pursue the VENUS programme in collaboration with US laboratories after completion of the VENUS-2 study. The objective of VENUS-3 will be to support the PLSA (Partial Length Shielded Assembly) concept [4], aiming at a strong decrease of the neutron fluence in the circumferential welds of a vessel. A feasibility study has been carried out and the validity of a simulation of this effect in VENUS has been confirmed.

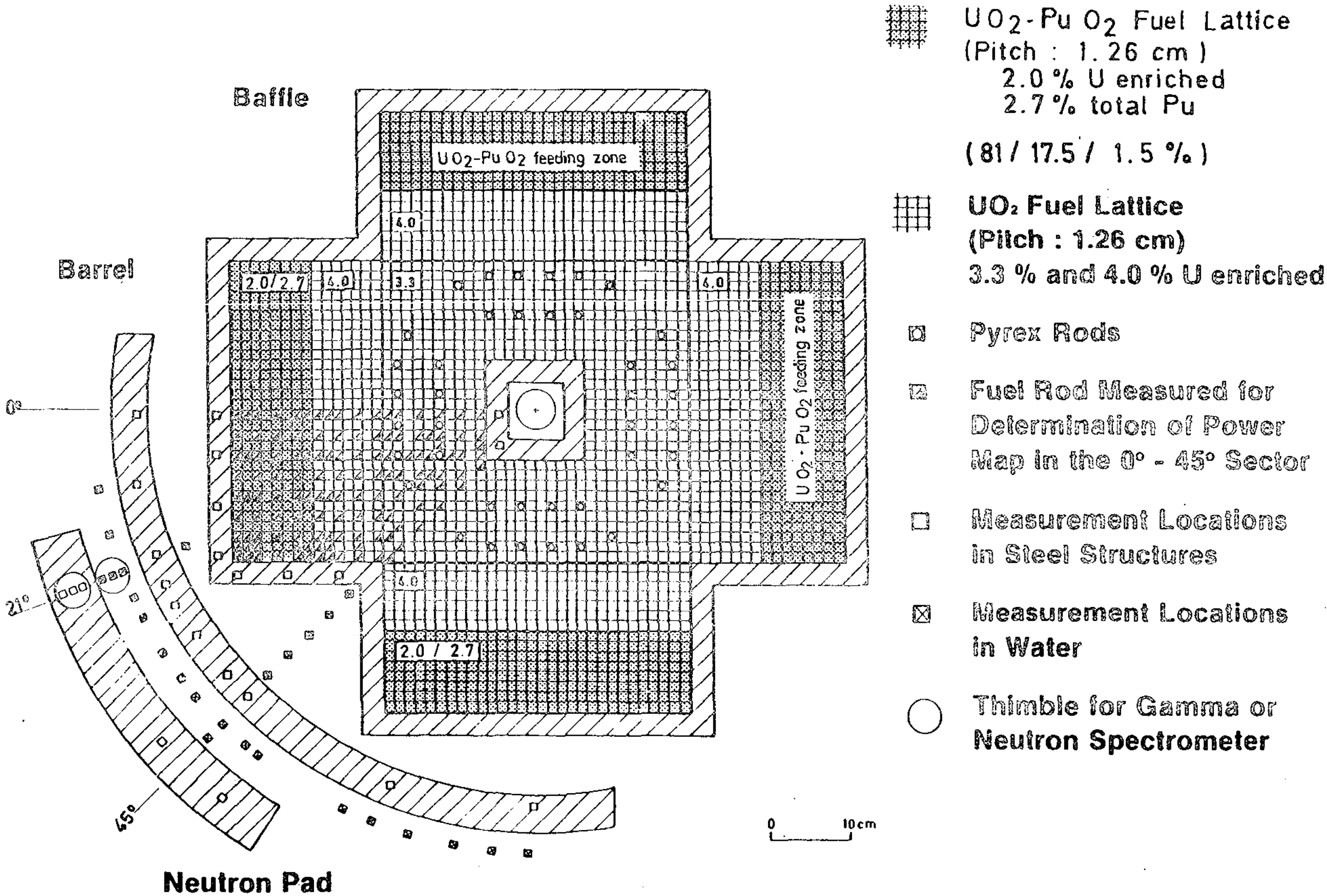

FIG. 1

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Development of reactor codes

The physics methods used by BELGONUCLEAIRE for calculating LWR performances have been reviewed [5]. The main tools are the LWR-WIMs code and MICROLUX, a three-dimensional simulator code, which provides depletion and load-follow calculation results. In MICROLUX, a nodal formalism calculates the 3-D power distribution and the critical boron concentration using a modified one-group coupling model; on the basis of nodal results, a fine-mesh option can determine the rod-to-rod power distribution over selected fuel assemblies. Typical accuracies are : $\pm$ 200 MWd/t on cycle length, $\pm$ 15 ppm on critical boron concentration, $\pm$ 5.10^{-5}/°C on moderator temperature coefficient.

The utilization of MICROLUX as a part of the expert system installed in the control room of a power plant has been investigated. The expert system provides information to the operators and allows a better understanding of the operating events. The comparison of experimental data obtained at the DOEL-3 power plant during two cycles with the MICROLUX theoretical results showed the ability of this code to calculate the main reactor parameters. In particular, a return to full power after a shut-down period of six hours was analysed [6]. Axial offset deviation outside the operational limits did not permit to restart the plant as expected, but only after a 18-hour operation at low power. The MICROLUX analysis allowed to clarify inconsistencies between the in- and ex-core instrumentation concerning this axial offset and demonstrated that the one-line application of such a code can contribute to a safer and more economical operation of the reactor.

MERCATOR, a nodal code developed by TRACTEBEL for solving the two-group diffusion equation on large reactor cores, has previously been described in a NEACRP paper [7]. The high accuracy of the M20 version of MERCATOR (2 D, two-group, homogeneous nodes) has been demonstrated in a 2 D IAEA benchmark exercise, with one or four nodes per fuel assembly (1 N/A or 4 N/A). Tests on practical data (Biblis, Tihange) led to the same conclusion, with a typical speed of 5 cycles in 1 N/A or 3 cycles in 4 N/A partition. Execution speed of MERCATOR enables to include it in a interactive assembly repositioning software which is currently used for the core management of the Tihange reactors. New developments have been undertaken (M25, M30 versions); they concern a 3-D extension and heterogeneity corrections [8].

At the University of Brussels, a new nodal method is being developed for the solution of 2 D multigroup transport problems. This method uses standard partial integration to, first, reduce the problem to a one-dimensional transport equation with transverse leakage. A formal integration of the 1 D problem is then performed, using trial functions which are harmonic polynomials, therefore solutions of the Boltzmann equation for c = 1 in infinite media. Circular permutation yields a set of "response matrix" equations, relating incoming and outgoing current moments in adjacent cells which are solved by classical techniques. The response matrix elements can be evaluated analytically using symbolic programming techniques. This allows a compact formulation and an easy reevaluation of these response matrix elements when cross-sections change by feedback. The proposed method has been applied to the Biblis benchmark problem. Concerning the computer time, this method is much faster than a similar diffusion code which for a comparable accuracy requires a much higher number of nodes per assembly [9].

Noise measurements

Several papers will be presented at the SMORN-V meeting in Munich (October 1987). They will describe the experience gained in the neutron noise monitoring of the nuclear power plants operated in Belgium. Neutron noise monitoring was performed by LABORELEC with ex-core neutron detectors on a regular basis since the commissioning of the first unit. In-core neutron noise monitoring was started later; furthermore loose part monitoring systems were finally installed. Two units are provided with turbogenerator vibration monitoring systems [10].

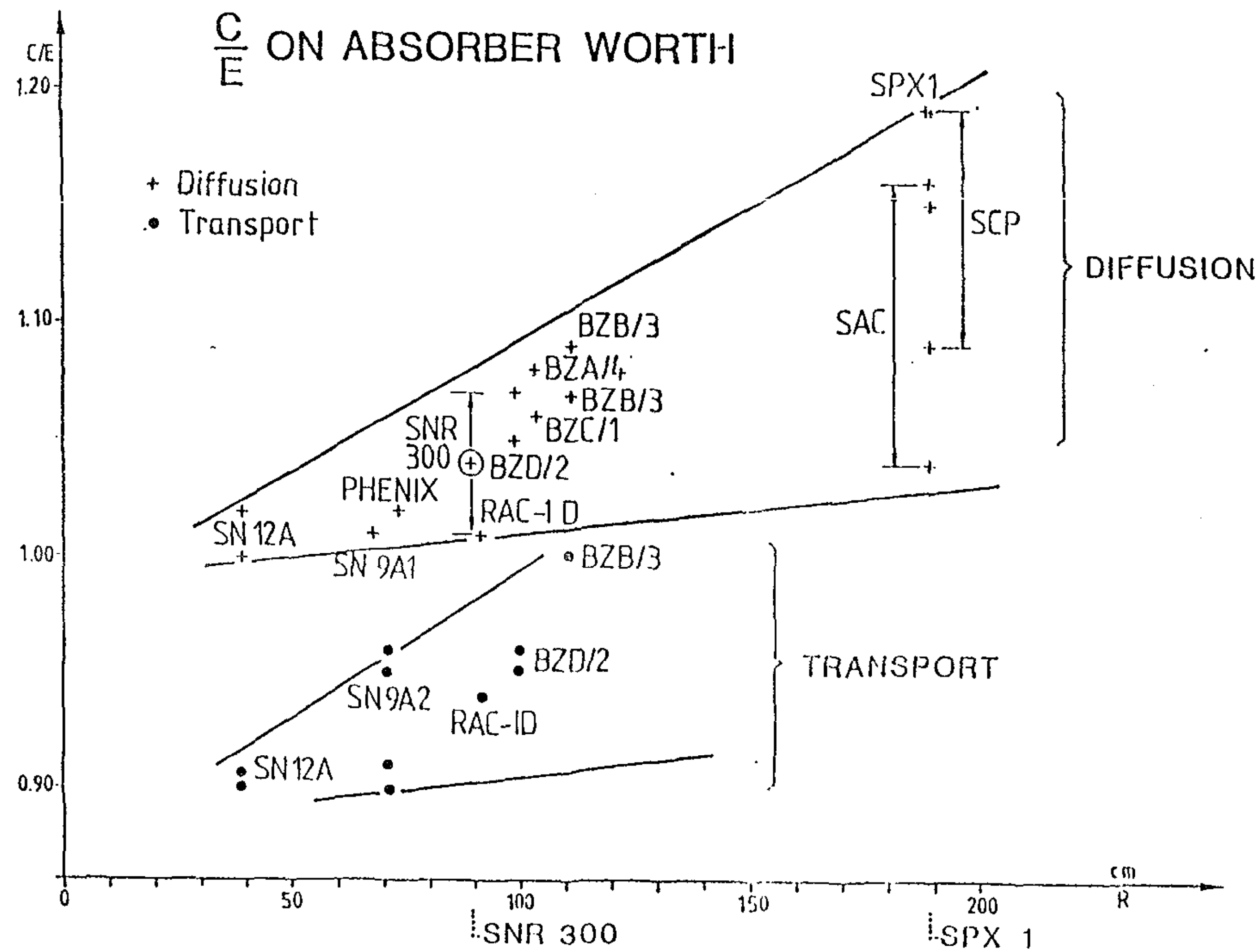
A specific application of the neutron noise detection technique concerned the observation of nucleate boiling in some fuel assemblies at the DOEL-2 plant. Inspection during refuelling revealed thick oxide layers on the fuel rods of these assemblies; these abnormal surface conditions were at the origin of the anomalies. The void fraction and the bubble speed were inferred from the noise spectra. This allowed to calculate the axial offset and the core outlet temperatures using simple thermohydraulic and neutronic models. The results agreed with the flux map and thermocouple readings [11].

In order to get a better understanding of the relationship between noise measurements and boiling phenomena, an experimental investigation was conducted at SCK/CEN on the detection of localized onset of boiling in water on a heated pin. Accelerometers placed on the pin and on the surrounding structures at appreciable distance from the boiling spot were used. A detailed analysis of the results and a comparison with the theoretical bases of boiling have been carried out in order to extract the actual boiling information from the structure background vibrations. The influence of pressure and temperature have corroborated the proposed relationship between the measured noise and the bubble growing and collapsing processes. The observations made on different structures have assessed the independence of the analysis with respect to the structure particular dynamic behaviour [12].

FAST REACTORS

1. In the frame of the co-operation agreements between France and DeBeNe, BELGONUCLEAIRE contributes to the evaluation of the SUPERPHENIX start-up experiments. It firstly consisted in calculations of typical configurations of the first criticality core and of the power start-up core with standard design methods used in the SNR project (KfK-INR data, diffusion, synthesis programme KASY), in order to bring the evaluation in line with the numerous results available from previous critical experiments (SNEAK, BIZET, RACINE). The new aspect and interest of SUPERPHENIX lies in the size of the reactor, with a core radius about twice as large as the biggest critical assembly of the BIZET family. The size of the core has a direct influence on the mutual interaction between control rods and therefore also on the accuracy of the calculation of the control rod worths. In figure 2, C/E on absorber worth is plotted versus core

BELGO NUCLEAIRE



RATIO : CALCULATION TO EXPERIMENT (C/E) ON ABSORBER WORTH AS A FUNCTION OF THE RADIUS OF THE CORE

FIG. 2

radius; the values obtained for SUPERPHENIX broadly confirm the general trend from the critical experiments; they also show a considerable increase of the dispersion.

An other task under progress, with the participation of BELGONUCLEAIRE, is the determination of the reactivity coefficients. These coefficients are determined by analysing the effect of power perturbations and variations of primary and secondary coolant flow at various power levels between 20 % and 100 %. The variation of the reactivity $\delta\rho$ between two equilibrium states is expressed by

$$\delta\rho = k\ \delta\ T_e + g\ \delta(\Delta T) + h\ \delta\ P$$

where T_e = sodium inlet temperature

ΔT = sodium temperature increase over the height of the core

P = reactor power

The discrepancies between measured and computed values of the reactivity coefficients vary at the various power levels, the largest difference being found for the g coefficient. This question is being investigated.

2. Concerning the validation of computation methods and codes on the basis of experiments in the critical facility MASURCA, SCK/CEN and BELGONUCLEAIRE contribute respectively to the measurement and to the analysis of gamma heating rates in the frame of the BALZAC programme. Irradiations of thermoluminescent dosimeters took place successively in Na-SS diluant assemblies and in the B_4C absorbing part of control rods.

3. PAHR (Post Accident Heat Removal) experiments are being conducted in BR2 as a part of the fast reactor safety programme of the Commission of the European Communities. In the first experiment performed in November 1986, a mixture of UO_2 particles in a crucible with sodium has been irradiated in the central channel of BR2 at a temperature of

about 1500°C. Two similar experiments will be performed in the future, at higher temperatures.

Neutron and gamma calculations (one and two-dimensional) contributed to the optimization of the irradiation device and determined the heating rate distributions and level. The latter compares satisfactorily with the results of an analysis of the data collected during the irradiation. As quality control of the particle bed, a neutron transmission measurement with a beam issuing from the BR1 reactor allowed, before the irradiation, to scan the bed in order to determine the porosity (Na volume fraction) variation. Similar but much more detailed measurements are performed on different UO_2-SS mock-ups with sodium, in preparation of the second experiment in BR2 [13].

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Reactor Physics Activities in Canada

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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 and other small reactor concepts.

This report summarizes the activities carried out by the various organizations during the period from October 1986 to September 1987.

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, a simulation model has been developed which can be used to simulate the power rundown 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. Existing modelling techniques were improved in two areas. Ontario Hydro has developed 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 have also been developed. A spatial modal kinetics code is used to compute the power transient from the reactivity as a function of time.

1.1.2 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.

1.1.3 Support of Operating Nuclear Generating Stations

The present occurrence of power manoeuvering by Ontario Hydro reactors and an expected increase in its frequency have necessitated the enhancement of the steady state diffusion theory reactor simulation code SORO to give it the capability of simulating the transients associated with such manoeuvers. The code development and validations have been proceeding in two directions:

a) The instantaneous flux and power distributions are inferred from a finite number of in-core thermal flux measurements (Flux Mapping) and channel power measurements (Channel Power Mapping).

b) Reactivity transients (Xe, Pu, Sm, Rh) are explicitly modelled in the SORO module that calculates neutron fluxes.

Lattice codes are being studied in order to improve reactor core simulation capability by eliminating errors introduced through modelling deficiencies in existing fuel tables. In particular, it has been shown that the tables have introduced a systematic burnup-dependent channel power discrepancy.

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 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, Royal Military College and McMaster University.

2.1 Groupe D'Analyse Nucléaire (GAN), Ecole Polytechnique, - Montréal, Québec

2.1.1 Activities related to diffusion Theory and Applied Mathematics

A FORTRAN-77 version of the fuel management code SIMEX was set up around the three-dimensional neutron diffusion code TRIVAC[3] and the hierarchical data bases NSES[4]. The nodal collocation method with appropriate approximation was concurrently introduced in TRIVAC as a new discretization algorithm to solve the coarse mesh neutron diffusion equation for light water reactors.

The 2-D version of the fuel management optimization code OPTEX was extended to provide simultaneous optimization of the cobalt adjuster rod distribution in a CANDU reactor core[5]. A time-averaged equilibrium refuelling model has also been introduced in the 3-D version (OPTEX-4), which is interfaced with TRIVAC for the diffusion calculations.

2.1.2 Activities Related to Transport Theory and Lattice Cell Calculations

Work in transport theory was directed toward the three following subjects:

- Production of a FORTRAN-77 version of WIMS-CRNL with a standard direct access version of WIMSLIB for use on various types of computers.

- Deterministic resolution of the transport equation using collisions probabilities in a mixed cartesian and cylindrical three-dimensional geometry. A computer code, named EXCELL, was produced and used to treat the reactivity devices effects in a CANDU reactor.[6]

- Conception of a generalized driver for performing power iterations and buckling search for a variety of discretization techniques, including collision probability methods and iterative interface current formalisms (such as EXCELL). The GOXS format was used throughout for cross section exchanges between each module.

2.2 McMaster University (see References 7 to 18)

2.2.1 Neutron Beam Extraction

This research relates to continuing interests in neutron radiography and involves Monte Carlo calculations in scatterers adjacent to the core. Some of the findings suggests that beam extractions based on density-varying scatterers may considerably enhance neutron beam uniformity.

2.2.2 Nonlinear Fission Dynamics

Interest in nonlinear fission reactor dynamics based on interactive canonical formulations is yielding a considerable diversity of representations. Power fluctuation which display chaotic patterns as well as those which show self-limiting features are of particular interest.

2.2.3 Muon Catalized Fusions

Muon catalized fusion reaction processes possess some interesting similarities with fission (e.g. operation at ~ 600 K, reaction chains, etc.). The McMaster group is continuing to examine the energy viability of such nuclear energy systems and accordingly is formulating sufficiently useful performance parameters. It appears that in the absence of considerable energy cost reductions of muon production, such fusion systems may need to be synergetically integrated with fission/breeding systems.

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.

3.1.2 Demonstration of Load-Following Capability of CANDU Reactors

Increasing interest is being shown by CANDU owners in performing load-following cycles which suit their grid demand.

Recently a simulation of a load-following cycle was carried out for the CANDU 600 reactor, and for actual load-following operations performed at Ontario Hydro CANDU stations.

Results of the simulation show that the cycle can be accommodated by the reactor regulating system (RRS) without exceeding fuel power limits or causing spurious trips in the reactor protection systems.

In practice the CANDU load-cycling capability has recently been confirmed by Ontario Hydro at its Pickering-B and Bruce-B CANDUs. A typical load cycle consisted of reducing power to 50% in about 30 minutes, holding at that power for a few hours, then returning to full power over a period of about six hours. Deeper load-following manoeuvers, where power is reduced to about 20-25% of full power, have also been successfully accomplished at Bruce-B.

In summary, a substantial load-cycling capability of the CANDU system has been effectively demonstrated both by simulation and in practice.

3.1.2 An Investigation into the Relationship Between Local and Global Power Excursions in CANDU

If a reactor exhibits large neutronic decoupling of one core region from another, it may be able to sustain localized power excursions. If such an excursion is too localized, there may be no in-core detector close enough to detect it promptly.

To confirm that CANDU reactors are unlikely to support local power excursions, a credible local reactivity-insertion mechanism was selected in the CANDU 600 to determine the resulting neutron flux transient with a three-dimensional model.

A coupled neutronics-thermalhydraulics simulation of the transient was performed. The transient was subdivided into appropriate time steps, and iteration between the neutronics and thermalhydraulics calculations was carried out at each step, so that the proper distribution of thermalhydraulic feedback reactivity was obtained.

The calculation shows that, with credible reactivity insertions, the neutronic characteristics of CANDU reactors do not allow local power excursions to occur without a comparable global power increase. The latter would not escape detection by the protective systems.

3.1.3 The Chernobyl Accident: Multi-Dimensional Simulations to Identify the Role of Design and Operational Features of the RBMK-1000

A multi-dimensional analysis of the CHERNOBYL accident was carried out to identify the role of the design and operational features of the RBMK-1000 and thereby identify implications for other reactor concepts. The results show that assumptions regarding the pre-accident fuel burnup and flux distributions are major determinants of the size and shape of the power pulse, especially due to their influence on effective system void reactivity and on the amount, if any, of positive scram reactivity.

3.1.4 CIRENE Startup Transients

The CIRENE reactor is a heavy water moderated, light water cooled, 40 MWe Italian designed prototype being constructed about 70 kilometers south of Rome. As part of a contract awarded to AECL by ENEL and ENEA, start up analyses for this reactor have been carried out. In this analysis the feasibility of optimizing the start up procedures to speed up the approach to full power and the possibility of providing some xenon over-ride capability during commissioning have been investigated.

3.1.5 Adapting Reactor Physics Computer Codes to Microcomputers

In order to assess the potential of micro/minicomputers in the realm of reactor physics calculations, some physics programs have been adapted to run on an IBM-PC/AT with one megabyte of central memory and a 20-megabyte hard disk.

Results obtained are considered quite positive. For example, typical MULTICELL calculations have been achieved with a calculational speed (in the usual one-group option) of about 7.5 s/iteration. Total run times for a MULTICELL calculation are of the order of one hour on the PC/AT. Inherent machine speed can be enhanced with mathematical co-processors.

Computing times are expected to be about a factor of ten smaller on the personal work stations such as the APOLLO.

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 Business Units, external partners and clients and the site. These groups are: the MAPLE Research Reactor Assessment Group, the Small Reactor Concepts Group and the Shielding and Criticality Safety Group. The current scope of reactor physics work for each group is described below.

3.2.1 MAPLE Research Reactor Assessment Group

The purpose of this group is to provide state-of-the-art reactor physics support to the MAPLE Research Reactor Business Unit for the continuing development of the generic MAPLE concept and for the evaluation of customer-specified variations as well as the MAPLE-X prototype being built at CRNL. 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 MAPLE-10 operating at 1 MW, for a proposal to Colombia as well as a 5 MW version for McMaster University. The group is also preparing reference supercell WIMS-CRNL models to provide cell-averaged cross sections for the MAPLE-X reactor. A newly developed 2-D kinetics code, TANK, has also been used to support design decisions for MAPLE-10 and MAPLE-X. TANK incorporates fuel temperature, coolant temperature and coolant void reactivity feedback effects. The heat transfer coefficient can be adjusted via the heat transfer package which was borrowed from the SPORTS-M thermalhydraulics code.

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:

- development of a code, to evaluate isotope production scenario, using multigroup activation
- the development of the 2-D kinetics code TANK.

Work in progress or planned for the near future includes:

- improvement of the diffusion modelling for more accurate determination of the reactivity worths of strong absorbers
- improvement of transport/diffusion modelling for the reactivity worth and neutron transmission of beam tubes. While Monte Carlo techniques can give definitive results they have prohibitive resource requirements when dealing with cases involving D_2O.
- further improvement of the WIMS-3DDT code system to develop an automatic fuel management modelling technique
- contributions to the development of WIMS-CRNL
- improvements to MAPDDT to simplify data input and allow the specification of distributions of fuel parameters other than burnup such as fuel temperature.

3.2.2 Small Reactor Concepts Group

The Small Reactor Concepts Group is evaluating the technical and economic feasibility of innovative reactor design for small-scale electricity generation (less than 1 MW(e)). Possible Canadian applications include power for remote villages and special end-user applications.

Most of the group's effort is being devoted to a heat-pipe-cooled, graphite-moderated concept referred to as the Nuclear Battery. Both 20% enriched TRISO coated particle fuel and 5% enriched uranium carbide (UC) fuel have been considered in design studies.

The WIMSD-CRNL neutron transport code, the 3DDT three-dimensional neutron diffusion theory code and the MCNP Monte Carlo code are routinely used for core neutronics studies. Some problems have been encountered in the calculated core temperature reactivity coefficient when using graphite cross sections derived from the ENDF/B-5 data library. This problem is believed to result from inappropriate processing of the thermal scattering data for graphite and is expected to be corrected shortly. Code validation work for graphite lattices and a modification of WIMSD-CRNL to incorporate a resonance treatment for particulate fuel will also be undertaken.

An alternate and more compact reactor concept, based on BeO neutron moderator and liquid Lithium-7 coolant, is in the early stages of feasibility investigation.

3.2.3 Shielding and Criticality Safety Group

The majority of the Shielding and Criticality Safety Group effort during the past year has been in support of small research reactor and special purpose reactor design projects. Detailed shielding assessments of the following systems were completed: 30 MWt Korea Multipurpose Research Reactor, 5 MWt Generic Maple-10 Research Reactor, 300 and 500 kWe Nuclear Battery Reactors for use in a special end-user application, and 10 MWt SES-10 Heating Reactor. A similar assessment of the 10 MWt Maple-X Research (MCNP) models was developed for specialized tasks, including: the upper access region of the Nuclear Battery, Maple-10 beam tubes using energy-dependent weight windows,and production rates of ^{16}N and ^{19}O activation products in Maple-X using a coupled K_{eff}/fixed-source method. The group also performed shielding calculations for a variety of spent fuel transfer flasks, handling facilities, and storage canisters.

Criticality safety assessments were performed for fuel handling and storage facilities for the Korea Multipurpose Research, Slowpoke Demonstration, and WR-1 Reactors.

Reactor physics assistance was provided in the start-up, initial criticality, and low-power commissioning program for the Slowpoke Demonstration Reactor located at WNRE. The low-power experiments have been completed. Shielding experiments have been designed and pre-start-up background measurements are complete.

The computer code upgrade effort continued with the acquisition of the SCALIAS 3.1 System. This system includes the criticality code KENO-5A and the isotope generation and depletion code ORIGEN-S, together with several cross-section libraries and processing codes. The system is currently being tested. The 3-dimensional graphics program SABRINA has been acquired and implemented for verification and illustration of shielding models. The code is compatible with both the Combinatorial Geometry system used in QAD-CG and the MCNP Surface Geometry system.

Finally, the Slowpoke Spent Fuel Transfer Flask was loaded with two ^{60}Co sources and the surface dose rates were measured. Values calculated with QAD-CG agreed with the measured values to within the uncertainty of the detector (approximately 20%) provided the measurements were not taken near lifting lugs, drain plugs, scattering paths, or voids in the lead.

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.

Some of the published information from the CRNL group is described in references 19 to 30. 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 experimental reactor physics work is centred on 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.

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 as time permits.

A major series of experiments on $(U\text{-}233,Th)O_2$ fuel will commence in 1988.

3.3.2 Measured Temperature Reactivity Coefficient

There is a discrepancy between measured and calculated moderator temperature reactivity coefficients of light water reactor cores. The differences are reported to be larger than the uncertainties in the measured coefficients and in the differential neutron cross section data used in the calculations. In attempts to remove the discrepancies several authors have postulated U-235 and U-238 cross sections that have energy dependencies in the thermal energy range different from those given in the best evaluated data libraries such as ENDF/B-V. To provide data that does not depend on the nuclear properties of H_2O, measurements were made of the buckling of a ZEEP rod lattice in the ZED-2 reactor as a function of moderator and fuel temperature from 10 to 40°C. The results are now being analyzed.

3.3.3 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. Low power commissioning of the reactor is in progress.

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

3.3.2.2 Advanced CANDU Reactor

The natural-uranium fuelled CANDU nuclear reactor system has proven to be a safe, reliable, and economical producer of electricity for over a quarter of a century. The CANDU system, however, is not restricted to the use of natural-uranium fuel; a wide range of advanced fuel cycles can be accommodated. In the short term, slightly enriched uranium (SEU) is the most promising of these advanced fuel cycles. SEU offers a reduction in the total fuel cycle cost of between 25% and 50% relative to natural-uranium fuel. Uranium consumption is decreased by 30% to 40%. In addition the volume of spent fuel is reduced by a factor of two to three, depending on the enrichment selected. SEU also offers greater flexibility in the design of future CANDU reactors.

A variety of fuel management options can be employed in CANDU with slightly enriched fuels. Fuel performance is expected to be good for the burnups of interest, but further fuel testing is planned and is currently in progress in order to confirm this. Programs in place at Atomic Energy of Canada Limited (AECL) will lead to the demonstration and introduction of slightly enriched uranium in CANDU. Ontario Hydro, a Canadian utility with twenty CANDUs operating or under construction, is considering a program which could lead to the implementation of a SEU in its nuclear generating stations.

Different studies undertaken over many years have all indicated an economic incentive for the use of slightly-enriched uranium (SEU) in CANDU reactors. The extent of this incentive has varied with the relative cost of uranium and enrichment services available at the time. Recent reviews within AECL to identify new fuels that could be developed for use in CANDU reactors have indicated that SEU is now one of the most promising candidates. This current view is based on not only the attractive economics associated with the prospect of low enrichment costs in the future, but also on a number of technical benefits, and a large reduction in spent fuel volume.

A part-core loading of 1.2% slightly enriched uranium (SEU) in the peripheral channels of a CANDU-600 reactor offers several benefits over a natural uranium (NU) fuelled CANDU. Annual total fuel cycle costs are reduced by about $5 M. There is a potential for total reactor power uprating, through increased flattening of the radial channel power distribution, of between 5-7%. These benefits can be achieved while using a simple fuel management scheme throughout the core. The transition from the NU core to the mixed SEU/NU core can be achieved in a straight-forward manner.

3.3.3 Code Development and Nuclear Data

3.3.3.1 In-115 Cross Section

To check the ENDF/B-V data the In-115 capture rate was calculated in nine CANDU-type lattices with the WIMS-CRNL code. Calculations for six of these lattices were also made with the Monte Carlo code MCNP and for two with the RAHAB code. The agreement of these results of the different codes was in general good. However the calculated values did not agree well with measured capture rates, and were 2% low to 11% high. These differences are significantly greater than the expected accuracy of the measurements. Because of the variation in the differences, the comparisons do not give useful information on the accuracy of the measurements.

3.3.3.2 Lu-176 Cross Section

The ENDF/B-V cross sections for Lu-176 were also checked with the same lattice calculations used for checking the In-115 cross sections. The calculated capture rates are about 1.5% higher than measured. This indicates that the Lu-176 capture cross section may be high by this amount or that the thermal scattering data are slightly in error.

3.3.3.3 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. Routine calculations for NRU included flux and power distributions and fuel management. A new 3-D simulation code is available using discontinuity factors, which gives improved results ($\pm 5\%$) for this highly heterogeneous reactor. 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 major effort is continuing 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 radioisotope requirements[16]. This reactor will be built in 1988, and will replace some of the capabilities of the NRX reactor.

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 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 was performed.

3.3.5.1 Comparison of the Neutronic Environments in an Li_2O Fusion Blanket and Several NRU Irradiation Facilities for Breeder Material Tests

Irradiations of potential fusion breeder materials in fission reactors have provided information on material properties and tritium migration. At the Chalk River Nuclear Laboratories, vented and unvented irradiations of solid breeder materials are ongoing. Neutronic calculations show that NRU irradiations can provide some of the key parameters for breeder material tests.

An assessment of the neutronic environments in several NRU irradiations facilities has been performed and compared with a fusion blanket (Li2O/He) environment. This information helps quantify the relevance of the present experiments, and indicates possible future tests. The fusion blanket calculations also provide insight into the neutronic characteristics of fusion devices. This is essential for the development and assessment of potential breeder materials and reactor concepts.

NRU irradiation facilities provide a good environment for breeder material tests. Tritium production and lithium burn-up rates that meet or exceed those in a fusion blanket can be achieved. Material damage from helium gas production and displacement damage from energetic charged particles [produced from the exothermic Li-6(n,reaction] can also be matched. Displacement damage from energetic neutron interactions is one area where the NRU experiments cannot reproduce the fusion blanket environment. However, by appropriately selecting the irradiation facility and/or the Li-6 enrichment, the relative contributions to the material damage from helium gas and atom displacements can be controlled. This would permit an investigation of the relative importance of these damage components to be performed.

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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REACTOR PHYSICS ACTIVITIES IN DENMARK

Compiled by Erik Nonbøl
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1. Calculations on Quad Cities

Core calculations on the BWR reactor Quad Cities cycle 1 have been analysed. The calculations are made with the code system LEWARD/NOTAM (1,2) and the results are compared with axial TIP measurement (3) to serve as a partial verification of the code system.

The LEWARD code is an LWR assembly code for neutron and gamma distribution calculations using the method of collision probability. It is used for producing cross sections tables including discontinuity factors to the core simulator programme NOTAM.

The programme NOTAM is a 3-dimensional neutronic-hydraulic code designed to make calculations on boiling water reactors in static conditions. The neutronics of the reactor is treated by a method based on the Trilux-principle (4) with coupling coefficients while the hydraulic model solves the equations for conservation of mass, energy, and momentum for the coolant flow in their one-dimensional form.

The calculation of the effective multiplication factor for the simulation of cycle 1 showed a standard deviation of about 0.2%. The TIP measurements were compared with the calculations at two different burnup values. In the beginning of the cycle the r.m.s. eror was of the order 5-10% while at the end 10-15%.

We hope to be able to improve the results by including burnup dependent weight factors in the detector response from the four adjoining fuel assemblies as well as ascribing some kind of control rod history to each node in the NOTAM code.

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2. Estimation of Radiation Levels in and around a TOKAMAK Fusion Reactor (NET)

Under contract with the NET (Next European Torus) Project, calculations of neutron flux distributions in the DN (Double Null) Tokamak design have been performed, using the MCNP Monte Carlo code.

A very detailed activation code, ACTIVA, has been written, which keeps account of almost any weird nuclide being created by neutron processes (19 different processes are considered) or by radioactive decays (β^+,β^-, and radiative transitions). The gamma yields from neutron captures and from the radioactive decays are calculated in a number af groups, which form the basis for new MCNP calculations on photons alone, giving the radiation levels in selected points in and around the Tokamak.

REACTOR PHYSICS IN FINLAND

STATUS REPORT TO THE NEACRP 1987

Compiled by M. Rajamäki
Technical Research Centre of Finland (VTT)

1 GENERAL

The very good performance of the two Soviet WWER plants (Loviisa) and the two ASEA-ATOM BWR (TVO) plants in Finland has continued in 1987. 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 validation of methods and codes and at their improvements. Since the Chernobyl accident, the particular attention has been given to RBMK-reactors and their possible transient scenarios.

2 CELL CALCULATIONS AND NUCLEAR DATA PROCESSING

The CASMO fuel burnup program has been used, as extensively as before, for generating few group constants for new BWR fuel types intended for reloads of the TVO-I reactor. With the newest version of the program (CASMO-3G) some modern BWR control rods containing hafnium have been studied, too.

The study on the implementation of the so called advanced homogenization methods in the CASMO-HEX/HEXBU-3D program system for the WWER-reactors is still continuing. However, the main effort in this field has been devoted to the validation of the cross section sets generated with the CASMO-HEX code for ICFM calculations of WWER-440 type reactors.

The nuclear data processing code NJOY (6/83) has been installed on the MicroVax computer of the Nuclear Engineering Laboratory. As a test case, the temperature dependent cross sections for graphite needed for RBMK-calculations were processed starting with data from the ENDF/B-V and JEF-1 libraries.

3 CORE CALCULATIONS

The length of the 1986 - 87 cycle for the TVO-I BWR was predicted very accurately by the CASMO-POLCA code system. The average value of the computed k_{eff} for the critical core was almost the same as in the year before (38 pcm smaller), but the variations during the cycle were slightly larger, about -160 to +220 pcm from the average result. The fresh fuel loaded in 1987 was mostly of the 9x9 type with blankets of natural uranium. Gamma detectors were installed first in TVO-I and later in TVO-II during the winter 1986 - 87. The result was a much better agreement between measured and computed power distributions than obtained with the old neutron detectors, especially in the horizontal direction. The errors were approximately halved. The use of some kind of expert system to make the planning of reloading and control rod patterns planning more efficient is still being considered.

The three-dimensional nodal program HEXBU-3D has been applied to the WWER-reactors at the Bohunice nuclear power plant in Czechoslovakia. The simulation of operating history gave generally as good results as obtained for the Loviisa reactors.

The verification of the methods to predict reactivity coefficients with HEXBU-3D has required still some effort. A benchmark problem for comparison of calculated and measured coefficients of the Loviisa reactors will be presented in the next meeting of the Temporary International Collective (TIC) for joint research into the physics of WWER-type reactors.

The preliminary results calculated with cross section data generated by the CASMO-HEX program have been in close agreement with measured reactivity coefficients. However, test calculations have shown that the calculated values are fairly sensitive to the reactor conditions applied in calculations of the two-group nodal cross sections.

4 DYNAMICS CALCULATIONS

The BWR dynamics code TRAB has been verified against the major transients of the TVO reactors including a pump stopping, an overpressurization transient and an instability transient but recently more complete verification calculations and sensitivity studies have been performed (in cooperation with the Finnish safety authority STUK) using data from a real BWR overpressurization transient occurring in 1985. The calculations overestimated slightly pressures but the differences between the measured pressures and the best estimate calculated ones were within half bar while the maximum of the pressure rise was about 8 bar. In addition, we believe that main factors in the difference are caused by the delays of the measurement system and by the fact that four steam lines were described with one line.

The transverse shape function dynamics of the neutronics model of TRAB has been employed in a geometry with up to three radial regions divided further axially into subregions with different shape function parameters. The model has been applied to a study of local stability characteristics in a realistic reactor case. The results are still very preliminary. Fig. 1 represents the power oscillations caused by a pressure disturbance and by a simultaneous disturbance in the initial half core powers.

The testing of the three-dimensional WWER dynamics code HEXTRAN is being initiated this autumn.

5 ACTIVITIES INDUCED BY THE CHERNOBYL ACCIDENT

Much of the Chernobyl-related work at VTT has been aimed at determining the applicability of our reactor physics codes to RBMK calculations. We have found that none of our codes is really adequate. The WWER cell burnup code CASMO-HEX appears to be our most suitable tool and it can give acceptable results at low void content. However, a serious deficiency (so far as RBMK studies are concerned) in the homogenization and condensation of cross sections following the microgroup calculations make it necessary to use a large number of macrogroups in the channel cell calculation, which is naturally somewhat expensive. With 36 macrogroups we get reasonably usable results at voidages below about 50 %.

Using CASMO-HEX and the ad hoc auxiliary program CHTRAB, we have calculated parametrized representations of the group constants for use in the dynamics code TRAB. They are represented as polynomial functions of the water density, fuel temperature, graphite temperature and xenon concentration at a given initial enrichment and burnup.

Demonstrative calculations for the Chernobyl accident were made by the dynamics code TRAB. The calculations indicated that the reactivity effect caused by the mean voidage increases essentially during power excursions. The initial axial power distribution also has a very strong effect on the energy production during the power burst.

After these preliminary simulations, we have started to construct a more detailed RBMK-type plant model using the TRAB-code. The reactor-physics parameters are based on the above mentioned CASMO-HEX-calculations. Currently the new model is under testing.

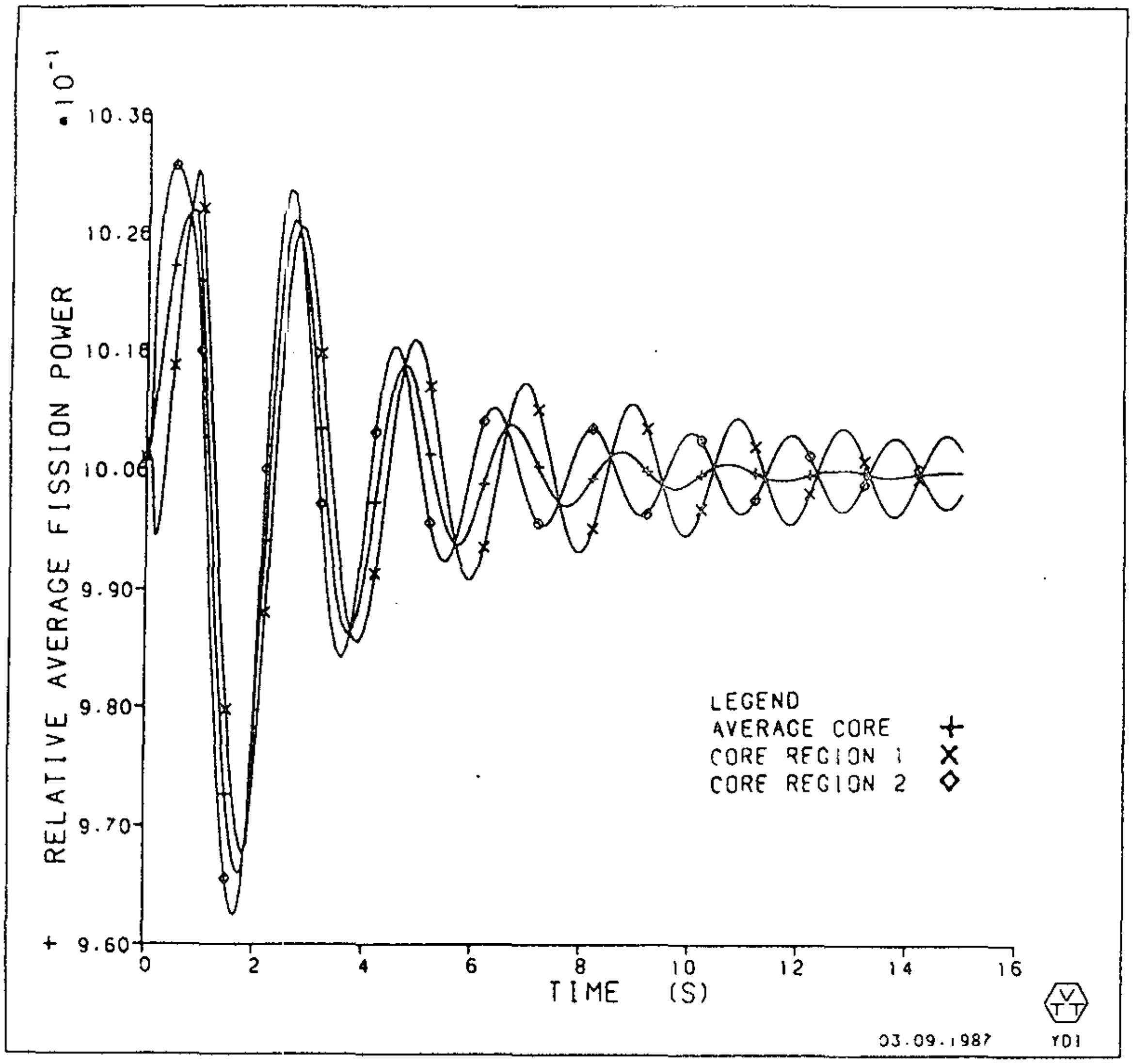

FIG. 1 . BWR STABILITY ANALYSIS BY TRAB

PHYSICS ACTIVITIES IN FRANCE

September 1986 - Deptember 1987

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Centre d'Etudes Nucléaires de Cadarache
F.13108 - SAINT PAUL LEZ DURANCE CEDEX

1 - INTRODUCTION

At the end of 1986, the state of the operational PWR units was as follows :

1 unit producing 300 MWe
33 units producing 900 MWe
9 units producing 1300 MWe

The reactors connected to the grid in 1986 and planned to be connected in 1987, 1988, 1989 are given in this table.

	1986	1987	1988	1989
900 MWe PWR	1	1	-	-
1300 MWe PWR	4	3	2	2

In 1987 the last 900 MWe has been connected (increase of power of 5 %). The number of units connected by year to grid discreases. Only one reactor will be order in 1987.

From the physics point of view 1987 is a important year for the following reasons :

- Modification of fuel management : first loading by quarter of core (Tricastin 2).

7 units of 900 MWe by year will be managed in this manner and then this mode will be extended to the 1300 MWe units.

- Loading of 16 subassemblies MOX in a 900 MWe (St Laurent B1) 12 reactors will be progressively loaded with MOX.

Moreover important studies have been undertaken by EDF to increase the margins in the 900 MWe units in order to allow new managements without discrease of capacity, availability and maneuverability (GARANCE projets).

SUPER PHENIX 1 has operated for a total $\sim$ 100 equivalent full power days. Most of the foreseen commissioning experiments (also at full power) have been completed. Some results will be given in Section 2. At the begining of April 1987 a sodium leakage of the storage tank (Barillet) was detected. The leak rate was estimated as about 20 litres per hour. The leaking sodium was retained by a double tank. While the leak position was not detected, the 330 elements in storage (including 289 dummy elements) have been discharged and a few slightly irradiated fuel elements have been loaded into the core, together with 24 new absorbers (SCP and SAC), at the periphery of the core. The storage tank and gap are being emptied.

The normal plant operation continued up to May 26, when the reactor was shut down as scheduled for various maintenance works, and for a period of at least four months.

The long term strategy is currently under discussion, together with feasibility and safety studies for temporary core operation without the storage tank (for approximately 400 equivalent full power days).

2 - FAST REACTOR PHYSICS

2.1 - Experimental studies

2.1.1 - SUPER PHENIX

The analysis of some of the neutronics experiments performed during the start-up phase of the reactor, has been the object of a number of publications / 1, 2, 3 /. The main results obtained this year concern :

- the critical mass of the first criticality core / 1 /. The excellent agreement of calculation and experiment (E-C $\approx$ 0.1 % $\Delta K/K$) was obtained / 1 / using sophisticated transport methods to allow for transport (radial and axial streaming) and heterogeneity effects related to rod followers, diluent and dummy steel subassemblies in the core. Even the inter-assembly streaming effects have been found to be not negligeable / 4 /,

- control rod worth. The $\sim$ 10 % discrepancy originally observed in now attributed to :

a) rod heterogeneity modelling problems,

b) basic data uncertainties, which have large effects on large cores flux shapes and, consequently, on rod reactivity worths,

c) uncertainties related to the MSM factors used to establish the experimental values.

Part of the results of the analysis, is reported in Ref. / 1 /.

The large sensitivities to basic data, has pointed out ouce more the difficulty of extrapolating the E/C values, observed in critical assemblies, to power reactors, if a systematic approach is lacking, to define bias factors and uncertainties (see paragraph 2.1.4).

Table I gives in a summary form the present state of the control rod worth experiment analysis.

- Reactivity coefficients have been measured and preliminary analysis has been presented / 2 /. A discussion of uncertainties associated both to experiments and calculations, has been presented at this meeting / 5 /.

- Some shielding experiments analysis has been presented in Ref. 3. A more recent result indicates an excellent agreement on the E/C value of the secondary sodium activation(see table II), which confirms the good performance of the PROPANE shielding formulaire, which includes bias factors derived from neutron propagation experiments / 3 /.

- The power distributions, measured using the subassembly temperature monitoring system, are being analysed. Simple design methods, tuned to account for, e.g. control rod experimental values, have to be improved to account properly for the power distortions due to control rod movements.

- Finally, some experiments are still to be performed, in particular fuel, fertile and diluent subassembly reactivity worth.

2.1.2 - PHENIX

An anticipated last year, the irradiated Np-237 sample analysis has provided valuable data to assess the data for the Pu-236 production. The JEF data involved in the analysis have been found to be of excellent quality (see table III).

The high burn-up fuel pin analysis experiment PAPEETE has confirmed the results obtained in the part for the actinides, and the indications obtained on the Cs isotope axial distribution in the pin, is discussed in a paper presented at this meeting / 6 /.

2.1.3 - MASURCA

The BALZAC experiments related to control rod heterogeneity have been presented and discussed in ref. 7.

The BALZAC-DE phase, devoted to γ-heating deposition in a steel-Na subassembly and to γ-heating and (n,α) reaction studies in a B_4C/steel/Na subassembly, has been completed and the analysis is underway.

The future phases of the BALZAC program will be devoted to the study of the Pu isotope (Pu-239, Pu-240, Pu-241) Am-241 and U-238 reactivity/atom (phase BALZAC-IL), followed by an internal storage simulation experiment (BALZAC-SI phase). The program is scheduled to be completed by March 1988.

2.1.4 - Future critical experiment program - The CONRAD program

The common European experiment CONRAD has been approved in the framework of the European Cooperation on Fast Reactors. This program is expected to last slightly over three years, starting from April 1988. The CONRAD program will have three phases :

- CONRAD-AX, devoted the neutronic study of an axial heterogeneous core.

- CONRAD-ST, devoted to a parametric study of basic data uncertainties on large reactor control rod worth. This point is related to the results observed at the SUPER PHENIX 1 start-up, which indicated that further studies are needed in that area. With simple radial deplacement of control-rods, a large range of sensitivities to basic data will be studied.

- CONRAD-DC. This is a very decoupled configuration in which strong interaction effects will be studied, and, among others, handling error configurations. In this configuration both reactivity worth and neutron flux monitoring system performance studies will be carried out.

2.2 - Theoretical methods

2.2.1 - ECCO Cell Code

The new European cell code ECCO is being developed. The functional specifications are now completed and some preliminary programing work has started. The code will be based on an ultrafine group treatment of the slowing-down and on the subgroup method to handle cell heterogeneities. A reference route (i.e. ultrafine group treatment for both elastic and inelastic scattering, and subgroup treatments at the ultrafine group level) will be provided, together with a "design" route in which only the elastic scattering will be treated at the ultrafine group level, and the heterogeneity self-shielding problem will be treated at the broad group level. Moreover, the code will provide a capability to generate Pn scattering matrices, based on a flux moment $\emptyset_n$ calculation. Macro-cell options will be used to treat special subassemblies or fertile regions. 1-2-3D collision probability routines will be used both in an "exact" option or in an "approximate" option, using Roth-type methods.

An important issue has been the assessment of the relative merits of the subgroup parameter generation methods. In particular, the code for the computation of the probability tables (with the statistical assumption, Cf. paragraph 3.2.5) has been modified to deal with very fine groups ($\Delta u \cong 1/120$) / 8 /. The group collapse, based on these tables, appears to be easy and rigorous.

2.2.2 - Data processing

The THEMIS code, derived from NJOY, is being extensively checked. In particular the neutron KERMA and γ-production matrices algorithms have being tested and a number of modifications have been implemented. At present, the inelastic matrices algorithms are being investigated.

2.2.3 - Equivalent cross-section definition

Extensive studies are being performed to improve the methods currently used to define equivalent cross-sections, in particular to account for heterogeneity and transport effects in special subassemblies (control rods, control rod followers, diluent etc...) but also group and mesh size effects. Two types of methods are used to establish the equivalences :

a) methods based an reactivity (or reactivity variation) conservation,

b) methods based an diffusion/transport equivalences. A particular method of this type / 9 / is being investigated to treat the large transport effects associated to control rod worths calculations. Results are very encouraging, as it is indicated in table IV / 10 /. (These results are obtained in cooperation with UKAEA-Winfrith).

2.2.4 - Basic data assessment

As part of the JEF-2 evaluation program, Cadarache is actively involved in a complete reevaluation of Pu-239. The resonance range re-evaluation in collaboration with ORNL (already reported last year) is continuing this year and has been extented to Pu-241 and U-235.

For the higher energy region, a complete $\bar{\nu}$ re-evaluation for Pu-239 has been completed (see fig. 1) and a preliminary re-evaluation of the fission cross-section above the resonance range has also been completed / 11 /. A complete re-evaluation for Pu-239 is expected by the end of 1987, including some uncertainty information.

As part of the basic data assessment, studies were performed on β_{eff} integral experiments performed at ANL, and the results are presented at this meeting / 12 /.

Decay heat data of the JEF library are being tested on integral experiments, and final results will be available by the end of 1988.

2.2.5 - γ-heating and shielding experiment analysis

The BALZAC-DE experiment analysis is being performed with sophisticated multidimensional transport methods. However, some discrepancies observed could indicate some inconsistencies in the basic γ-production data. These discrepancies are being presently analyzed in a joint effort with ENEA and Belgonucleaire.

The JASON neutron streaming experiment at HARMONIE has been analyzed with different techniques (modified diffusion, Monte Carlo, S_N transport). The results so far obtained indicate an acceptable consistency among the different methods used for the experiment analysis / 13 /.

3 - LIGHT WATER REACTOR PHYSICS

The main experimental activities have been devoted :

- to increase the fuel burn-up,
- to plutonium recycling in PWR and advanced PWR,
- to the improvement of operation, availability and safety of the 900 MWe PWR.

From the theoretical point of view, the studies concern the development of a new generation of neutronic codes (multicell code, 3D on-line code and so on).

3.1 - Experimental studies

3.1.1 - EOLE

The objective of the ERASME program is to validate the calculation of undermoderated Pu lattices.

The first experiment ERASME S was a hexagonal lattice with a ratio of moderation close to 0.5.

The second experiment ERASME R had a 0.9 ratio of moderation, and it was started at the beginning of 1986 and completed at the end of April 1987.

The standard parameters (material buckling, spectrum indices, voidage effect) have been measured.

Many configurations with clusters of absorbers water holes and fertile rods have been studied.

Boron capture effect and the spectrum in axial blanket have also been measured.

The ERASME L, now loaded in EOLE, is the third in this series of experiments. It is a lattice without driver core.

The pitch is square and the ratio of moderation is close to the one in a PWR (~ 2).

This experiment is devoted to the undermoderated lattice but also to the recycling studies in PWRs.

Material buckling, spectrum indexes, reactivity effects of absorbers, power distribution near a heterogeneity will be studied and also radial blanket effects.

The measurement of β_{eff} will be performed and measurements of in core instrumentation.

In the future, an important program for plutonium recycling will be undertaken.

3.1.2 - MINERVE

The measurement of fission product capture by using spent fuel samples has been performed in UO_2 - PuO_2 tight lattice (MORGANE/S, 0.5 ratio of moderation).

Experimental samples were issued from PWRs (corresponding to burn ups ranging from 20 000 MWd/t to 60 000 MWd/t).

A calibration with well known material worth values (^{239}Pu, ^{235}U, ^{10}Boron) was carried out.

The interpretation is in progress. The uncertainty will be less than 10 %.

A second experiment (MORGANE/R) is foreseen in 1988.

3.1.3 - MELUSINE

The GEDEON 2 experiment aims to study the $UO_2Gd_2O_3$ fuel depletion with a representative PWR environnement (interferences between water hole and poisoned rods, and between two poisoned rods).

This irradiations will last in the reactor until March 1988. The first results on fuel irradiated at 3 000 MWd/t have already been obtained.

This irradiation has been interrupted for six months in order to irradiate the ICARE/S experiment which had been finished in October 1986.

The main objectives of the ICARE experiments were to obtain the capture ratio of the different heavy isotopes (U, Pu, Am) and the capture ratio of some important fission products.

The first results give a good agreement between experiment and calculations performed with the APOLLO code and CEA 86 cross-section set.

A second experiment (ICARE/R) will be carried out in 1988.

3.1.4 - Spent fuel analyses

The program is devoted to the study of fuel irradiated from French PWRs.

The program involves analyses of fuel from the Fessenheim plant irradiated during 4 and 5 cycles respectively.

- Gravelines ($UO_2Gd_2O_3$ fuel).
- SNA (UO_2 PuO_2 fuel).

3.1.5 - Instrumentation, core surveillance

3.1.5.1 - Surveillance of vibration of in core instrumentation by noise analysis is planned this year.

3.1.5.2 - Furthermore studies on fixed in core instrumentation are continued (self powered platinum detectors, gamma thermometers). Irradiations in pile during one or more cycle were performed.

Miniature (diameter 1.5 mm) regenerative fission chambers (made of mixed ^{234}U and ^{235}U) are also studied at present. Irradiations in the Saclay research reactor OSIRIS are in progress.

3.2 - Theoretical studies

3.2.1 - Cross-sections set "CEA 86"

A new cross-section set called "CEA 86" resulting from a mixture of JEF-1, ENDFB4 and 5 or CEA evaluations, has been developed.

An important work on secondary actinides and fission products has been completed.

3.2.2 - Multicell APOLLO 2 code

This code will take the place of APOLLO 1 in 1990.

The "version 0" is realized which allows to calculate a multicell without evolution and selfshielding modules.

The "version 1" will be finished at the beginning in 1988 ; this version will have evolution at the selfshielding modules.

3.2.3 - 3D codes

At the present time, new fine mesh codes are developped (COCINELLE, CRONOS).

In paralled 3D on line codes (PHAETON, RITME) for the protection and the surveillance of PWR cores are also developped.

3.2.4 -

The modular system GOLEM which uses perturbation methods, is still in progress.

3.2.5 - Studies on residuel heat and source term

In order to understand the discrepancies existing for γ residual heat of Pu^{239} and U^{238} pulse function at cooling time around 1000 sec., a comparison of the mean energies of the most contributing nuclei in several libraries (CEA, JEF1, UK, ENDF/BV, JENDL,...) has been undertaken. The first results are that, for some nuclei like the technetium, the mean γ energies are suprisingly low.

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

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TABLE I

THE EVOLUTION OF THE E/C VALUES FOR CONTROL ROD WORTHS AT SUPER PHENIX START-UP

AVERAGE VALUE OVER MANY CONFIGURATIONS AND, (IN PARENTHESIS), THE DISPERSION

TYPE OF CORE	ORIGINAL E/C WITH SIMPLE HETEROGENEITY MODEL (1D)	E/C WITH IMPROVED HETEROGENEITY TREATMENT (XY)	E/C WITH SPECIFIC DATA MODIFICATIONS*
SUPER PHENIX	0.87 ($\pm$ 0.05)	0.92 ($\pm$ 0.04)	0.97 ($\pm$ 0.03)
RACINE-1E (MASURCA)	0.96 ($\pm$ 0.02)	0.96 ($\pm$ 0.02)	0.99 ($\pm$ 0.02)

* Main modifications to CARNAVAL-IV :

σ_{tr} and σ_{el} O-16 ($E \approx$ 300 keV) ; σ_c, σ_{el} and σ_{tr} of stainless steel ($E < 1$ MeV),

σ_{tr} U-238 ($E > 1$ MeV)

=> These modifications compensate and do not affect the critical mass prediction.

TABLE II

SECONDARY SODIUM ACTIVATION C/E COMPARISON IN PHENIX AND PRELIMINARY RESULT FOR SUPER PHENIX

	C/E
PHENIX	0.85 ± 0.60
SUPER PHENIX	1.0 ± 0.80

Both calculations are performed with the PROPANE-DO formulaire, including bias factors derived from neutron propagation experiments.

TABLE III

Reaction	$C^{(a)}/E^{(b)}$
σ_c (^{237}Np)	0.90 ± 0.05
$\sigma_{n,2n}$ (^{237}Np)	1.19 ± 0.15

(a) Calculated with JEF-1.

(b) Deduced from the mass spectrometry analysis of a pure ^{237}Np sample irradiated in PHENIX (PROFIL II experiment).

TABLE IV

Analytic Solution	Standard Diffusion	S_N transport	Diffusion with Benoist Σ_a and D	Diffusion with Rowlands modified constant (a)
561	622	563	562	628

One group calculation in XY plane geometry. Results of radial transport effects on control rod worths in a large LMFBR ($\delta(1/K) \times 10^{-5}$).

a) Nucl. Sci. Eng. 76, 263 (1980).

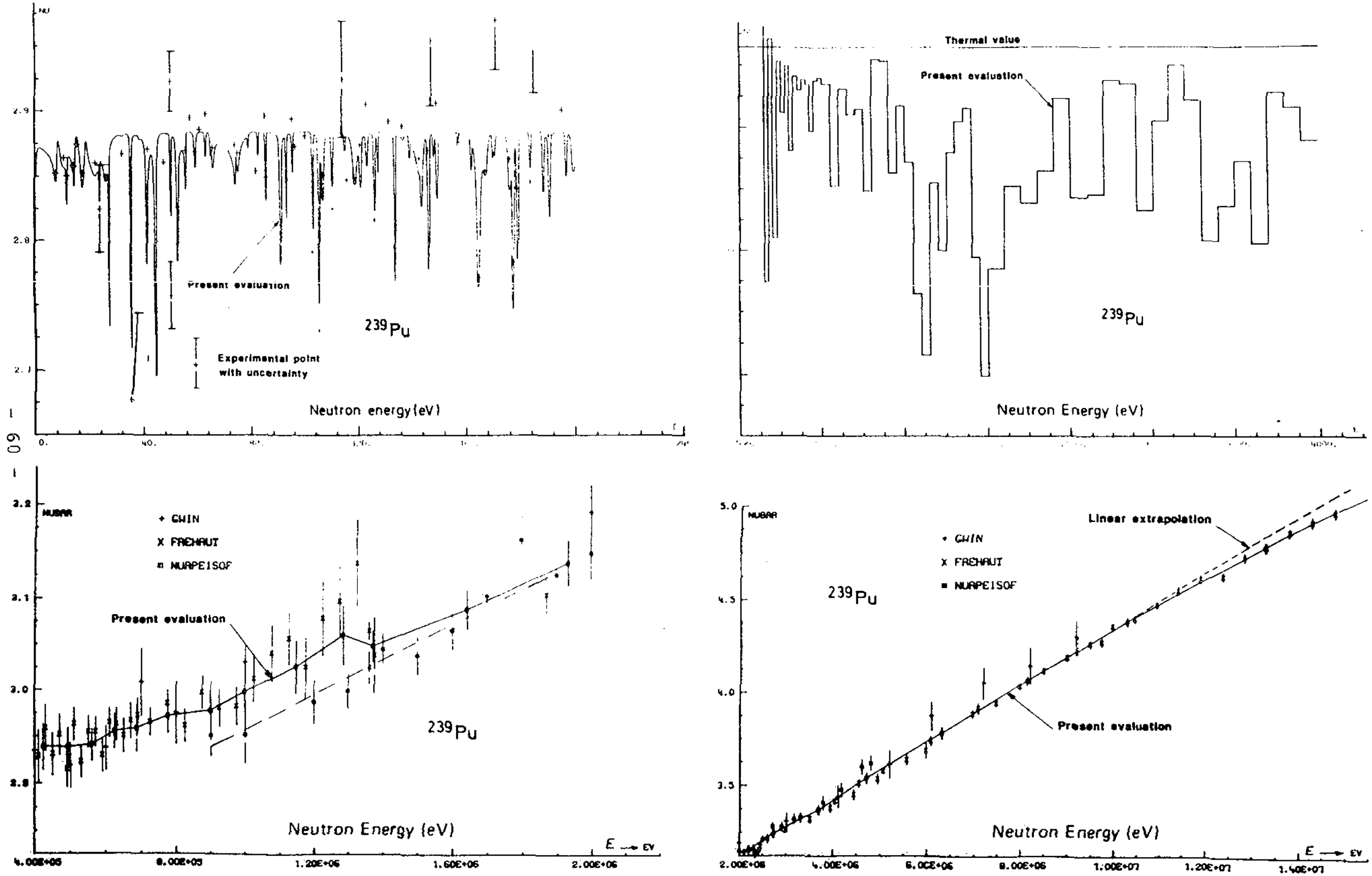
NU
Present evaluation
239Pu
Experimental point
with uncertainty
Neutron energy (eV)
Thermal value
Present evaluation
239Pu
Neutron Energy (eV)
NUBAR
+ GWIN
X FREHAUT
NUAPEISOF
Present evaluation
239Pu
Neutron Energy (eV)
E → EV
4.00E+05
8.00E+05
1.20E+06
1.60E+06
2.00E+06
NUBAR
Linear extrapolation
+ GWIN
X FREHAUT
NUAPEISOF
239Pu
Present evaluation
Neutron Energy (eV)
E → EV
2.00E+06
4.00E+06
6.00E+06
8.00E+06
1.00E+07
1.20E+07
1.40E+07

Fig. 1

REACTOR PHYSICS ACTIVITIES IN THE FEDERAL REPUBLIC OF GERMANY

Compiled by

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GENERAL

In March 1987 in the Federal Republic of Germany 21 nuclear power stations with 19 957 MWe brutto were in operation, 4 power stations with 4312 MWe are under construction, and 9 blocks with 12 106 MWe brutto are planned. The above mentioned 4 power stations are expected to go into operation in 1988/1989, 3 of them are PWRs, one is the still delayed fast prototype reactor SNR 300. For this reactor, the licensing authorities request additional information about the safety equipment of the plant.

The KWU-PWR, KBR-Brockdorf (north-west of Hamburg) with 1380 MWe is being operated commercially on full power since 22 Dec. 1986.

The availability of KWU-LWRs is very high. The power station Grohnde (KWG) with 1969 MWe reached in 1986 a value of 92,7 % time-availability.

After the Chernobyl accident an IAEO Operational Safety Review Team (OSART) is inspecting the safety features of nuclear power plants. On invitation of the Federal Ministry for Research and Development, the PWR-power plant BIBLIS and the BWR-plant KRÜMMEL were inspected by OSART: no negative aspects were found in both cases.

On April 22, 1987 a cooperation between the FRG and the USSR was signed. This cooperation deals mainly with safety features of LWRs, HTRs and FBRs; in addition, scientific exchange is foreseen for fusion research and high energy physics.

I. REACTOR PHYSICS ACTIVITIES AT THE NUCLEAR RESEARCH CENTER KARLSRUHE

1. Fast Reactor Physics

After decommissioning of the SNEAK facility, some evaluations of experiments in SNEAK 12 C and in BIZET/D were performed in the last year. The code development mainly was concentrated on the preparation of code versions to be run on a vector computer VP-50. In the data evaluation field a re-evaluation of the fission spectrum of Cf252 was performed and subsequently the fission spectra of all fissionable isotopes were determined; within the frame of JEF-2 evaluations especially the U238 capture data were critically discussed. To prepare group constants, the LANL-code system NJOY was adapted and improved to be able to generate group constants from the JEF-file.

1a. Evaluation of critical experiments

In SNEAK 12C2 the test zone was built by Al, steel, $(Pu/U)O_2$, UO_2 and Na platelets. Dissatisfying results were obtained in the C/E comparison for the radial fission rate traverses, e.g. by about 40 % for the U238 fission rate traverse. In the re-evaluation /1/ it was found that only a combination of a quasi three-dimensional transport calculation with properly prepared heterogeneous group constants could reduce the previous deviation to experiment to a value below 10 %. It should be mentioned that the results for SNEAK 12C are extremely sensitive to an accurate representation of the thin buffer and driver zones.

In the small-zone sodium-void reactivity measurements of the heterogeneous assembly BZD/1 the calculations using diffusion theory and first order perturbation theory underestimated the reactivity effects due to an overestimated leakage /2/. The largest C-E differences up to about 20 % were found in those fuel positions with sodium removed over the total core height. The use of exact perturbation theory reduced mainly the leakage terms; the deviation from experiment was then ±12 %.

1b. Support of the evaluation of SPX1 control rod experiments

It was found in 1985 that the efficiency of the SPX1 control rods was overestimated by theory by more than 10 %. A more precise modelling of the heterogeneous control rod geometry and a refined processing of the original experimental data reduced the deviation of calculational results from experiment to an acceptable level /3/.

1c. Processing of nuclear data to group constants, using NJOY

In applying NJOY for preparing group constants and selfshielding factors on the basis of JEF, many irregularities were observed, especially for the resonance selfshielding factors for heavy nuclides with high concentration. For some fission products in high concentrations even negative shielding factors occured /4/. The use of multi-level Breit-Wigner parameters from JEF-1 led to non acceptable results for the selfshielding factors for some nuclides. These irregularities were removed, most of them only formally /5/.

1d. Computer codes

On the basis of two-dimensional neutron transport perturbation theory in triangular geometry a Fortran 77 program for the calculation of reactivity, effective neutron fractions and mean generation times was established in connection with the corresponding S_N-code DIAMANT /6/. A major revision of the DIAMANT code led to DIAMANT-2, which solves the static multigroup neutron transport equation in planar geometry using the S_N-method. Spatial discretization is accomplished by taking finite differences on a meshgrid composed of equilateral triangles /7/. This version is designed to run on scalar and vector computers.

2. Safety investigations

The present international program for the study of transient phenomena in the CABRI reactor was completed. The evaluation of the experiments is still underway.

Following parameters, affecting the sequence of accidents, were studied in the test program:

- The energy added to the fuel during an excursion (variation between 0.4 and 2 kJ/g of oxide).
- Burnup (fresh fuel, low and medium burnup).
- The time of initiation of the power excursion relative to the onset of sodium boiling in the cooling channel (before, during and after the onset of boiling) in experiments conducted at a reduced coolant flow.

The results of the CABRI test program have greatly added to the knowledge of the phenomena associated with accidents. They have supplied extensive measurement data for further advancement and verification of models to be used in the safety analysis of fast breeder reactors. As a very important result, the reactivity feedback of the axial thermal fuel expansion and of the fission gas driven fuel dispersion in fast reactor accident situations could be clarified.

A mixture flow model has been developed to describe the two-phase flow in a sodium boiling model. Thermodynamic equilibrium along the saturation line is assumed while slip correlations are used to describe the different velocities of the vapor and liquid phase. The model has been tested against a TREAT experiment and the results were found encouraging. As the code has a multichannel structure, sodium boiling is treated in a two-dimensional way /8/.

A new approximation for calculating the pressure gradient term in the SIMMER-II momentum equations of a steady state flow of a single-phase and incompressible fluid across a singularity, has been developed. This new code has been tested against experiments performed at SRI-International and the results are in a better agreement with experiment than the results of the original SIMMER-II code /9/.

3. Advanced pressurized water reactors

The investigations to evaluate the physics characteristics of a tight lattice advanced PWR (APWR) were continued. In cooperation with EIR-Würenlingen (Switzerland), KWU and KfK experiments in the PROTEUS-II facility were performed and analysed. Reaction rate and material worth measurements were conducted in a moderated test lattice and in a lattice without moderator. The evaluation of these experiments and the comparison with theory is given in a separate paper to this meeting.

KfK participated in the NEACRP benchmark on tight lattice configurations. The results, together with those from other contributors, are also given in a separate paper to this meeting. KfK performed some special investigations on this benchmark by using the JEF-1 group set with a proper APWR weighting spectrum for the normal and a weighting spectrum for the voided configuration. These new group sets were also applied to a PROTEUS experiment to judge the quality of the used theoretical procedures and nuclear data sets. This intercomparison is given in a separate note /10/. Up to now it can be concluded that the JEF-1 set with appropriate weighting spectra gives satisfactory resulst in comparison with experiment; the accurate determination of burnup effects probably needs improvement of the Pu240 and Pu241 nuclear data (see section 4).

The theoretical investigations are performed with the KfK-KARBUS code system. The calculational procedure was improved by using isotope dependent fission spectra, and by improving the various calculational modules in the frame of the Karlsruhe program system KAPROS.

Fluiddynamic investigations for a large APWR reactor showed that the homogeneous and heterogeneous core designs are almost equivalent /11/. Both physics and fluiddynamic studies strongly indicate that the tight lattice should be widened to guarantee a negative void reactivity coefficient for all burnup stages up to 70 GWd/t and to guarantee the desired reactor power as well as an acceptable reactor behaviour under transient conditions.

4. Fuel cycle analysis for PWRs

As already mentioned in NEACRP-L-270 (1986), a comparison between theory and experiment for the activation of PWR-fuel element end-pieces showed a fairly good agreement; the results are now published in /12/. Furtheron, the theoretical assessment of C14 formation in a PWR fuel pin was compared with experimental results /13/. C14 in LWR-fuel strongly depends on the nitrogen contamination in the fuel. New data sets for the calculation of the C14 build-up via ternary fission and via O17 (n,α) C14 were prepared. The theoretical results are in a satisfactory agreement with experiments.

For the test of the JEF-1 nuclear data library, a complete new group set for PWR application was applied to the NEACRP "Benchmark on Recycling of Reprocessed Uranium". The study was performed by single replacemens of the JEF-1 group data for important nuclides into the KEDAK-4 group library. The impact of these single replacements on k_∞ and the EOL-nuclide concentrations was studied and compared with the results of the KEDAK-4 group set and the various contributions to the benchmark. The k_∞-values, obtained with the complete JEF-1 group set almost coincides with the UK-contribution, i.e. JEF-1 gives results which are at the upper boundary of all contributions /4/.

In the application of JEF-1 data to a KWO-PWR configuration, the nuclide concentrations at 30 GWd/t burnup were compared with experimental results from postirradiation analyses of KWO-fuel. Good agreement was found for U236, Pu238, Pu239, Am241 and Am243. Slight differences were found for U235, major differences for Pu240, Pu241, Pu242, and Cm242 /4/. Especially the JEF-1 data for Pu240 and Pu241 should be re-evaluated; this is underway within the evaluation effort for JEF-2.

5. Investigations on fusion reactor blankets

Two design concepts are studied by KfK: a helium cooled ceramic blanket and a blanket with Pb-17Li eutectic as breeder material and coolant. The study includes small scale experiments and collaboration with industry for special feasibility problems. The studies are coordinated with efforts of CEA and UKAEA in a common working group. As an example, Fig. 1 gives an artists view of the new outboard blanket canister design /14/.

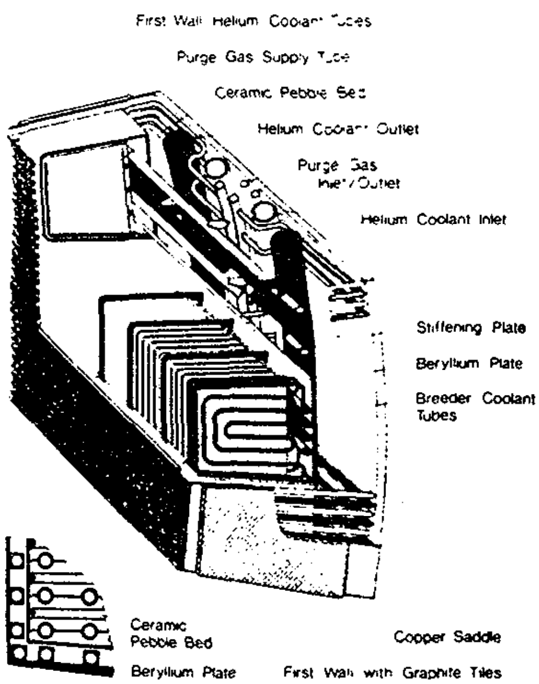

Fig. 1: Pebble Bed Breeder Canisters

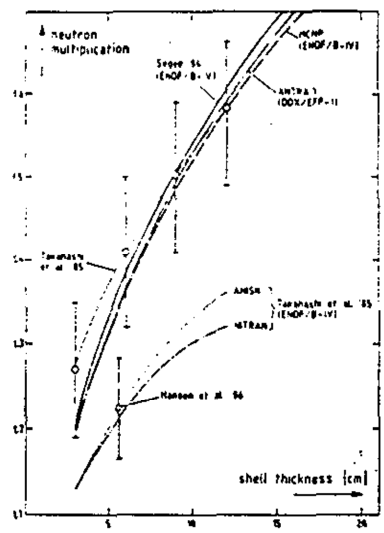

Fig. 2: Neutron multiplication in lead: a comparison between calculated values using different procedures and data and experimental values.

One of the most important problems in fusion technology concerns the use of a water cooled first wall. Potential and problems of an aqueous lithium salt solution are discussed in /15/.

The development of computational tools for neutronic calculations comprises the evaluation and processing of nuclear data, the development of new techniques, as the adoption and application of existing methods and programmes (e.g. Monte Carlo methods) for use in fusion neutronics.

For a rigorous treatment of the anisotropic neutron scattering, a general transport code system GANTRAS, making full use of double-differential cross-sections (DDX), is under development. The one-dimensional module, ANTRA 1, treating plane and spherical geometry, already had been set up and documented /16/. Furthermore, ANTRA 1 has been extended to include the cylindrical geometry option. The introduction of a new interpolation scheme for the angular fluxes, needed for the calculation of the scattering source, also necessitated the introduction of a new quadrature set being more appropriate for the applied procedure. The cylindrical geometrical option has been tested numerically and is now applied to cases of interest.

A comparative study of the neutron multiplication of a 14 MeV neutron source in a spherical lead assembly has been performed. For this purpose, the ANTRA 1 programme has been used, together with a newly developed processing system for the provision of the angular-dependent transfer matrices, generated from the lead single- and double-differential data contained on the European Fusion File EFF-1. It is seen from Fig. 2 that, for the same geometrical assembly, there is no significant difference when using different procedures and data. On the other hand, it is concluded from this comparison, that the experimental values by Takahashi et al are still too high, because the calculations refer to an ideal spherical assembly. Taking into account the real geometry (beam channels, target etc.), however, there will be a signigicant decrease of the neutron multiplication. This is also confirmed by the experiment of Hansen et al. (see Fig. 2).

Monte Carlo methods have been proven to be very powerful for treating complex geometrical arragements in fusion reactor blankets in two and three dimensions. The Los Alamos code MCNP has been used for analyzing the ceramic "pebble bed canister blanket" /17/. A true geometrical model of a torus sector without any idealizing approximations has been used.

The nuclear data describing the energy release are not very reliable in the MCNP library. Therefore, the efforts to improve the data base of MCNP have been intensified.

For the calculation of the activation and afterheat of fusion reactor blanket components the DKR activation code, developed at the University of Wisconsin, together with its data library based on ENDF/B-IV, has been implemented at KfK (Siemens 7890 computer). DKR has been coupled with the ONETRAN transport code in a twofold way: ONETRAN provides the spatially varying neutron spectra for DKR and finally it performs the transport calculation for the photons created by the radioactive decay of the activation products. The source strength of the decay photons is calculated by DKR. In a first step this procedure already has been applied to a NET blanket design. At present this analyis is deepened.

The Karlsruhe neutron transmission experiment KANT to study the neutron multiplication and neutron leakage in concentric shells of Pb and Be, very recently started operation.

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/1/ G. Henneges, U. v.Möllendorf: Über die Auswertung von radialen Spaltratentraversen in SNEAK 12C, Private communication, 1987.

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/3/ H. Giese: A comparison of BN and CEA input data for SPX-1 control rod worth calculations, Private Communication, 1987

/4/ A. Mateeva, H. Küsters: Testing of JEF-1 Data on the Burnup Behaviour of PWR-Fuel up to 33 GWd/t, JEF-DOC, 1987.

/5/ B. Krieg, I. Broeders: The KfK-Version of the Neutron and Photon Cross Section Generation Code NJOY, Implementation, Testing, and Modifications of the Code, JEF-DOC, 1987.

/6/ K. Kobayashi, G. Buckel, K. Küfner: TPTRIA, A Computer Program for the Reactivity and Kinetic Parameters for Two-Dimensional Triangular Geometry by Transport Perturbation Theory, KfK-Report 4116, October 1986.

/7/ K. Küfner, J. Burkhard, R. Heger: An updated FORTRAN-77 Version of the 2D Static Neutron Transport Code DIAMANT 2 for Regular Triangular Geometry, KfK-Report 4133, June 1987.

/8/ P.R. Henkel: The SANDCMOT Sodium Boiling Model, KfK-Report 4256, 1987.

/9/ J.J. Foit: Analysis of Expansion Phase Experiments with Improved Approximation Schemes, KfK-Report 4245, 1987.

/10/ C. Broeders, A. Mateeva, H. Küsters: Some Special Methodical and Nuclear Data Investigations for Tight Lattice PWR Cores, 1987, Separate Note to this NEACRP Meeting.

/11/ M. Cigarini, M. Dalle Donne: Parametric Thermohydraulic Calculations for the Advanced Pressurized Water Reactor with the Code HADA-2, KfK-Report 4148, October 1986.

/12/ U. Fischer: Aktivierung von Endstücken eines DWR-Brennelementes, Atomwirtschaft/Atomtechnik, Heft 1, Januar 1987.

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/14/ Nuclear Fusion Project, Semi-annual Report of the Association KfK/Euratom, October 1986 - March 1987, KfK-Report 4276, 1987.

/15/ M. Küchle, E. Bojarski, S. Dorner, U. Fischer, J. Reimann, H. Reiser: Potential and Problems of an Aqueous Lithium Salt Solution Blanket for NET, KfK-Report 4271, 1987.

/16/ A. Schwenk-Ferrero: GANTRAS, A System of Codes for the Solution of the Multigroup Transport Equation with a Rigorous Treatment of Anisotropic Neutron Scattering-Plane and Spherical Geometry, KfK-Report 4163, 1986.

/17/ U. Fischer: Multi-dimensional Neutronics Analysis of the "Canister Blanket" for NET, KfK-Report 4255, 1987.

II. REACTOR PHYSICS ACTIVITIES AT KRAFTWERK UNION

1. Nodal Diffusion and Transport Theory Methods for Fast Reactors

The development of the three-dimensional nodal diffusion and transport theory program HEXNOD /1/ for fast reactors is continuing. The low order nodal transport theory method was extended to solve 3-D fast reactor problems with a uniform mesh of hexagonal prisms. The method assumes isotropic scattering and is based on the nodal discrete-ordinates method (NDOM) which was developed earlier /2/ for LWRs in Cartesian geometry. A response matrix technique is used for the efficient solution of large-scale 3-D fast reactor problems. The computing times of nodal transport theory calculations are small and only a factor of two greater than for corresponding nodal diffusion theory calculations of the same problem, provided that sufficient memory space for storing the response matrix elements is available.

For the 3D SNR Benchmark Problem /3/ excellent accuracy is obtained in diffusion theory compared to accurate reference solutions. The HEXNOD results with 8 axial layers are of comparable accuracy as is obtained with the finite difference code DIF3D using a mesh of 24 triangles per hexagon and 36 axial planes, while the computing time of HEXNOD is smaller by a factor of about 25. Due to the lack of a 3D transport theory reference solution a similar comparison of HEXNOD was made with a conventional discrete ordinate S_4 calculation for two related 2D SNR Benchmark Problems /3/. In both diffusion theory and transport calculations HEXNOD computes the average assembly power densities with a maximum relative error smaller than 0.4 % in the core and 2 % in the outermost row of blanket elements.

The development of HEXNOD is in part supported by the European Communities /4/. In a joint effort of JRC Ispra, University of Brussels and KWU the code is currently being implemented as the neutronics module of the generalized quasistatic space-time kinetics program CASSANDRE /5/ which will later be incorporated as part of the European Accident Code EAC.

/1/ NEACRP-L-295, Reactor Physics Activities in NEA Member Countries, p. 64 (1986)

/2/ M. R. Wagner and B. Müller, The Nodal Discrete Ordinates Method and its Application to LWR Lattice Problems, Proceedings of ANS Topical Meeting, Chicago, Sept. 17-19, 1984, Vol. 1, p. 376

/3/ ANL-7416, Benchmark Problem Book, Supplement 3, p. 871 ff, Argonne National Laboratory, Argonne Illinois, USA (1985)

/4/ NEACRP-L-295, p. 165 (1986), see also this report: Reactor Physics Activities at the JRC-Ispra.

/5/ J. Devooght, E.H. Mund, B. Arien, A. Siebertz, Fast Reactor Transient Analysis Using the Generalized Quasi-Static Approximation, Nucl. Sci. Eng., 88, 191 (1984)

2. Parallel Solution of the Neutron Diffusion Equation on Large MIMD Systems

The numerical solution of the neutron diffusion equation plays an important role in the simulation of a nuclear power plant. Highest safety requirements ask for a permanently growing accuracy of calculations. Thus efficient numerical algorithms together with adequate computer architectures are required to solve these problems. The neutronics and the thermohydraulics of the reactor core are described by a coupled system of partial differential equations. The stationary solution of the neutron diffusion equation as an eigenvalue problem presented in /1/, /2/ results in a nonlinear partial differential equation system, which is discretized and solved by an iterative algorithm. Thus, a large amount of data is generated, which, in turn necessitates powerful computers with large storage capacity. As a consequence, the possibilities for implementing the existing programs on conventional computers are limited. At the present stage of scientific development we can make use of

innovative hardware	-	Vector computers
	-	Multiprocessor systems and
improved algorithms	-	Multigrid methods

to reduce running time. In order to prove that a reasonable combination of these hardware concepts and numerical techniques yields very fast and efficient programs the iterative solution of the neutron-diffusion-equation was parallelized and implemented on the multiprocessor system DIRMU 25 (Distributed Reconfigurable Multiprocessor kit) of the University of Erlangen-Nuremberg. The DIRMU 25 kit consists of 25 identical microprocessor modules which can be coupled by way of multiport memories. The neighbourhood relationships of this memory-coupled system are achieved by simply plugging in the respective cables. Thus, there is a great choice for forming multiprocessor configurations (ring, cube, array etc.)

For the acceleration of the solution of the neutron diffusion equation both multiplicative and conventional additive multigrid methods were used. Two different techniques to map the problem on multiprocessor configurations were examined. First, according to a given problem size, the most adequate DIRMU-configuration depending on the number of available processors was chosen. As a result, we could show that the solution of the problem can efficiently be parallelized and implemented for the existing hardware /1/. To achieve predictions for larger systems the problem size then was adapted to the number of processors. Thus the efficiency of the parallel program in case of an optimal number of processors could be evaluated and extrapolated to an assumed and higher number of processors /2/. Based on these results it is possible to predict a very high efficiency of the parallel programs for large MIMD systems (even with more than 1000 processors), provided the problem is sufficiently large and the processors are equipped with powerful (vector)-coprocessors (MSIMD-architectures).

/1/ H. Finnemann, J. Volkert:
Parallel Multigrid Algorithms Implemented on Memory-Coupled Multiprocessors, Int. Top. Meeting on Advances in Reactor Physics Math. and Comp., Paris 1987

/2/ H. Finnemann, J. Brehm, E. Michel, J. Volkert:
Solution of the Neutron Diffusion Equation through Multigrid Methods Implemented on a 25-Processor-System Proc. Conf. on Vector and Parallel Processors in Comp. Science III, Liverpool 25th to 28th Aug. 1987

3. Status of Investigations related to Thermal Recycle of Plutonium and Uranium

In the Fed. Rep. of Germany the technical feasibility and economic use of reprocessed plutonium have been proved primarily for PWRs. The programme is scheduled to be extended to include BWRs in order to increase the recycling capacity. No technical problems have arisen in the design and use of reprocessed uranium after enrichment as ERU (= enriched reprocessed uranium) fuel assemblies. Strategy and current status of this recycling programme was presented in /1/ and /2/. There is also a connection to the investigations on thorium use in PWRs /3/, where plutonium is proposed as initial and make up fissile material.

For the validation of the nuclear design methods and data all experimental data of the recycling demonstration programmes in the reactors KWO, GKN-1, KKU and KKG/BAG and special irradiation experiments are used /1, 2, 4, 5/. The insertion of MOX and ERU fuel assemblies shows that thermal recycling is feasible on an industrial scale and that common levels of reliability and safety can be achieved in reactor operation.

/1/ G. J. Schlosser, S. Winnik:
Thermal Recycle of Plutonium and Uranium in the Federal Republic of Germany, Strategy and Present Status
IAEA-SM-294/33

/2/ Horst Roepenack, Fritz U. Schlemmer, Gerhard Schlosser:
Development of Thermal Plutonium Recycling
Nuclear Technology <u>77</u>, p. 175 (1987)

/3/ M. Peehs, G. Schlosser:
Prospects of Thorium-Fuel Cycles in a Standard Pressurized Water Reactor
Siemens Forsch.- und Entwickl.-Bericht <u>15</u>, p. 199 (1986)

/4/ D. Porsch, G. Schlosser, W. Stach, M. Zimmermann:
Überwachung der Leistung von MOX-Brennelementen in DWR: Methoden und Verifikation
Jahrestagung Kerntechnik '87, p. 47 (1987)

/5/ W. Goll, D. Porsch, F. Schlemmer:
Betriebsverhalten von Brennstäben mit Wiederangereichertem Uran (WAU 1)
Jahrestagung Kerntechnik '87, p. 421 (1987)

4. KHCR, an Advanced PWR with Tight Lattice

The high commercial success of current PWR-Nuclear Power Stations is an incentive to carry this technology even further. Among the goals for future plants, great importance is placed on significantly improving fuel utilization and energy autonomy: burning plutonium in standard light water reactors can improve fuel utilization only marginally and the commercialization of the fast breeder reactor may be delayed into the next century.

Thus various other measures (e. g. better reflectors, spectral shift displacement rods, low-leakage management strategies) have been proposed to improve fuel utilization in present PWRs on the basis of a standard fuel rod lattice. But even if applied in combination, they will not bring an ore savings effect of more than 20 %.

The KWU High Converter Reactor (KHCR), which is the natural development of the Convoy PWR can increase the fissile material recovery rate to values of 0.9 and more, drawing to the largest extent possible on the proven technology in hand, being in line and compatible with the ultimate goal of breeder development and allowing a high degree of adaptation in case of the energy scenario changing form low to high growth /1/.

Our actual design basis to gain values nearby the optimum is the socalled "homogeneous core", with a very tight lattice as a yardstick (the essential geometrical parameter p/d being 1.12, corresponding to V_M/V_F 0.5) and a fissile material concentration of app. 7.5 % Pu. Since a Pu fuelled tight lattice renders the physics characteristics intermediate between those of common LWRs and FBRs, for increasing safety margins expecially with respect to the void coefficient a broad experimental data base is actually prepared /2/. Although there are still a lot of detailled aspects to be checked, results achieved so far showed that all important design problems currently under investigation can be solved, and thus the gradual realization of a KHCR can be guaranteed.

/1/ to be published ANS-Winter-Meeting 1987

/2/ R. Chawla, F.J. Erbacher, H. Moldaschl
Achieving high conversion
Nucl. Eng. International, June 1987, P. 62

5. Neutron Physical Performance of 9x9-9Q Fuel Assemblies for BWRs

The neutron physical performance of 9x9 fuel assemblies with a large internal water channel, that replaces 3x3 fuel rods of the 9x9 array (9x9-9Q) was analysed /1/. The design meets the demands inposed on advanced BWR fuel.

Developed on the basis of the proven 9x9 concept it is characterized by large thermal margins and - due to the reoptimization of the moderator distribution - by a large hot-cold reactivity difference, two features that are especially useful if extended load following operation and therefore high operation flexibility is desired. Furthermore, when compared with standard BWR 8x8-2 fuel, it exhibits significantly higher discharge burnup at equal enrichment and smaller void feedback. The 9x9-9Q design therefore offers considerable potential for further enrichment and burnup increase.

Lead test assemblies have been inserted since 1986 in the KKK-plant, where complete reloads are scheduled from 1988 on. Additional 9x9-9Q assemblies will be supplied in 1987 and 1988 to at least four more BWRs.

/1/ D. Bender, O. Bender, P. Urban
Boiling water reactor reload fuel for high burnup:
9x9 with internal water channel
Kerntechnik 50 (1987) No. 4, pp 222-226

6. Thermal Hydraulic Tests of 9x9-9Q Fuel Assembly

In order to determine the thermal hydraulic characteristics of the advanced 9x9 fuel design various test series were performed in the ATLAS-Loop of General Electric Co. using full scale test bundles with a large squared water channel replacing the 9 central fuel rod positions. Experimental conditions were chosen in such a way that the integrated water structure series matched the earlier 9x9 tests with 1, 5 and 9 water rods.
Single and two phase (steam-water) pressure drop measurements were taken in vertical upflow tests of 72 electrically heated rods simulating fuel bundles with modified 9x9 spacers. The measured pressure drop values which could be well predicted by design methods were comparable with those of 8x8 bundles thereby confirming the hydraulic compatibility of both fuel types in a mixed core.

The critical power tests confirmed the expected improvement of the modified 9x9 spacer design. The results were analyzed using the XL critical power correlation /1/ which was developed on the basis of 9x9-1 measurement. The analysis showed that the data could be predicted with a standard deviation of 1.1 %.

/1/ W. Kraemer, W. Uebelhack, G. Preusche, M. Schrader: Anwendung der thermohydraulischen Analysenmethode THAM auf Siedewasserreaktorkerne mit 9x9 Brennelementen. Jahrestagung Kerntechnik 1982, Conf. Rep., pp. 85-88

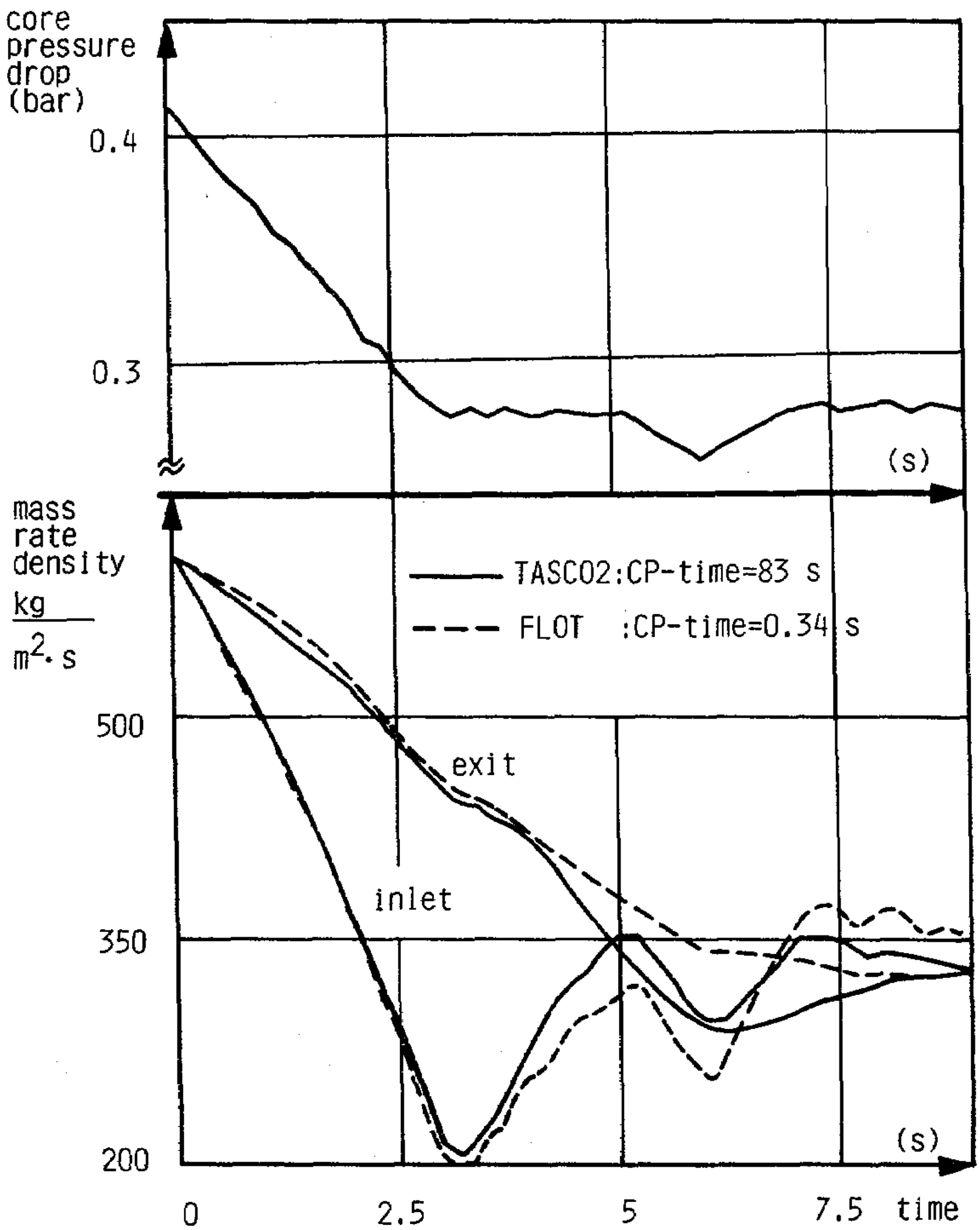
7. A BWR Fuel Channel Model with Low Calculation Time and Memory Requirements

This model, called FLOT, was developed as a module of a BWR-training simulator /1, 2/, which has to operate in real time on a small computer demonstrating even 3D effects occuring in the reactor core.
Additionally it is an appropriate tool for 3D-stability calculations in the time domain, just if the regional behaviour has to be studied and nearly every fuel channel has to be taken into account.
In both cases a high calculation speed combined with low memory requirements are necessary conditions. This could be reached by introducing empirical simplifications for the void distribution. Integrating the conservation equations in the single and two phase regions separately ordinary differential equations for the mass rate, the boiling boundary and the steam quality at the channel exit can be obtained /3/.
Due to this method the computation time was reduced by a factor of 200 compared with the design code TASCO2 without loosing a significant amount of accuracy, as can be seen in the figure.

/1/ Reactor Physics Activities in NEA Member Countries
NEACRP-L-295 (1986), p. 67

/2/ H. Finnemann and H. Raum
Nodal Expansion Method for the Analysis of Space-Time-Effects in LWR's, Proceedings of Spec. Meet. on the Calc. of 3D Rating Distrib. in Operating Reactors
November 1979

/3/ F. Wehle, D. Kreuter and M. Schrader
Kanalstabilität von BWR-Brennelementen
Siemens Forschungs- und Entwicklungs-Bericht, Bd. 14 (1985)



Comparison of TASCO2 and FLOT - calculation for a pressure drop driven transient

III. REACTOR PHYSICS ACTIVITIES AT INTERATOM

1. Computational Methods

The threedimensional coarse-mesh code DEGEN with mesh-internal cubic flux approximation has been qualified by comparing its results with those of the finite difference code D3E. The comparison has shown that the deviations with respect to k_{eff} and absorber worth predictions are smaller than the normal mesh-size effects of D3E. The power distribution within the core can be predicted with DEGEN with an accuracy of ± 2,5 % whereas in the radial blanket somewhat larger deviations of up to 10 % do appear which need some further investigations. In general, it was concluded that the errors caused by the coarse-mesh method are much smaller than the normal computational uncertainties. On the other hand, the reduced computer time of DEGEN (factor of 3 to 10 depending on the mesh-size used in the finite difference code) and its pronounced flexibility have made this code to a valuable tool for the design of reactor cores.

2. Global Reactor Analysis

The problem of flux and power tilting due to an unintentional withdrawal of an absorber has been investigated for SNR 2. Due to the pronounced decoupling effects in large cores with a fissile zone diameter of about 4 m, these effects have much more severe consequences than in smaller cores like SNR-300. In case of the withdrawal of one single absorber out of the whole bank of the primary shutdown absorber the level of maximum linear rating increases by about 50 % and thus reaches the level of local fuel melting. In case of two neighbouring withdrawn absorbers, the linear power is increased by up to a factor of two and the level of fuel melting is exceeded in a large number of fuel S/As. In order to initiate reliable measures against these unacceptable perturbations, the flux monitoring and coolant outlet temperature measurements shall be used as independent and diverse systems. It is still being investigated whether it is possible to avoid reactor shutdown in case of such perturbations by identifying the withdrawing rods and initiating special measures.

With the goal of an optimization of the fuel element management of SNR 2, two possible management policies have been compared: a scatter loading with the fuel subassemblies being loaded at various positions throughout the core where they remain during their whole life and a zone loading where fresh fuel is loaded into a certain zone, burnt fuel is unloaded from another one and inwards shuffling of partially burnt subassemblies from one position to another is necessary. Threedimensional power and burnup calculations have shown that in case of the management scheme with shuffling the level of the linear rating can be reduced by about 10 % and the maximum displacements and fuel burnups are smaller by about 10 % for the same average discharge burnup. This management scheme has therefore been chosen as reference for SNR 2.

3. Evaluation of Fast Critical Experiments

The threedimensional transport code MOCA has been used for the evaluation of the critical configurations and absorber worth measurements in SUPERPHENIX-1. The ratios of experimental to calculated values (E/C) are

E/C (critical mass) = 0.992 ± 0.002 (2σ)
E/C (absorber worth) = 1.05 ± 0.03 (2σ)

They fit very well with the experiences gained over the past 15 years in a large number of zero power experiments. The average E/C-values are

E/C (critical mass) = 0.991 ± 0.005 (2σ)
E/C (absorber worth) = 1.06 ± 0.05 (2σ)

Due to the good consistency of the E/C-values of the critical experiments and the large power reactor SPX-1, the reactivity values calculated for SNR 2 can be considered to be well proven.

The evaluation of the fission and capture rate measurements of SPX-1 is in preparation.

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

1 Development of the Code ICM2D

(Th. Rückle, D. Emendörfer)

For multigroup assembly calculations in xy-geometry a fast transport code has been developed based on an interface current method (ICM) with consistent mean partial current coupling.

Transmission-, collision- and escape-probabilities are calculated with linear space- and angular-dependent neutron surface distributions and volume sources. The admixture coefficients for the linear space- and angular-dependent correction terms are determined in first approximation by the mean partial surface current densities of the incoming and outgoing neutrons and the mean neutron flux density in the node. In this approach we have the same number of equations and unknowns as in the classical $J^{\pm}$-method (with constant source- and isotropic angular-distribution), but with different and additional matrix elements, accounting for a higher degree of transport information. Linear anisotropic scattering is included. The multiple integrals of the different components of the linear space- and angular-dependent transmission- and escape-probabilities are reduced analytically to single integrals, which can be evaluated fast by Gauß-Integration and tabulated as functions of the total cross-section. The collision probabilities follow from the transmission- and escape-probabilities by the respective neutron balance equations.

The ICM2D equations are solved by outer and inner iterations. The direction of the inner iterations changes alternately. The outer iterations are accelerated regarding upscattering in the thermal groups.

A 45-group ICM2D-calculation of power distribution and eigenvalue for a 14x14 PWR fuel assembly with 4 Gd-rods takes 13 s on CRAY-2. The results agree well with a S_8-calculation of 3x3-subdivision of the nodes.

/1/ Rückle, Th.; Emendörfer, D.:
ICM2D, Ein Programm zur Lösung von Vielgruppen-Neutronentransportproblemen in (x,y)-Geometrie nach dem $J^{\pm}$-Konzept mit konsistenter Partialstromkopplung.
Stuttgart: IKE, 1987 (IKE 6-172)

2 Coupling of Diffusion- and Transport Theory for 3D-Applications

(W. Bernnat, F.A.R. Schmidt)

The 3D-treatment of neutron streaming in cavities e.g. in the cavity between pebble bed core and top reflector of a HTR is realized using a 3D-finite element method /1/ for non-cavity regions and a 3D-response matrix method for cavity regions in which control rods may be inserted /2/. The cavity region and the non-cavity regions are coupled by means of internal boundary conditions which are depending from the flux and its space-derivatives at the cavity surface: the currents going out of the cavity at the cavity surface elements can be calculated by multiplying the incoming currents with corresponding response matrix elements and integrating over the surface. The application of this method is planned for the calculation of reactivity effects due to movement of control rods in the cavity of the HTR-500.

/1/ Schmidt, F.A.R.; Fremd, R.; Wörner, D.:
DIFGEN - A Program Package for the Solution of the Diffusion Equation by the Finite Element Method.
Stuttgart: IKE, 1978 (IKE 4-75)

/2/ Neumann, K.; Bernnat, W.; Emendörfer, D.:
Treatment of Neutron Streaming through Cavities in S_N-Transport and Diffusion Calculations.
Intern. Topical Meeting on Advances in Reactor Physics, Mathematics and Computation, Paris 1987

3 Dynamical Behaviour of a Graphite-Moderated Boiling Water Reactor (RBMK-1000) in the Point Kinetics Approximation

(B. Lukas, D. Emendörfer)

After the power excursion in the Chernobyl RBMK-1000 a sensitivity analysis of parameters describing the dynamical behaviour of BWR with different feedback-features has been carried out. The simulation model consists of five basic equations covering neutron kinetics and thermal hydraulics:

- integro differential equation for the power as a function of reactivity including 6 classes of delayed neutrons,
- reactivity as a function of fuel temperature, steam void and control rod position,
- differential equation for the mean fuel temperature,
- differential equation for the height of the single phase region in the coolant channel,
- differential equation for the mean volumetric steam content in the core.

With this model reactor dynamics has been investigated for perturbations in coolant mass flow and external reactivity. Parameters are:

- void reactivity coefficient,
- initial power level,
- heat transfer with and without film boiling.

/1/ Lukas, B.; Emendörfer, D.:
Dynamisches Verhalten eines graphitmoderierten Siedewasserreaktors (RBMK-1000) nach einem Punktmodell.
Stuttgart: IKE, 1987 (IKE 6-173)

4 Monte Carlo-Calculations for the Activation Rates in Incore Detectors of KWO-Reactor

(W. Bernnat, D. Lutz)

For the calibration of the aeroball incore detectors in a power reactor the flux distribution near detectors has to be well-known. The usual calculational method, the superposition of the coarse mesh fluxes and the heterogeneous assembly fluxes lead to inaccurate results in MOX assemblies and their neighbourhood and near the reflector. For this reason a Monte Carlo-calculation has been performed for 1/8 section of the KWO-reactor (PWR) with a fully heterogeneous representation of the 2 included MOX assemblies and 9 homogeneous meshes for each Uranium assembly. The 45 group cross-sections have been calculated using the local distribution of burn-up, Xenon density, fuel temperature and moderator density resulting from a 2-dimensional coarse mesh calculation with 2 group cross-sections. The differences of the calculated activation rates for the 8 included incore detectors and the axial integrated measurement results are between 0 and 5 %.

5 Use of JEF-1 Data for Reactor Physics Problems

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

JEF-1 data were used for a large number of thermal benchmarks /1/ and - more and more - also for applications in PWR-cycle calculations, in criticality safety calculations and for research reactors. For many different nuclear systems reliable results could be achieved compared with corresponding experiments or results derived from ENDF/B-V. The largest differences were found for homogeneous Pu-systems, for which k_{eff} was overestimated up to 1.8 %. For uranium systems very good results were found for both homogeneous solutions and for wide and tight LWR-lattices.

/1/ Bernnat, W.; Mattes, M.; et al.:
Analysis of Critical Experiments for the Validation of JEF-1 Data in Thermal Reactor Applications.
Intern. Topical Meeting on Advances in Reactor Physics, Mathematics and Computation, Paris 1987

6 Neutron Spectrum Calculations for the UBR-Reactor

(G. Prillinger)

A series of two- and one-dimensional transport calculations have been performed at IKE to characterize the neutron environment of two out-of-core irradiation experiments in the University of Buffalo Reactor (UBR), New York. The work funded by US-NRC, has been done under contract with "Materials Engineering Associates", Maryland.

For the two long-time experiments the influence of changes in core loadings, of burn-up and the perturbation of the neutron spectrum by the experimental set-up and surroundings have been investigated. The DOT4.2 code has been used to calculate neutron spectra. The third dimension has been taken into account by space and energy dependent bucklings. Nuclear data have been taken from the Joint European File (JEF) and the International Reactor Dosimetry File (IRDF).

The transport calculation results show small effects, generally below 1 %, for changes in the core during the irradiation. Changes in the experimental set-up and surroundings are much more influential to the neutron spectrum and must be handled carefully.

7 Software for Evaluating the Local and Time Dependent Distribution of Radionuclides in the Lower Atmosphere and after Deposition on Ground

(G. Hehn, A. Böhm, Th. Müller, M. Mattes)

For evaluating the radiological effects of the reactor accident of Tschernobyl all measurements officially made in the land Baden-Württemberg, have been stored in a computer library. Appropriate interpolations were performed to get the local and time dependent distribution of radionuclide concentrations considering the decay of the active products and the relations between the integral quantities measured. Since also the measured data of food activities were included, the radiological burden of all groups in the population were derived and plotted by modern colour graphics.

In the first year after the Tschernobyl accident the increase of the effective dose equivalent in south-west Germany varies between 10 mrem and 80 mrem for adults and about 20 % higher values for babies.

V. Reactor Physics Activities at the Technical University of Braunschweig (IfRR)

J. Axmann

Three different main activities have been carried out during the last year in the field of neutron physics at the Institute for Spaceflight and Reactor Technology (IfRR), Technical University of Braunschweig.

a) Code development:

The 35 energy groups-, multi-zone-cellcode SPEKTRA has been implemented as a source of weighted group constants for the two-dimensional diffusion code DITUBS /1,2/. The treatment of the neutron-scattering in the thermal region has been improved.

b) Verifications of the data base and benchmark participations:

Main activities have been evolved to validate the cross section library, basing on ENDF-B IV and V. The aim is to achieve a revised data base for tight PWR-lattice calculations, especially in the case of coolant-loss. With a view to this, the IfRR is participating in the PROTEUS experiments in Würenlingen as an associated partner of KWU-Erlangen in coöperation with KfK-Karlsruhe and EIR-Würenlingen. The theoretical interpretations include cell- and whole-core-calculations, but also eigenvalue corrections basing on pertubation theory. Additionally the evaluation of the lattice-eigenvalue in a direct manner has been proposed /3/. The theoretical NEACRP-Burnup benchmark /4/ represents an excellent completion, because of the design near to the Proteus geometry buth with temperatures of power reactors.

c) Parametrical studies

Several design studies have been performed showing the potential of different tight lattices /5,6,7/. Parametrical investigations concerned the fuel compositions, the neutron-physical behaviour of steel additives and the fuel radius. Additionally the consequences of the coupling between necessary enrichment and fuel consumption for a fixed burnup on the fuel cycle costs could be shown.

/1/ Klüver, B.: Erstellung eines FORTRAN-Programms zur Berechnung eines Multigruppen-Neutronenspektrums von Druckwasserreaktoren. Report K 8705 IfRR 1987

/2/ Zeggel, W., W. Oldekop, J. Axmann, S. Bethke, H. Moldaschl, G. Ulrych, H.D. Berger: Analytical Methods in the High-Conversion Reactor (HCR) Core Design. Accepted paper for the special issue of "Nuclear Technology".

/3/ Axmann, J.: Progress Report K 8719 IfRR 1987

/4/ Akie, H., Y Ishiguro, H. Takano: Preliminary Report of HCPWR Cell Burnup Benchmark Calculations. To be published at the 29th NEACRP meeting

/5/ Axmann, J., X. Cheng, W. Oldekop: Neutronenphysikalische Aspekte unterschiedlicher Moderator-zu-Brennstoff-Volumenverhältnisse bei Druckwasserreaktoren. Kerntechnik Vol. 48 No. 4, 1986

/6/ Axmann, J., W. Oldekop: Neutronenphysikalische Optimierungsgrößen eines fortgeschrittenen Druckwasserreaktors. Proc. Jahrestagung Kerntechnik 1987 Karlsruhe, S. 19ff, 1987

/7/ Axmann, J., W. Oldekop: Parametric studies for the nuclear design of high-conversion pressurized water reactors. Kerntechnik Vol. 50 No. 4, 1987

ITALY

Reactor Physics Activities In Italy

R. Martinelli, ENEA

INTRODUCTION

Most thermal Reactor Physics activities in Italy have been carried out to provide support to the design and/or licensing reviews of BWR/6 "Alto Lazio" and of HW/BLW "Cirene", the requests for new analyses being obviously prompted by the Chernobyl accident (the event itself has been -and will continue to be- analyzed by an "ad-hoc group). In the fast reactor area, the halt called in 1986 by the Parliament for new committements has not been waived; so, all Reactor Physics activities are concentrated under the European AGT-3 umbrella.
In the meantime, the nuclear power program remains frozen, both for the planning of new plants and for the three operational stations shut down months ago for reload and/or unscheduled inspection and still awaiting the authorizations to restart. In fact, neither the conclusions of the National Energy Conference (March) nor the outcome of the general election (June) have led to a firm decision on the nuclear issue. Decisions are unlikely to be reached before the "indirect" referendum (November?) is held.

1) INTEGRAL EXPERIMENTS

No integral fast experiments have been performed in Italy (TAPIRO, which was used in recent years to feed blanket or shield configurations, is shut down to allow the installation of a computerized system for plant and environmental surveillance). Interpretation work is carried out at ENEA in cooperation with CEA, in the framework of the Research and Development Agreement existing between the two organizations.
One thermal critical facility is still operational, the heavy water moderated RB-3, which is mainly used for experiments in support to CIRENE core design and low-power commissioning tests preparation.

1.1) BALZAC (MASURCA). ENEA has partecipated in the design and implementation of specific configurations simulating large LMFBR cores, like "DE2", aimed at reducing the uncertainties related to the gamma-ray heating and neutron energy deposition for absorber and diluent subassemblies, and validating the appropriate "formulaire" /1/ and "IL", aimed at reducing the uncertainty of the heavy isotopes component of the reactivity loss per cycle /2/.
The status of the programme is illustrated in the French progres report to the Committee. (ENEA/VEL, Casaccia)

1.2) PROFIL (PHENIX). ENEA has continued to significantly cooperate in the interpretation of selected irradiation experiments of pure separated isotopes, the main objective being the validation of the basic data for the cross-section of the Np-237 (n, 2n) reaction. As a result of this interpretation, an integral value has been obtained of the branching factor for the reaction, and a very good agreement between calculation and experiment has been reached by using Pu-236 production cross-sections derived from JEF1 /2/. (ENEA/VEL, Casaccia)

1.3) Beta effective (SNEAK/ZPR). The SPX1 reactivity scale study previously carried out for four SNEAK experiments /3/, has been continued with the interpretation of six beta effective measurements in ZPR configurations, made available by ANL. The resulting, more significant analysis of ten values, has shown that the SNEAK and ZPR results are substantially coherent; a general calculation/experiment agreement is obtained using Tuttle's 1975 data, with the notable exceptions of ZPR cores containing high volume fractions of structural materials. On the other hand, no correlation has been found for fuel compositions or sodium contents, which can lead to the identification of systematic errors /4/. (ENEA/VEL, Casaccia)

1.4) ERESIA (RB-3). After completing the simulations of most physics measurements scheduled for CIRENE commissioning, void reactivity effects in CIRENE mock-up bundles have been measured in the critical facility RB-3. Various measurements have been made with natural and 1.15% enriched fuel, with different bundle configurations and with "coolant" (H2O) densities varying from 1 to 0 (intermediate densities were simulated by H2O-D2O mixtures). The interpretation via the WIMS D-CITATION route showed excellent agreement between calculation and experimental results, so that the latter can be confidently extrapolated to CIRENE core design /5/. (ENEA/CIRENE, Casaccia and Montecuccolino)

2) LWR CODE SYSTEM

ENEL, with the technical support of CISE, have completed the implementation and the systematic checking of their Integrated Code System (EICS) for LWRs, an accurate and effective system for core plant design and operation. In particular, the interface code for 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 /6/. These activities have shown that some aspects of thermal-hydraulic models -particularly, their application procedures- are still worth investigating, whereas it is hard to identify any incentive to further develop LWR neutronics calculational methods. On the other hand, the drastic reduction of RP activities in many laboratories makes it difficult to find expert support when one needs to use "reference" codes (often set-up in the sixties) to validate simplified neutronics models suitable for operation applications. (ENEL/DCO, Roma and CISE, Milano)

3) DEVELOPMENT OF CALCULATIONAL METHODS

3.1) Re-evaluation of the Chernobyl accident. A preliminary analysis of the neutronics and thermal-hydraulics of the first two phases of the event (initial positive reactivity insertion and first Doppler transient) has been made using standard computational methods, and the conclusions drawn are reported below /7/:

- in the accident configuration, the average void coefficient for the whole core is assumed to be .3 mk/% void. If the voided regions are the ones nearest to the core axis, the void coefficient is three times higher. This fully explains the enormous reactivity increase which characterizes the first phase of the transient;
- the design error in the scram rods (reduced length of the graphite follower) can generate an unwanted positive reactivity introduction when the rods are inserted for the first meter. This effect amounts to about 1 mk if all the other rods are withdrawn and becomes about 4 mk only if some of the regulating rods are inserted about one third (this, though, does not seem to be the case);
- the Doppler (negative) reactivity introduced during the transient is lower than one could envisage, because the effective Doppler temperature strongly depends on the fuel surface temperature. At fuel melting(2830°C) the effective oxide temperature is only 1700°C. The model for the calculation of the effective fuel temperature must be re-evaluated;
- the design dynamics model is zero-dimensional, but in some way accounts for space-time corrections and can be considered reliable up to the first peak. However, it is not reliable during the last phase (core disassembly and generation of the thermal and the mechanical energy which caused the explosion). For a self-consistent evaluation of the accident it is necessary to use the kinetics and dynamics models developed in the last ten years for the analysis of the reactivity accidents (TOP) in fast reactors. These models must be adapted to the RBMK reactor (and extended to LWRs, aiming at re-assessing reactivity accident conditions and consequences in these concepts). (ENEA/VEL, Casaccia)

3.2) Fuel Transportation Casks. In the framework of the NEACRP Working Group for the intercomparison of codes for the shielding assessment of transportation packages, a calculation methodology has been applied, based on the Monte-Carlo code MCNP with a new point-energy neutron cross-section library derived from JEF-1 via the THEMIS-ACER /10/ system.

This and other methods -based on the SCALE-2 modular system or developed at ENEA for special applications /8/- are being used in design verifications and licensing reviews of casks for the transportation of irradiated CIRENE-type fuel and other fissile material in solid or liquid form. (ENEA/TIB and PAS, Bologna)

3.3) Advances in Generalized Perturbation Theory

An effort is being made to develop a code for sensitivity analysis in the thermohydraulic field, making specific use of the heuristically based generalized perturbation theory. In particular, the interface conditions between the fuel pin and the coolant are accounted for by simply extending the thermohydraulic operator. This code will be based on COBRA IVC, which will be used for the calculation of the reference cases.
In general, the code will be capable of allowing extensive studies to be made, relevant to functionals which depend on the variables describing the system (for each channel: fuel, cladding, coolant and wall temperatures, material densities, coolant pressure, cross-flow).
The main structure of the extended jacobian matrix operator for the calculation of the importance function has been already defined.
A first version of the code is foreseen for the end of the present year and will be limited to problems at steady state, operating, conditions /9/. (ENEA/VEL, Casaccia)

4) JEF-1 DATA PROCESSING AND VALIDATION

The ACER module of NJOY for the production of point libraries for the Monte-Carlo code MCNP, has been implemented into the CRAY version of THEMIS /10/. A first set of 37 materials from JEF-1 has been processed; the validation of the library, which includes both neutron and photon data, is underway /11/.
A 219 group library from JEF-1 has been produced in the AMPX Master Interface Format for criticality safety calculations and is being validated by means of some benchmarks of the OECD-NEACRP Criticality Working Group /12/. (ENEA/TIB, Bologna)

5) NOISE ANALYSIS AND RELATED METHODS

5.1) BWR Stability Monitor. The system has reached a very advanced stage of development. The hardware for signal conditioning and sampling (up to 128 data acquisition channels) has been set up, as well as the software for computation and management. The whole system has been assembled and its compliance to all functional and design specifications has been successfully tested on analogue data recorded at Caorso NPS /13/. The field testing of the system is scheduled for the next Caorso reload. (ENEA/TERM, Casaccia)

5.3) Core Barrel Oscillation Studies. The time series analysis methods have been applied to the study of core barrel vibration as seen by ex-core neutron detectors. It turns out that the separation of neutronic noise components method, (Dragt, Turkan, Prog. Nucl. En. Vol.1, p.2931) slightly modified, is quite sensitive to detect direction of oscillations and frequencies /15/. It has been found that the barrel oscillates along the various directions one after the other and not simultaneously as it appears from the linear analysis. (Polytechnic of Milan)

5.4) Control Rod Vibration Studies. The detection and localization of a control rod anomalously vibrating in a PWR has been studied. Preliminary results indicate that some recent methods appeared in literature are too sensitive to variations of the selected reactor physics model and to the presence of white noise superimposed to the signals coming from in-core neutron detectors /16/. The potential of least-square and correlation methods is now under investigation. (Polytechnic of Milan)

5.5) Plutonium Assay. The problem of measuring the Plutonium content in irradiated fuel samples or in drums from reprocessing plants via statistical methods, has been tackled. The analysis is non destructive and is based on the detection of spontaneous fission neutrons from even isotopes of Plutonium and on their discrimination from the "background" neutrons originated by (α, n) reactions in Oxygen.
The general assessment of the problem has been completed, within the frame of a multigroup transport theory and of the Kolmogorov-Dmitriev theory of branching stochastic processes /17/. (Polytechnic of Milan)

6) FUSION TECHNOLOGY

6.1) Blanket Neutronics. At the conclusion of the study on "Il Mantello" (a ceramic breeder in toroidal tubes blanket cooled by helium in cross-flow) /18/, a "mixed" poloidally cooled blanket for NET has been studied that is perfectly compatible with the oblique removal scheme for NET first wall boxes /19/. In connection with this and in the frame of Design Studies B1(Blanket), N2 (Shield) and N4 (NET Water Cooled LiPb Blanket), detailed 3-D neutronic analyses have been performed by using the continuous Monte-Carlo code M.C.N.P.. This code allows the most accurate treatment for neutrons and related gamma-rays transport problems in complicated geometries including the real features of the plasma source. The analysis has been applied to the reference design of the NET-Double Null First Wall/Blanket and Shield systems. Both the liquid (LiPb) and solid (ceramic) breeder blankets (inserted within the closed box structure envisaged for the NET first wall) have been analyzed by using two 3-D complementary geometrical models. The first is a 3-D detailed geometry of a single blanket module reprepresenting the "almost real" features of different materials, cooling system, box sides and gaps, avoiding any material homogenization: this model was used to calculate the fine space distribution of tritium production and energy deposition. The second model consists of a 3-D toroidal geometry representing, in the NET device, the real shape of outboard and inboard blankets, divertor zones and ports: this model was used to calculate the overall tritium breeding ratio, the poloidal distribution of neutron wall load, the tritium production, the energy deposition (also on the superconducting coils behind the shield) and the radiation streaming through the vacuum pumping duct /20/. (ENEA/FUS, Frascati)

6.2) Neutron Dosimetry. An irradiation experiment with neutrons up to 800 MeV, performed at LAMPF in Los Alamos, has been analysed in order to collect data of importance for the first wall damage in future fusion reactors. The neutron energy spectrum has been determined by the adjustment technique applied to the activation data; a neutron cross section library for 14 activation or production reactions has been prepared and shown to be adequate to the problem /21/.
The work was performed as a collaboration with CCR-EURATOM of ISPRA. (Polytechnic of Milan)

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REACTOR PHYSICS ACTIVITIES IN JAPAN (OCTOBER 1986 - SEPTEMBER 1987)

Compiled by

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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) fabricaters, and also at some universities. Critical experiments are being performed at FCA (JAERI) and KUCA (Kyoto University). Improvement and assessment of HCLWR core design methods were proceeded. Interest for the innovative design study with aiming at increasing the inherently safe features has also been emphasized for light water reactors and advanced fuel loaded FBRs. Some intensive analyses of Chernobyl Power Plant have been made. Nuclear and thermal hydraulic charasteristics have been clarified to some extent. Another remarkable trend is an attention to the criticality safety problems. 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.

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, the conversion from use of 93 % enrinched uranium fuel to 45 % has been completed in JMTR of JAERI.
Concerned with radiation shielding, continuous studies have been made 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 connected with NEA.

Data and Method

(Benchmark Test of JENDL-3)

The preliminary version of JENDL-3 evaluated based on purely nuclear physics view point is almost completed. The working group for the benchmark test of the library is organized under Japanese Nuclear Data Committee. The working group including 44 reactor physicists is sub-divided into six sub-groups, (1) Standardization of reactor constants, (2) Integral test on FBR, (3) Integral test on LWR including high conversion LWR, (4) Integral test on fusion neutronics, (5) Integral test on shielding, (6) Integral test on dosimetry.
The modification of the library is scheduled on this autumn reflecting on the results of the above integral tests. The benchmark test will be repeated for the modified library. The final results will be presented at International Conference in Nuclear Data for Science and Technology, held at Mito Japan, May 30 - June 3 1988.

A new calculation method of sensitivity coefficients of cell parameters has been developed and applied for a high conversion light water reactor[1].

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UTILIZATION OF THORIUM

As a part of Special Project Research on Energy under Grant-in-Aid of Scientific Research of the Ministry of Education, Science and Culture since 1980, a research group named Thorium Fuel Study Group was organized and extensive studies have been performed by large number of group members. They held a Japan-U. S. Seminar on Thorium Fuel Reactors in October in 1982, and the proceeding of this seminar[1] and the mid-term report of the project[2] were published by the group. The final reports will be published soon.

In parallel with research works on fuel and structural material, reprocessing and radioactive waste management and biological effect to the human being, active works on nuclear data for thorium fuel cycle and on reactor physics of thorium fuel reactors have been carried out. Main subjects concerning the nuclear data are : (1) Fast neutron cross sections (fission, inelastic scattering, threshold reactions etc.), (2) Resonance parameters, resonance integral and resonance self shielding, (3) Fission and decay heat, (4) Integral test by neutron spectrum measurement, and (5) Compilation and evaluation of related nuclear data. Most of the main nuclear data were presented at the International conference on Nuclear Data for Basic and Applied Science, Santa Fe, in May, 1985[3]. Recently, an experimental study of self-shielding factor in unresolved resonance energy region for ^{181}Ta, ^{238}U and ^{232}Th has been started using the standard time-of-flight method at the electron linac facility of Research Reactor Institute, Kyoto University. A set of transmission and self-indication ratios obtained for several transmission sample thicknesses gives a self-shielding factor for an arbitrary dilution cross section. Measured self-shielding factors are compared with calculations where average resonance parameters are quoted from JENDL-2 and ENDF-B-IV[4].

In order to investigate the reactor physics of thorium fuel reactors, metallic thorium plates have been loaded in Kyoto University Critical Assemblies, KUCA and in Kinki University Reactor, UTR-KINKI. In the first stage of the KUCA experiments, every assembly consisted of a central test section surrounded by driver and reflector regions[5]. The composition of the test section was combination of graphite and metallic thorium plates. After completion of this stage, started was a mixed core series[6], in which each core consists of fuel elements with 93% enriched uranium aluminum alloy plates, metallic thorium plates (natural uranium plates for comparison) and polyethylene plates surrounded by a polyethylene reflector. Experimental results, not only critical mass but also other important data such as control rod worth, neutron flux distribution, capture and fission distributions, lattice parameters, have been compared with the calculated values which are mainly obtained using the SRAC code system. They have been trying to measure neutron spectra in those cores with recoil proton proportional counters and by the time-of-flight method. Smaller assemblies composed of metallic thorium plates and graphite have been loaded into the central vertical stringer in the internal graphite reflector between the two divided cores of UTR-KINKI[7]. Good agreement is seen between the experimental and calculated results. For the study on reactor physics of molten salt reactors, reactivity worths for ^{233}U, Be, LiF and teflon $(CF_2)_n$ have been measured with KUCA recently[8] Experimental results reasonably agree with the predicted values.

Neutronic properties of spectral shift light water reactors with thorium fuel were investigated[9]. A comparison of a thorium fueled pressurized water reactor PWR with urnaium fueled one shows that the thorium fuel is useful for achieving both high discharge burn up and high conversion ratio. Sensitivity analysis of a spectral shift PWR core has been made to find that the resonance capture cross section for Th-232 and fission cross sections for U-233 and U-235 are important to improve the accuracy.

New method was developed to solve neutron and nuclides density distributions at the equilibrium cycle of pebble bed reactors, and implemented in a new computer code PREC[10].

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THERMAL REACTOR PHYSICS

Continuous efforts have been made for performing feasibility studies on HCLWRs at JAERI, PWR and BWR fabricators and several universities.

The Phase-1 physics experiment on HCLWR core has been carried out from May 1986 to May 1987 at Fast Critical Assembly (FCA), JAERI. Measurements were made for criticalities for the FCA-HCLWR cores, infinite multiplication factors of the test zone cells which simulate the HCLWR core spectra, reaction rate ratios as spectrum indices, and reactivity worths of absorber materials such as boron carbide with different B-10 contents and Hafnium. Analyses for criticality and infinite multiplication factor were made with use of the SRAC code system, and the consistent C/E values were obtained through the Phase-1 experiment. The detailed analyses with respect to other measured items are in progress.

The first core of the Phase-2 experiment using plutonium fuels went on critical at the end of May, 1987. In the Phase-2 experiment particular emphasis is put on the measurement of conversion ratio, moderator void worth and plutonium isotopic effect.

In the analytical study at JAERI, a new group constant library was produced for use in the SRAC code system on the basis of the JENDL-2 and has been used in the following studies: The self-shielding effect of 2.67eV resonance of Pu-242 was investigated in connection with HCLWR neutronic characteristics. A new fission product chain model with 65 explicit nuclides and one pseude nuclide was established for HCLWR burnup analysis. On the other hand, evaluation works have been made for nuclear characteristics of the HCLWR cores of Vm/Vf = 0.75 and 0.81. Moreover, accuracy of conventional methods calculating control rod worth has been also evaluated in complicated HCLWR fuel assemblies. At LWR fabricators, detailed conceptual design study has been concudcted for HCPWRs with looser pitch lattice (MITSUBISHI), while an interesting design concept was proposed for HCBWR (HITACHI).

At LWR fabricators, detailed conceptual design study has been concudcted for HCPWRs with looser pitch lattice (MITSUBISHI), while an interesting design concept was proposed for HCBWR (HITACHI).

In order to obtain the basic reactor data of HCLWR, a small core with harder neutron spectrum was constructed at KUCA (Kyoto University Critical Assembly) using lower effective enriched uranium fuel. Analyses using the SRAC code system well predicted the measured quantities such as criticality, reaction rate distribution on gold wire, and Cd ratio. Here, a special attention was paid to calculate the transport corss section of reflector, due to smallness of the core.

In order to verify the nuclear design accuracy related to high temperature gas cooled reactor, various experiments have been conducted using VHTRC (Very High Temperature Reactor Critical Assembly). In this period, two kinds of core were constructed, that is, VHTRC-1 and VHTRC-3 cores which were loaded mainly with 4% and 6% enriched uranium coated particle fuels, respectively. The following parameters were measured: (1) Critical mass, (2) Temperature coefficient of reactivity, (3) Neutron flux distribution, (4) Reactivity worth of burnable poison rod. The predicted values by the SRAC code system agreed well with the experimental results. A detailed analysis has been made for the Doppler effect of coated particle fuel rod previously measured in SHE-14 (Semi-Homogeneous Experiment) using a sample heating device. The analysis using the SRAC code system again well predicted[1)].

For the investigation of the nuclear characteristics of the demonstration power plant of Advanced Thermal Reactor (ATR), studies of reactor physics parameters have been continued by using both uranium and plutonium oxide fuel (MOX) in Deuterium Critical Assembly (DCA). Recent measurements have been made by using the 36 rod fuel cluster which is the same size as that of the ATR demonstration power plant. The measurement items are: (1) lattice parameters such as δ^{25}, ρ^{28}, δ^{28}, δ^{49}_{25} and δ^{49}, (2) coolant loss reactivities and (3) microscopic and macroscopic power distributions. Comparing the experiment and the calculation, prediction accuracies on these core parameters for the ATR demonstration power plant are examined and discussed.

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FUSION NEUTRONICS

Analyses were continued for the previously reported experimental program at the Oktavian facility by the joint University faculties on a lithium metal sphere of 120 cm diameter, with and without a lead neutron multiplier layer inside it. Tritium breeding ratios tended to be overestimated by calculations, and the deviation is larger for the assembly with lead multiplier.
As a new method, total tritium production from ^{6}Li in the experimental system was estimated based on the neutron balance using the measured results of neutron multiplication and neutron leakage spectrum. The result also gave an overestimation of the calculated value.
An addition of 20 cm thick graphite layer outside the Li sphere was evaluated as a future plan to achieve a tritium breeding ratio greater than unity.

The results of the Phase I experiment of the JAERI-USDOE collaborative program using FNS facility at JAERI have been summarized in a joint report that will come out soon. In general, both deterministic and Monte Carlo calculations gave larger values than the measured values of tritium production rates (1.05 - 1.15). Deviations were even larger in breeder front face and around berrylium neutron multiplier zone. It was also pointed out that cross section of $^{7}Li(n,n'\alpha)^{3}T$ reaction should be examined.
A new series of integral experiments has started from August, 1986

at FNS as the Phase II of the collaborative program. A "closed geometry" system was adopted, i.e., the neutron source and the test blanket region were enclosed by a neutron reflecting zone in order to give a better simulation of the reactor neutron spectrum. The effect of Be neutron multiplier has been examined by comparing the tritium production distributions in the three different compositions for the test region.
Fast neutron angular fluxes from berrylium slabs were measured in the energy range from 50 keV to 15 MeV at FNS. Calculations with three different nuclear data files showed deviations ranging 20 - 30 % from the measured results in different manners.
Streaming experiments in various configurations have been conducted both at universities and JAERI.

SHIELDING

A Monte Carlo code, MORSE-CV, for calculating the covariance of the scalar neutron spectrum has been developed[1]: The calculated value from the spectrum agreed well with the directly calculated one; The standard deviation also agreed well each other when the spectral correlation between different energies was treated correctly.

The efficient albedo Monte Carlo method (AMC) has been evaluated by analysing two types of experiments on neutron streaming[2]: Through the analyses of neutron streaming experiments, the calculated results agreed with the measured data within a factor of 2 for a benchmark experiment at the YAYOI reactor and within a factor of 3 for a SNR sodium duct mockup experiment.

A Sn-albedo Monte Carlo-albedo Monte Carlo (Sn-AMC-AMC) coupling technique has been developed[3] and then implemented in the AMC code system MORSE-ALB to make an effective and accurate shielding calculation for large and complex geometry: Application of the present method was studied by analysing the neutron streaming experiment in the JOYO; It was shown that the Sn-AMC-AMC coupling calculation can reproduce the measured data within the accuracy of one order of magnitude.

In JAERI the decommissioning program of the JPDR (small BWR) has been undertaken since 1981[4]. In order to estimate accurately the radioactive inventories containing in power reactors at dismantling, a code system DOIC has been developed and verified by analysing the measured radioactivitieis in the reactor vessel, internals, and bioshield of the JPDR. The DOIC system consists of the AMPEX-II, ANISN, DOT3.5, GRAY (modified PALLAS), DCAHINMD (neutron activation calculation) and QAD-CG.

A review meeting for the joint U.S./Japan Program for Experimental Shielding Research (JASPER) was held at ORNL on April 1987[5]. The penetration experiments, the first item of the series of JASPER experiment, were analysed in Japan using the standard transport codes ANISN and DOT3.5. Two cross section sets were used in the analyses, i.e., JSD100 and JSDJ2, which had been generated from the ENDF/B-IV and JENDL-2 libraries, respectively. Through all the cases the C/E values spanned between 1.5 and 0.7 for JSD100 and between 1.2 and 0.6 for JSDJ2. Both the U.S. and Japan analyses indicate similar trends of decreasing C/E values for increasing sodium thickness. More analyses would be needed to resolve the discrepancies. Transverse leakage effects in the square slab mockups were identified as a possible cause of the discrepancies and should be investigated in future analyses.

It was shown that the presented geometric-pogression (GP) method can represent accurately the buildup factor as a function of distance[6]: Exposure buildup factors for point isotropic sources in an infinite medium approximated by the GP fitting parameters are in good agreement with the basic data calculated by the PALLAS code for boron for low energies and for lead considering the bremsstrahlung and fluorescence; The GP parameters were ascertained to be useful for interpolating the buildup factors.

A design philosophy on radiation shields of the fusion experimenral reactor (FER) was proposed[7]. Geometrical models and calculational parameters were studied to establish a standard design calculational method and to evaluate its accuracy. Then irradiation property of the in-vessel components and bulk shielding property were summarised for future design work in a useful form.

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CRITICALITY SAFETY

A kind of neutron noise technique proposed by Mihalczo was examined in TCA for a well-moderated cylindrical core. The method was successfully applied to determine the subcriticality up to 15 dollars utilizing a fission chamber containing ^{252}Cf of 0.2 mCi inserted into the core center[1].

Excitation of surface waves in a tank containing a fluid is called as "soloshing" in the area of earthquake engineering. The reactivity effects with sloshing of solution fuel were also measured in TCA for mock-up cores. A safety limit required to maintain the reactivity in negative range was proposed through the experimental study[2].

For the purpose to evaluate criticality accidents at fissile solution systems, a kinetic code AGNES (Accidentally Generated Nuclear Excurtion Simulation Code) has been developed[3]. The code treats the reactivity feedback with radiolytic gas void formation by so-called "pressure model" which was proposed through the KEWB program conducted in U.S.A.. The calculational results using AGNES for some benchmark problems showed a fairly good agreement with the experimental ones.

In the framework of NUCEF (Nuclear Fuel Cycle Safety Engineering Research Facility) program, two kinds of solution fuel critical facilities, STACY (Static Experimental Critical Facility) and TRACY (Transient Experimental Critical Facility), are being designed[4]. The principal objective of STACY is accumulation of basic criticality data, and that of TRACY is to study on critical accidents, which are required for criticality safety evaluation in nuclear fuel cycle facilities. The government safety review of the basic designs will be finished during 1987.

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Fast Reactor Physics

1. Improvements of Core Analysis Methods

1.1 Two-dimensional (2-D) Modelling for Plate Cell Calculation[1)]

The cell-averaged cross sections and diffusion coefficients for plate lattice cell fast critical assembles were calculated using the 2-D cell model by the CASUP code. The method was applied to criticality calculations of the Zebra CADENZA cores. When using the conventional 1-D model, the C/E values for the pin cores were larger by 0.3-0.5 %$\Delta k/k$, compared with the plate cores, depending on the 1-D cell homogenization models. However, by using the 2-D model for the plate cores, the difference between the pin and plate cores was reduced to about 0.2 %$\Delta k/k$. This is because the intra-flux effect on the plate core criticality was evaluated by the 2-D model calculation to be somewhat positive, compared with the 1-D model calculation.

1.2 Self-shielding Effect on Criticality of Plate Cell Assembly[2)]

For the preparation of cell calculation of plate cell, the Tone's method was applied to calculate the background cross sections of each plate in the cell, and then to derive the self-shielding factor of each plate. These self-shielding factors were used to obtain the cross sections of each plate, instead of the Dancoff factors. The Dancoff factor method overestimates the self-shielding, and hence underestimate the cross section of the plate.

The two methods were applied to the criticality calculations of the ZPPR-9, -10A, -10D and -13A cores. The k-eff values for these cores calculated by the Tone's method were lower by 0.14-0.16 %$\Delta k/k$, compared with the values calculated by the Dancoff factor method. The difference is quantitatively significant for the analysis of criticality, and should be considered and corrected.

1.3 Streaming Effect of Double Heterogeneous Structure of Subassembly[3)]

The neutron streaming in a typical prototype fast reactor caused by the double heterogeneous structure of subassembly has been estimated by means of multi-group (70 group) calculations. The sum of the streaming effect caused by the pin cell structure and the streaming effect caused by the wrapper tube structure was evaluated to be -0.13 %$\Delta k/k$. On the other hand, the streaming effect caused by the heterogeneity of the wrapper tube and inter-assembly sodium was evaluated to be -0.07 %Δ

k/k. Then, the total streaming effect becomes (-0.13)+(-0.07) = -0.20 %Δk/k, which is almost twice of the streaming effect of -0.09 %Δk/k obtained by the conventional pin cell structure. This magnitude of streaming effect of -0.20 %Δk/k is significant, compared with the double heterogeneity effect of 0.5 %Δk/k caused only by the resonance self-shielding and intracell flux distribution. The total heterogeneity with the streaming effect is reduced from 0.5 %Δk/k to 0.3 %Δk/k.

1.4 Three-dimensional Transport Correction[4)]

A three-dimensional transport code TRITAC for solving eigenvalue problems in reactor cores has been developed on the basis of discrete ordinates method with the diffusion synthetic acceleration (DSA) technique. The Larsen's procedure for the diffusion synthetic acceleration method has been extended to three-dimensional geometry.

Three-dimensional (3-D) transport corrections for k-eff, control rod worths, neutron spectra and power distribution are calculated for the fast critical assemblies ZPPR-9, -10 and -13. The calculations are performed in 7 energy groups using the 3-D and 2-D transport codes TRITAC and TWOTRAN-II. The evaluated 3-D transport corrections are compared with those predicted by a 2-D RZ model and a combined XY, RZ and R model to assess the accuracy of these 2-D models. The RZ model yielded some errors in estimating the transport corrections to k-eff, control rod worths, neutron spectrum and power distribution, while the combined model predicted them with good accuracy.

1.5 Three Dimensional Transport and Diffusion Codes Based on Nodal Method[5)]

A three-dimensional transport/diffusion code TRITON has been developed by Osaka University. The TRITON code has been applied to transport and diffusion calculations in a simple 3-D (XYZ) fast reactor core model. The calculations were performed in 4 energy groups. k-eff values and computing times were compared between TRITON and CITATION/TRITAC. In case of the nodal diffusion calculation, the k-eff was estimated as accurate as that by the finite diffusion calculation, and the computing time was reduced by a factor of 10. In case of the nodal transport calculation, the k-eff was improved greatly over the discrete ordinates calculation using the same number of mesh points, and the computing time was reduced by a factor of 5.

1.6 Improved Coarse Mesh Method[6)]

For an efficient computation of performance parameters of a large fast reactor, HICOM has been developed. This code utilizes an improved coarse mesh method to reduce required core memory and computing time, and an efficient algorithm is used in solving the diffusion equation, which is suited for vector processing computers. This code can be applied to a fuel exchange program and a control rod program.

1.7 New Treatment of Sodium Void Effect in Fast Reactor Transient Study[7)]

A new treatment of sodium void coefficient in the fast reactor transient analysis is proposed. The method is based on the multigroup exact perturbation theory and utilizes the void reactivity map constructed by the multigroup first order perturbation theory. The change in the spatial distribution of neutron flux is corrected by the flux ratio of the fluxes in the transient and the steady states calculated with few-energy groups. The void reactivities for a homogeneous and a heterogeneous reactor are calculated by the new method with 4 and 6 energy groups and are compared with standard values obtained by the 25 group exact perturbation calculation. It is demonstrated that the new method gives almost the same values as the standard values while the ordinary calculation with 4 and 6 groups gives considerable deviation from the standard calculation. The present method has been incorporated to the two-dimensional quasistatic transient analysis code QUASAR and the code has been applied to an analysis of a control rod withdrawal accident.

2. Critical Experiments[8), 9)]

The JUPITER-III program, the joint physics large FBR core critical experiments program between the U.S. DOE and PNC, Japan, using the ZPPR facility at ANL-Idaho, started on January 1987, and will continue for one year. The experimental program consists of two different series of experiments. The ZPPR-17 series are 650 MWe-size benchmark cores of axially heterogeneous core concept, including a clean benchmark core, an engineering benchmark core with 25 control rod channels and a beginning-of-life core with 13 half-inserted control rods. The ZPPR-18 series are 1000 MWe-size two-zone homogeneous cores, including an engineering benchmark core with 24 control rod channels, a beginning-of-life core with 18 half-inserted control rods and its one-rod-stuck phase core with 17 half-inserted control rods.

3. Advanced Core Design Studies

3.1 Axial Heterogeneous Core[8)]

The axial heterogeneous core (AHC) concept was selected as a reference core of the cost reduction design study of the Japan Atomic Power Company (JAPC) on the pool type Demonstration FBR. Design studies were performed to optimize the 1000 MWe AHC configuration with a view of obtaining more compact core systems, longer fuel lifetimes and better safety characteristics. In the present core design studies, a disk-shape internal blanket is located at the mid-plane of the core, or at the lower part of the core. The purpose of the latter asym- metric arrangement of the internal blanket is to make the power distribution flat throughout the whole burn-up cycle, even when the control rods are partially inserted from the top of the core.

3.2 High Burn-up Core[10)]

High burn-up potential of various core concepts has been studied from the viewpoint of fuel cycle cost reduction and reactor availability factor improvement. In order to achieve a high burn-up core of 150-200 MWd/kg, it is necessary to develop long life fuels with low-swelling and high creep-strength materials. However from the viewpoint the core design tech- nology, following items are important to moderate the require- ments for those material properties and to attain the long cycle length: reduction of ratio of peek neutron fluence to discharge burn-up, reduction of burn-up reactivity and reduc- tion of assembly power decrease due to burn-up.

The Super Long Life Core (SLLC) or Ultra Long Life Core (ULLC) concept has been evolved as one of the targets of the innovative approaches for FBR cost reduction. A neutronic feasibility study was done for the SLLC concept, where core life time was extended up to the plant life of about 30 years, by applying the radially and axially multi-zoned core concept with mixed oxide fuels. The study shows that to achieve small enough burn-up reactivity swing makes the core volume enlarged to about three times as large as that of the conventional design.

3.3 Metal, Carbide and Nitride Fuelled Cores[11), 12)]

Feasibility studies were started for the advanced fuel loaded FBR cores, i.e., metal, carbide and nitride fuelled cores. It is commonly understand that the metal fuelled core has a potential of high breeding and inherent safety character- istics. The nuclear and safety characteristics of a large metallic fuel core have been evaluated in comparison with the mixed oxide core. Preliminary safety calculations indicate that higher thermal conductivity of metal fuel enhances the inherent safety of the reactor with respect to loss of flow without scram and loss of heat sink without scram. Design and safety evaluation will be continued for the metal fuelled core.

Carbide and nitride fuels have also potentials to be used in future FBR cores. High thermal conductivity and high content of heavy metal of these fuels yield high linear heat rate and high breeding ratio. Nuclear and safety characteristics of carbide and nitride fuels, as well as fuel cycle and economic implications, have been evaluated, and will be compared with those of oxide and metallic fuels.

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National Programs

JOYO

Following the 11th duty cycle operation, the 12th duty cycle operation started on September 1985. At the end of the 12th duty cycle operation, the natural circulation test from the 100% reactor power of 100MWt was conducted, where the sufficient residual heat removal capability by the natural circulation was confirmed. At the same time, it is also confirmed that the predicted changes in the temperature of coolant sodium show fairly good agreement with experimental ones, thus, the validation of the analysis method on the thermohydraulic behavior of the reactor plant at the natural circulation condition is established.

A special test cycle operation, called the 12'th cycle operation, was conducted after the 12th duty cycle operation, starting on November 1986. At the 12'th cycle operation, two tests concerning the fuel behavior and the feedback reactivity were conducted. The aim of the former test is to clarify the behavior of the fresh fuel pellet of which specification is almost the same as the one of prototype fast breeder reactor MONJU, at the reactor operation with rather rapid power ascent. In the latter test, the reactor was operated with various power levels and flow rates of coolant different from rated ones to separate each effect which contributes the feedback reactivity at power change.

Following the 12'th cycle operation, the 6th periodical inspection of the reactor was started on December 1986, and is now in progress. The reactor will commence the 13th duty cycle operation in the beginning of September of this year. From the 13th duty cycle operation, both the period of a duty cycle operation and the maximum burn-up of the driver fuels, of which maximum are increased in new license from 45 days to 70 days and from 50,000MWd/t to 75,000MWd/t, respectively, will be gradually extended using modified driver fuels, named J2 fuel.

MONJU

The construction work of the MONJU plant, a 280 MWe loop-type FBR plant, has been making steady progress since its commencement on October 1985. The erection of the containment vessel was begun in July 1986 and completed in February 1987. Pressure and leak tests were completed in April 1987. Five sodium tanks for the secondary loop and two sodium tanks for the ex-vessel fuel storage were installed in November/December 1986. The construction work is now more than 30 percent complete.

Besides construction work, development of a code system supporting reactor core operations and the management of MONJU was performed. This code system is made up of the following three parts: (1) engineering data base, (2) calculation modules, (3) system access software. The main parts of the calculation modules were completed and are now available for 3D-diffusion burn-up calculations and thermo-hydraulic calculations.

Demonstration Fast Breeder Reactor (DFBR)

The Japan Atomic Power Company (JAPC) has carried out the cost reduction design study both on loop and pool type DFBR sponsored by ten private electric power companies. In view of making the commercial FBR compete with LWR, JAPC also carried out investigations of inovative technologies and evaluation of cost reduction effect.

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

The studies for the breakthrough to the commercialized FBR were started. High burn-up cores were studied from the viewpoint of their feasibility. In this study core characteristics such as burn-up reactivity swing, power distribution, and peak fast fluence at the burn-up of 150,000MWd/t to 200,000MWd/t were evaluated for three type cores, which are homogenious core, axially heterogenious core, and plutonium assymmetric parfait core.

As for the inherent safety core design, metal fueled 1,000MWe homogenious core was studied in comparison with oxide fueled core by JAPC. CRIEPI also has been studying the metal fueled core.

JAPC, PNC, Japan Atomic Energy Research Institute (JAERI), and Central Research Institute of Electric Power Industries (CRIEPI) have organized Steering Committee on FBR R&D in order to coordinate effective allocation of FBR R&D works implemented by each organization and related international cooperation under nationally authorized FBR development strategy. Steering Committee has been discussing the DFBR project plan and the commercialization strategy in about 1990. JAPC is performing concept evaluation and design for DFBR and the commercial FBR.

FUGEN

The 6th annual inspection and the 10th refuelling were carried out from January to April 1987. At the 10th refuelling, 18 UO_2 and 18 MOX fuel assemblies including 2 demonstration MOX fuel assemblies were charged. At present, FUGEN has continued stable full power operation. 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 high performance fuel.

Up to date, 235 UO_2 and 216 MOX fuel assemblies have been discharged for refuelling. The maximum burn-up is 19,200MWd/t for UO_2 fuel and 18,500MWd/t for MOX fuel, and no leaking fuel has been found more than 1,880 effective full power days of operation up to May 1987.

ATR Demonstration Plant

The construction program of the ATR Demonstration Plant has started with the decision given by Japan AEC in 1982 that EPDC (Electric Power Development Company) be responsible, in a close cooperation with the government, electric utilities and PNC, for the construction and operation of the plant.

The ATR is a heavy water moderated boiling water cooled pressure type reactor originally designed by PNC. EPDC took over information/results of ATR design development from PNC, and started the conceptual design work including design rationalization. The capacity of the plant was fixed to be 606MWe. At present EPDC is preparing for the application for the construction permit.

The environmental survey is almost completed, which was carried out at the expected construction site in Ohma-machi, Shimokita-gun, Aomori-ken. The results are being prepared for the environmental examination by the national authorities.

According to the current schedule of the project, the construction is expected to start in 1991, and the commercial operation in 1997.

Report on the Reactor Physics Activities in the Netherlands in the period September 1986 - August 1987

compiled by H. van Dam (IRI, TU Delft)

1. Reactor physics activities at the Netherlands energy research foundation ECN, Petten (J. Slobben)

The reactor physics activities performed at ECN 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 (HFR) at Petten.
2. Neutron spectrum calculations for certain irradiation positions in the HFR.
3. Neutronics calculations for the shielding blankets of NET and JET.
4. Development of methods and systems for noise measurement 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)

The nuclear-data activities in 1986/1987 were directed towards the forthcoming second version of the Joint Evaluated File (JEF-2). Some new evaluations have been made for Ru-101 [1], I-129 [2] and Ru-102 (to be reported). In addition an effort was made - in cooperation with the NEA Data Bank - to update the thermal cross-sections, resonance integrals and the cross-sections in the resolved-resonance range. This range was in may cases extended towards higher energies. The results will be included in JEF-2.

In the MeV range some work has been performed to improve the inelastic scattering cross-sections by studying the effect of direct components obtained from DWBA or coupled-channels calculations [3,4]. It appears that the inelastic scattering cross-sections at low incident energies are systematically underpredicted in the current evaluations due to the neglect of direct components. Corrections are needed in particular for the even-mass isotopes of Ru, Pd, Nd and Sm.

With respect to the lumped fission product cross-sections the radiative capture cross-section is relatively well-known. Therefore it is important to study other sources of errors in the calculated reactivity effect of fission products in fast power reactors at the end of cycle. Possible error sources are the neglect of direct effects in inelastic neutron scattering cross-sections (see above) and the leakage of gaseous products or the migration of volatile products out of the core of a fast reactor. The last-mentioned problems have been studied in cooperation with CEA-Cadarache [5]. Recently, attempts have been made to develop a simple model to describe these effects, using phenomenological parameters which have been fitted to experimental results obtained in the French PHENIX reactor.

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

Under contract with JRC-Ispra the US activation and transmutation file REAC[1] has been updated. The new file is called REAC-ECN and contains all stable (non-fissile) nuclides and nuclides with a half-life greater than 1 day [2]. Recently, additional work has been performed to further update the data file by introducing new systematics of cross-sections [3] and isomer ratios [4]. The second version of the REAC-ECN file wil contain better evaluations of cross-sections of activation reactions producing long-lived nuclides or isomers. This work is in good progress.

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1.3. European Fusion file (EFF) (H. Gruppelaar)

The aim of this project is to achieve an evaluated nuclear data file for neutronics calculations for blanket and shielding engineering. The project is part of the European Fusion Technology Programme of the

European Community, to which several laboratories contribute. The file management and maintenance is performed at ECN-Petten.

A first version of the file (EFF-1) has been distributed in March 1986 to European laboratories. Status reports are given in Refs. [1-3]. The main contribution of ECN-Petten consists of a revision of the lead double-differential cross-sections [3] in the new MF6 format of ENDF-VI.

The calculation of multi-group cross-sections and transfer matrices was organised. Several European groups have made contributions. The ECN-Petten contribution consists of processed lead data using a recently developed code that treats double-differential cross sections in the above-mentioned new format [4]. This code and a preliminary version of the multi-group constants library GEFF-1 has been distributed to European laboratories.

Recently, the requirements for EFF-2 have been defined, see also Ref. [3]. At ECN-Petten some work on the re-evaluation of nuclear data for the Ni-isotopes was initiated. Furthermore, a first version of a separate activation file (EAF) has been made.

An essential part of the new evaluations is based upon calculations with nuclear-model codes. Some contributions to pre-equilibrium nuclear-model theory and codes are listed in Refs. [6-16].

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1.4. Neutronics calculations for the neutron diagnostics at JET

(K.A. Verschuur)

During a stay of nine months at JET the neutronic program system FURNACE [1] was installed on the IBM/CRAY-XMP at Harwell. The geometry options of the system were extended to enable separate modelling of the coil/structure assembly and the vacuum vessel respectively. Apart from double-differential albedo's now double-differential transmission coefficients are required as well. Subroutines have been introduced to calculate scalar flux spectra at arbitrary positions in the vacuum regions, and to calculate angular flux spectra for arbitrary angular and spatial coordinates. A first series of calculations was performed to obtain information on the back scattered neutron flux which has to be expected for the neutron profile monitor system [2]. This system consists of a horizontal and a vertical set of neutron spectrometers that view the plasma through an array of collimators (KN3). The responses of these detectors have to be corrected for the contribution of the back scattered neutron fluxes. The calculations for d-d operation show that with an energy threshold of 2 MeV the contribution of the back scattered neutrons to the detector responses is maximally 3.5% of the highest rsponse (i.e. of the channel that looks through the hottest part of the plasma). The calculations also show that the back scattered flux is rather flat, i.e. that it has about the same value for each channel.

Further it was shown that the 14 MeV neutrons from the secondary t-d reaction can be measured with a back scatter contribution of about 3.5% at a threshold setting at 3 MeV, which is rather fortunately. Calculations were also started for the foil activation diagnostic system (KN2). However, due to some calculational problems results are not yet available.

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1.5. Experience with on-line power reactor noise monitoring system for Borssele Power Reactor (E. Türkcan)

Power operations of the cycles thirteen and fourteen of the Borssele Nuclear Power Plant (PWR - 450 MWe, 2 loop-system built by KWU) have been followed continuously by an on-line noise monitoring system [1]. In the course of the plant operation the following topics have been studied:

- in-situ test of the thermocouples (core inlet, core outlet and core exit), neutron detectors and pressure sensors using the inherent noise of the plant;
- detection of the short term trends (e.g. during start-up and shutdown) and long-term trends in the physical parameters depending on burn-up, etc.;
- monitoring of core support barrel (CSB) motion (amplitude, in order of 5-10 µm, and direction) and reactivity changes;
- monitoring of the secondary system with regard to steam pressure, steam flow (e.g. damping coefficient and frequency), turbine and generator output.

On-line monitoring and surveillance methods have been improved continuously [2]. The next attempt will be to get the experience of the reactor operating staff from such a system and to improve "dialogue" between reactor operator and monitoring system for easy communication or changing set-ups, e.g. type of analysis, cycle time for the surveillance period etc.

[1] E. Türkcan, Nuclear Europe, Journal of ENS no.6, June 1986.
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1.6. Application of Noise Analysis to the study of the Thermal and Mechanical Behaviour of a fuel pin during irradiation (E. Türkcan)

The measurement of the time constants provided a better insight in the thermal characteristics of the fuel pin under power load change operation. The noise measurements have been applied to the HFR-TOP fuel pins with UO_2 pellets with initial cold gap size 35-120 µm. From experimental observation the heat transfer characteristics are determined using the transfer function between heat source and temperature fluctuations either in the fuel or in the coolant. For the interpretation of the results two-dimensional time-dependent calculations of the response time have been performed and compared with experimental results [1].

Developed methods and the software for numerical calculations have been applied to LMFBR mock-up fuels such as $(UPu)O_2$ (KAKADU), (UPu)N (NILOC), irradiated in the HFR at Petten [2].

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1.7. Neutron metrology (H.J. Nolthenius)

Attention has been given to the evaluation of participants results of the REAL84 exercise. The REAL-84 was a follow-up of the REAL-80 exercise [1] and has been organized by the Nuclear Data Section of the International Atomic Energy Agency. The aim of the exercise was to improve the assessment of accuracies in the prediction of radiation damage parameters by various laboratories using good quality input data and proper calculation methods. The emphasis was put on radiation damage parameters for reactor pressure vessels and related nuclear technology. Therefore the upper limit of the neutron energy range of interest was 20 MeV.

The long term objective of the exercise is to strive towards the establishment of standardized metrology procedures and recommended nuclear data for use in neutron spectrum adjustment and damage parameter calculations. The short term objective of the improvement of information on the neutron spectrum adjustment technique and its nuclear data needs. The scope of the exercise and the input data sets have been described in information sheets [2] and [3]. In September 1986 calculation results were available from 10 different laboratories. The evaluation results were presented in a series of progress reports (with restricted distribution). An IAEA consultant meeting on the assessment of the results of the exercise was held in Budapest [4].

[1] W.L. Zijp, et al. 'Final report on the REAL-80 exercise', Report INDC(NED)-7; BME-TR-TES-6182; ECN-128 (Netherlands Energy Research Foundation ECN, Petten, February 1983).

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2. Reactor Physics at the Interuniversity Reactor Institute (H. van Dam)

Extensive measurements have been performed on the stability of the BWR with natural circulation at Dodewaard in order to establish which operational factors determine the reactor stability and which parameters are good indicators for the stability margin. With a method for on-line measurement of two-phase flow velocity, based on the application of correlation techniques on the signals of incore neutron detectors, it was proved to be possible to measure low-frequency velocity variations which is of importance for the analysis of thermohydraulic stability.

An out-of-core experimental loop was constructed for simulating the fluctuations in a radiation field caused by two phase flow in order to improve the insight into the interpretation of the signals. In particular it aims at improving the interpretation of data obtained by correlation analysis of signals from incore neutron and gamma detectors.

In the framework of the NIOBE project (Noise Investigations of Boiling Effects) a new boiling loop was installed in the swimming pool type research reactor HOR. This new rig has a higher heating power than the previous one and an improved controlability of the thermohydraulic conditions. Preliminary measurements were performed of thermocouple and neutron detector signals in non boiling conditions. In two out-of-core boiling loops the information content of thermocouple noise was investigated. It was shown to be possible to infer flow velocities from signals of out-of-stream thermocouples, positioned in the walls of coolant channels; in this case the interpretation of the frequency dependent phase of the thermocouple signals should be done with care because of the interference between fluid transport effects and small fluctuations in the heating power, the so-called local-global noise interference.

The computer code system of the group was used for reactor physical calculations in support of the HOR reactor and for analysis of minimization of critical masses. The latter study was performed in order to test the merits of data sets and codes for systems with small dimensions that make high demands on the calculation accuracy. A critical system was dimensioned with a lower critical mass for Pu-239 than any system known up to now. It consists of a berylliumhydride moderated core with a berylliumoxide reflector. The fuel is distributed in such a way that a flat fuel importance distribution is obtained; in the minimum critical mass configuration, containing 87 grams of Pu-239, about 30 % of the fuel is positioned in the reflector near the core-reflector interface [1].

In the framework of an international benchmark excercise performed by the Nuclear Energy Agency of the OECD, data have been generated for an artificial noise benchmark, using a computer controlled reactor simulator. These data were made available on analog and digital tape to participants; evaluation of the results will be reported on the occasion of the Fifth Specialists' Meeting on Reactor Noise in Munich, October 1987.

Two other projects concerned the on-line determination of reactivity of the HOR reactor and transport calculations on an experimental set-up for "in vivo" determination of cadmium in kidneys by measurement of capture gamma rays produced by irradiation with a Cf-252 neutron source. The first project aims at solving interpretation problems encountered by applying inverse kinetics to subcritical transients. The latter project is focused on optimal dimensioning of the irradiation device and an accurate calculation of the dose received by the patient.

The analysis of the reactor physical aspects of a gaseous core reactor was resumed (see the 83/84 activities report). In this reactor type the fuel is at high temperature and in a partly ionized gaseous state, surrounded by a graphite reflector; the fuel has such a composition that it is in thermodynamic equilibrium with the reflector material [2]. Present research is focused on criticality aspects, reactivity coefficients and neutron generation times in dependence of the system dimensions.

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3. Reactor physics at KEMA (J.A.M.M. Kops)

A 3 D core simulation program is used for fuel management calculations as well as analyses of transients for the Dutch nuclear power stations (a small 54 Mwe BWR and a 450 MWe PWR). Furthermore last year the methodology is used for a detailed reconstruction of the Tsjernobyl accident. The latter analysis shows an essential role in the accident sequence of the graphite displacers of the control rods.

The programmature is continuously validated by comparing the calculation results with the measurements of reactor physical paramters as well as gamma spectrometrical measurements of fuel elements. A significant deviation in k_{eff} may result from a neglection of the neutron activation of ^{10}B tot ^{11}B in PWR primary water. To investigate this, measurements of the ^{10}B /^{11}B ratio will be performed.

An in vessel water level gauge using groups of heated thermocouples (BICOTH sensors) has been installed in the Dodewaard reactor. The work is performed in close cooperation with the Japanese research institute JAERI and with Halden Reactor Project Group in Norway. Up till now the system operates satisfactorily and the results compare quite well with the conventional water level instrumentation. An analog equivalent resolution was obtained by superimposing a smoothing transition function on the distinct water level indications that follow directly from the 8-digit binary code generated by the BICOTH sensors as a function of water level height in the vessel.

A first presentation on this new water level measuring system has been given last April at the Japan Atomic Energy Society Conference at the Nagoya University in Japan. A second presentation of the system is planned for the OECD/NEA Specialists' meeting on In-core Instrumentation and Reactor Core Assessment in Cadarache, France, next year.

STATUS REPORT TO NEACRP
(1986-1987)

REACTOR PHYSICS ACTIVITIES IN NORWAY
SEPTEMBER 1986 - AUGUST 1987

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Efforts in reactor physics in Norway are made through the activities of Scandpower International Consultants (based at Kjeller), and through the work at the OECD Halden Reactor Project.

1. FMS CODE SYSTEM

Scandpower A/S provides continued support and maintenance of the code package FMS. This modular code system for light water reactor calculations is currently in use by a number of power utilities and other organisations in Europe, USA, Mexico and the Far East for both BWR's and PWR's.

A fairly large effort has recently been completed in upgrading the RECORD code, the data generation unit of FMS. A number of new models and features have been added reflecting user requirements for analysis of new fuel designs, control rods and detector systems. An outline of some of these implementations and features of the RECORD code was presented at the Paris meeting in April (Ref. 1).

Effort continues on the Core Master PRESTO for on-line applications on operating BWR power plants. PRESTO has now implemented an automatic search option for the worst control rod in shut down margin calculations.

Scandpower has carried out many projects involving reactor dynamics studies with the RAMONA code. RAMONA is a code for BWR transient analysis in 1- to 3-dimensions.

2. HALDEN PROJECT DEVELOPMENT OF A CORE SURVEILLANCE SYSTEM

A complete version of the core surveillance system SCORPIO is being implemented in Ringhals Unit 2 PWR in Sweden. A demonstration version of the system will be installed this year at Duke Power Co., USA, simulating the Catawba Unit 2 PWR.

Present physics work around SCORPIO is concentrated on systems improvements for handling of reactors with axial xenon instability.

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SWEDEN

STATUS REPORT
to the NEACRP meeting
Helsinki, September 14-18, 1987

Nuclear Power in Sweden 1986/87

According to Swedish statistics electricity accounted for 29 % of the total energy supply in Sweden, according to international statistics the figure should be 52 %. Either way the electricity consumption is high compared to many other nations. During 1986 nuclear power supplied 67 TWh or 50 % of total electricity production, hydro power 60 % TWh or 44 % and remaining 6 TWh came from oil- or coal-fired stations. The total domestic consumption was 129 TWh or about 2 % less than 1985. If the figures are adjusted for temperature differences the result is that the consumption of electricity 1986 was 2 % higher than the previous year.

1986 was the first whole year with Forsmark 3 and Oskarshamn 3 in commercial operation. The average capacity factor of the twelve reactors was 80.6 % and the availability 87.1 %. Oskarshamn 3 had the highest capacity factor with 90.4 %. During certain periods the nuclear stations operated with reduced power for load optimization, corresponding to a reduction in production of 2.2 TWh.

Reactor Physics Calculations on Tight PWR Lattices

(Erik Johansson)

Recent work at Studsvik on plutonium recycling in tight PWR lattices has been summarized in a paper to appear in Nuclear Technology in the beginning of next year. The work consists of test of the calculational model, essentially the

CASMO code and its standard 70-group data library, against the PROTEUS experiments, and of application of this model on various power reactor systems with tight lattices.

Although there were some discrepancies, the outcome of the test was considered good enough to justify the power reactor calculations. The main results from these calculations concern the consumption of natural uranium and separative work, and the void reactivity. As an example, the consumption of natural uranium for 50 yr of operation in a tight lattice was about 25 % below the value for recycling in a normal lattice. The void reactivity was positive in some of the calculations. The calculational uncertainty is fairly large, but at least for one of the systems this reactivity is very likely positive also in reality.

At present there are no plans for further calculations on tight lattices at Studsvik, but of course this subject might be taken up later on. It can be mentioned, finally, that the calculational model has been applied to the cases of the international benchmark study for tight lattices. The results are included in a paper from Japan to be submitted to this meeting.

Approximative Analysis of Inadvertent Control Rod Withdrawal in the Super Phenix Start-up Core
(Klas Jirlow)

An approximative analytic relation between the maximum temperature rise in the most rated fuel pin section and the main parameters (reactivity insertion, Doppler constant and power form factor) has been derived for the complete withdrawal of one inner control rod in the present Super Phenix fast breeder reactor core.

The maximum cunter fuel temperature increase is in the range 300-400 °C and it depends primarily on the form factor of the perturbed power distribution and on the Doppler constant. In comparison with previous studies the values are lower but this is consistent with the more recent set of main parameters.

Thorium Fuel in PHWR
(Gunnar Andersson)

Physics calculations have been performed for heavy water moderated and cooled reactor with pressure tank and thorium containing fuel.

LWR In-Core Fuel Management
(Kim Ekberg)

STUDSVIK ENERGITEKNIK AB, with its subsidiary STUDSVIK of AMERICA, INC, has been active in developing computer codes and methods for LWR ICFM for a long time. The codes CASMO-2E, SIMULATE-2 and MBS have been used by a large number of utilities and institutions world-wide.

A continuous development work, intended to improve the computing tools, is being carried out by Studsvik. Recently the new versions CASMO-3 and SIMULATE-3 have been introduced (1, 2, 3, 4, 5). CASMO-3, while retaining the original reactor physics models, offers many new options, and its nuclear data library has been revised. Data for the most important nuclides are now based on ENDF/B-4.

SIMULATE-3 is a completely new code. Its central neutronic algoritm is the two-group QPANDA model, which in turn is a refinement of the QUANDRY model. The very general cross section representation in SIMULATE has been completed with assembly discontinuity factors related to the very sophisticated homogenization techniques used.

A pin power reconstruction model, based on pin power distributions from CASMO-3 and intra-nodal flux distributions from QPANDA, has been introduced in SIMULATE-3. It allows a complete 3-dimensional evaluation of pin powers in a PWR in little more time than is normally needed for a 3-D nodal power distribution calculation.

Benchmark studies with CASMO-3/SIMULATE-3 have been completed for PWR and are in progress for BWR. Several utilities are in the process of making their own benchmark studies with the intention of using the codes for licensing purposes.

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REACTOR PHYSICS ACTIVITIES IN SWITZERLAND

October 1986 to September 1987

P. Wydler

1 GENERAL BACKGROUND

In Switzerland, as in other countries, nuclear energy is facing an increasing number of political obstacles. The most important of these are two new antinuclear initiatives, expected to be put to the vote in 1989. Advances in the Federal Parliament to discontinue the operation of the existing power plants and to withdraw the general license for the Kaiseraugst project have been rejected. However, the concerns expressed by the parliamentarians have prompted the Federal Council to charge a commission to study the consequences of phasing out nuclear energy and to propose appropriate scenarios for this by the end of the year. At the level of the cantons, unfavourable developments include parliamentary decisions at Berne and Geneva to rule out the construction of nuclear power plants on proposed sites within their territory.

In anticipation of a widening gap between production and demand, the Swiss electricity companies have decided to further increase their electricity imports from France. Imports have now been agreed with EdF which, by the middle of the 1990's, will correspond to 25% of the home-produced electricity. Since Switzerland has always been self-sufficient in electricity, this development represents a major policy change.

As a result of an optimisation study the two Federal nuclear research laboratories, EIR Wuerenlingen and SIN Villigen, are about to be merged into a single organisation. As regards future activities of this laboratory, it has been recommended that fission reactor research and development should be reduced unless the laboratory can participate in an international project such as the HTR-500 (on which Swiss industry collaborates with industry in the Federal Republic of Germany), or can support Swiss industry in the development of a small district heating reactor in the 10 to 50 MW thermal energy range.

Proposals for three heating reactor concepts, the (water cooled) Swiss Heating Reactor (SHR), the so-called "GEYSER" reactor, and the Gas-Cooled Heating Reactor (GHR), are currently being evaluated. The SHR, proposed by EIR, is based mainly on standard BWR technology; the GEYSER, an innovative concept proposed by SIN, uses a 50 m high water column to produce coolant pressure, and self-regulating boron poisoning for reactivity control; and the GHR, supported mainly by industry, relies on HTR technology.

In March the Federal Parliament approved funds for a spallation source, to be added to the cyclotron complex of SIN. With a target power of 650 kW the source will be about five times more powerful than existing sources of this type. Although not intended to be competitive with powerful international facilities such as the high flux reactor at ILL Grenoble, the source will outperform EIR's existing neutron source, a 10 MW swimming pool reactor, and thus provide an important tool for home-based fundamental and applied research.

Among the activities to be reported the Light Water High Converter Reactor (LWHCR) work at the PROTEUS reactor in Wuerenlingen, which receives larger funds than other reactor physics work, has to be emphasised. Recently, an agreement has been signed to cover LWHCR collaboration with partners in the Federal Republic of Germany. Other reactor physics activities at Wuerenlingen comprise core analyses in support of the SHR, and basic method development and data testing.

At the Federal Institute of Technology Lausanne fusion blanket experiments with a lithium blanket module, developed by the Princeton Plasma Physics Laboratory, have been completed. Measurements on other blankets are continuing.

2 LWHCR PHYSICS EXPERIMENTS AND ANALYSES

Investigation of the PROTEUS-LWHCR Phase II lattices has been continued. The Phase II lattices are single rod configurations, using mixed oxide fuel with a fissile plutonium content of 7.5%, typical LWR plutonium isotopic composition and a fuel-to-moderator ratio of 2.07.

An NEACRP-A-paper, submitted for this meeting, reports on comparisons of calculated and measured integral parameters for the Phase II reference lattice, both wet (with H_2O) and dry (100% void). The k_∞ void coefficient, α_v (0 - 100% void), and its components have been analysed in a manner analogous to that applied in the PROTEUS-LWHCR Phase I programme (cf. NEACRP-A-papers 584 and 636).

Results for α_v in the Phase II lattice have been found to be positive. Consideration of the individual void coefficient components show this to be a consequence of the more LWHCR-representative fuel being used in the current experiments (larger fuel diameter, plutonium isotopic composition corresponding to LWR-discharged plutonium, etc.). The most important single effect producing qualitative change to the conclusions drawn from the Phase I experiments is the significantly smaller absolute value of the negative component due to C_8/F_9 (^{238}U capture, relative to ^{239}Pu fission). Also important are the increased positive contributions of reaction rate ratios involving captures in the plutonium isotopes.

It needs to be stressed of course, that the void coefficient for the corresponding power reactor could still be negative once extrapolations have been made for the effects of leakage, temperature, control absorbers, burnup, etc. While the applicability of a zero-power facility for the investigation of such power reactor features is limited, numerical studies have indicated that the uncertainties involved in such extrapolations could be much less significant than the basic α_v uncertainties being resolved by the present type of experiments.

An important aim of the PROTEUS-LWHCR Phase II programme has been the improvement of measurement accuracies relative to those achieved in Phase I. Thus, efforts to develop reliable experimental techniques for the determination of k_∞ via cell worth measurements - in both dry and wet test lattice configurations - have been continuing and have already provided invaluable supplementary information to that obtained via buckling measurements. New experimental techniques have also been applied for the determination of ^{241}Pu fission and ^{242}Pu capture - reaction rates which in the Phase II lattices are of much greater importance than they were in Phase I.

Investigations related to an intermediate H_2O-voidage state for the current Phase II reference lattice have been carried out using Dowtherm as moderator, as was done in Phase I. The measurements have been completed and are currently being evaluated, so that void coefficient results over the corresponding partial voidage ranges will soon be available. A series of experiments concerning the determination of k_∞ in uniformly poisoned test lattices has now been initiated, in order that more direct information pertaining to the effects of control absorbers can be obtained. Heterogeneity effects will also be investigated in this context by carrying out appropriate reaction rate distribution measurements. For 1988-89, experiments are planned on a wider spaced reference lattice, so that the final set of integral results from the PROTEUS programme should provide a sufficiently broad basis for validating LWHCR physics design tools.

As regards the status of calculational comparisons for the experiments evaluated to date, the inadequacy of LWR cell codes used in conjunction with "standard" data libraries has been confirmed - specific, partly compensating errors being identified in several cases. Analysis, using a data library generated for the WIMS-D code from JEF-1 files, has demonstrated the advantages of applying recent "best choice" evaluations of differential data rather than using libraries incorporating adjustments based on non-LWHCR integral tests (Ref. 1). While certain points remain to be clarified, the overall consistency of results appears to be much more satisfactory in such analyses. Efforts are currently underway to provide detailed numerical checks on the theoretical methods used for the resonance and fast energy ranges, so that the improved quality of results obtained with the WIMS-D/JEF-1 analysis can be put on a firmer footing.

In the context of burnup physics calculations for LWHCRs, an aspect which cannot be directly investigated in PROTEUS-type experiments, two solutions have been prepared for the NEACRP HCLWR-burnup benchmark problem. These employ the EIR Light Water Reactor code system and a generalised version of the Los Alamos DANDE system (cf. section 4), in conjunction with data from ENDF/B and JEF-1, respectively.

3 STUDY OF SMALL DISTRICT HEATING REACTORS

For the SHR, a small heating reactor concept based on BWR technology, further core optimisation studies have been carried out using the EIR Light Water Reactor code system. New design features include the use of fuel elements with a central water channel (so-called "Q" elements) and cruciform control rods with radial-position dependent boron content.

"Q" fuel elements feature a central, non-boiling water channel which has the effect of increasing the reactivity and flattening the power distribution within the fuel element. In the SHR core, with an average void fraction of 15%, the reactivity reaches a maximum if the channel represents about 10% of the cross section of the fuel element. Core simulations confirm that the concept offers a fuel inventory saving of 10% and thus a significant economical advantage. The main effect of the water channel on the physics parameters of the core is to reduce the void coefficient: the (negative) void coefficient is reduced by 50%, but remains compatible with safety criteria.

In the reference design of the SHR core the reactivity is controlled by 9 identical control rods arranged as closely to the core centre as possible. With this configuration the reactivity worth of the centre rod is large enough to jeopardise the safety of the core in the event of an inadvertent withdrawal of the rod. The problem can be overcome by increasing the number of control rods and adjusting their boron content: calculations for a configuration with 13 control rods of three different types indicate a reduction of the maximum reactivity worth by a factor of 2 to 3.

The main features of the SHR core are illustrated in Fig. 1. The core consists of 32 BWR type fuel elements with a central water channel. The fuel is 4.5% enriched UO_2 and, in the inner zone of the core, 12 out of 70 pins per fuel element are gadolinium poisoned. The cruciform control rods are identical in geometry but have three different boron contents.

Details of the reference design of the SHR can be found in Ref. 2.

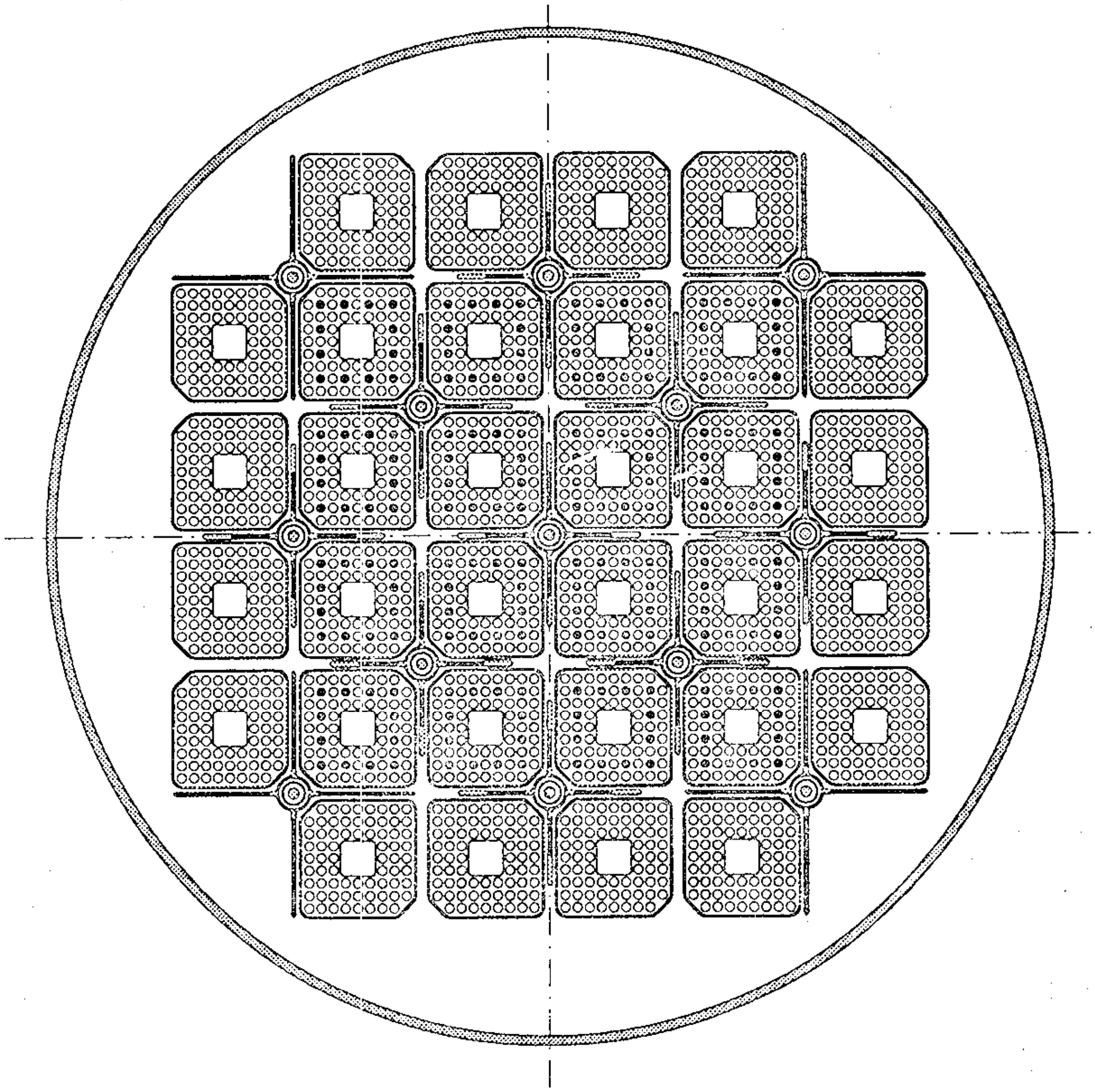

Fig. 1: Cross Section of the SHR Core

4 DATA AND METHODS FOR FISSION AND FUSION REACTOR CALCULATIONS

In the framework of a collaboration with Los Alamos National Laboratory EIR has implemented and is generalising a calculational scheme based on the NJOY cross section preparation route and the DANDE modular depletion system. NJOY processes basic data, mainly taken from the JEF-1 evaluation, and generates group cross sections for DANDE in the so-called "MATXS" format. The present EIR version of DANDE contains modules for resonance shielding and group condensation (TRAMIX), cell calculations (ONEDANT), two-dimensional transport calculations (TWODANT) and depletion calculations (DTOC-CINDER-3). Links to the cell code MICROX-2, the two and three-dimensional diffusion code FINELM and the Monte Carlo code MCNP are also available.

Starting from JEF-1a several group libraries have been prepared. These include libraries in MATXS format using the WIMSD/BOXER group structure, the VITAMIN-J group structure, the General Atomic 193 group structure and the EIR/HRB 308 group structure. The VITAMIN-J library is a coupled library with 175 neutron and 42 photon groups, suitable for fusion reactor and shielding calculations; the other libraries are intended for the neutronic analysis of various types of fission reactors. The 193 group library was also prepared in the format required by MICROX-2 (GAR, FD and GG files). A JEF-1a based point cross section library for MCNP is currently being tested.

TRAMIX is an extended version of the Los Alamos code TRANSX-CTR. Recently developed TRAMIX options include the calculation of energy dependent Dancoff factors and escape probabilities for slab, cylindrical and spherical cells in rectangular, hexagonal and pebble bed lattices. The double heterogeneity of the fuel in pebble bed lattices is also taken into account. Another useful feature of TRAMIX is its ability to automatically repeat resonance shielding calculations in burnup problems.

Developments relating to the one-dimensional S_N code ONEDANT comprise the application of a streaming correction to the diffusion coefficient (e.g. the Behrens-Lohnert correction for pebble bed geometry) and the incorporation of the B_N leakage spectrum calculation method. Dynamical dimensioning has been introduced in DTOC-CINDER-3 to allow general group structures and arbitrary numbers of energy groups to be used. Finally, the development of auxiliary modules for burnup dependent editing of isotopic densities, cross sections and reaction rates is worth mentioning.

The interface between NJOY and MCNP has been further tested and updated in cooperation with Bologna University and AEE Winfrith. In this context current EIR efforts to develop a new shielding formalism for the unresolved resonances may be mentioned. To make MCNP users more familiar with the code, an MCNP workshop was organised at Wuerenlingen in January 1987.

The work on acceleration techniques for transport theory codes has been continued and currently concentrates on the implementation of a QP_1 synthetic acceleration method in the Los Alamos finite element code TRISM. A one-dimensional DP_N version of the method has been tested successfully (Ref. 3).

More detailed information on the LANL-EIR cooperative work in the field of nucleonics and particle transport can be found in a recent EIR report (Ref. 4).

5 TESTING OF DATA FOR FUSION REACTORS

A 175 group fusion library, in which ^9Be and ^7Li data derived from the European Fusion File (EFF) are substituted in the aforementioned JEF based library, has been tested for various fusion blanket designs. In particular, predictions using the JEF/EFF and ENDF/B-V data were intercompared in collaborative exercises with General Atomic and Rensselaer Polytechnic Institute.

In a comparative study with emphasis on tritium breeding, a new lead evaluation by H. Gruppelaar has been tested extensively. Another study, involving intercomparisons for conceptual blankets for NET and TIBER-II, was aimed primarily at investigating resonance shielding effects on the tritium breeding. In this study, which included a one-dimensional sensitivity analysis, it was found that appropriate resonance shielding is important, particularly in the case of water-cooled blankets using tungsten as a structural material. Particular differences in JEF/EFF and ENDF/B-V predictions were identified.

6 LOTUS FUSION BLANKET PROGRAMME

Over the past twelve months activities within the LOTUS fusion blanket programme have been devoted to the conclusion of planned experiments with the Lithium Blanket Module (LBM), and new activation and tritium production rate measurements in various slab assemblies.

The LOTUS-LBM experimental programme has been described in previous activities reports. After completion of the first phase of the programme, and the presentation of results at the Seventh Topical Meeting on the Technology of Fusion Energy (Reno, June 1986), further reaction rate and spectrum measurements were carried out.

Measurements of the neutron spectra at various points in the LBM were performed using a miniature NE-213 spectrometer, specially developed at Lausanne, which can be slid into a LBM test rod. Measurements were made at two depths, 30 cm and 50 cm behind the LBM front face. A total of 18 spectra were taken in the central rod and at some off-axis points. The response functions of the spectrometer were generated using the O5S Monte Carlo code, and the FORIST code was used to unfold the recoil spectra. To validate the response function calculation, measurements at three different energies were made at SIN using a neutron beam produced by the $^{7}Li(p,n)$ reaction, and a time-of-flight technique for selecting the neutron energies. Preliminary analysis shows good overall agreement, although a more detailed analysis may still lead to some modification of the response functions.

New tritium breeding measurements in the LBM, preceded by either a 5 cm thick lead plate or a 6 cm thick beryllium assembly, have also been carried out. Lithium oxide sample disks were placed in four test rods: the central one, and those located 6 cm, 18 cm, and 39 cm off-axis. A sample disk was placed every 3 cm inside these rods, up to a depth of 30 cm in the lithium oxide zone of the LBM.

Activation measurements in blanket assemblies composed of the LBM preceded by either a single thorium oxide zone of 27.7 cm thickness, or the same zone associated with one of the two multiplier slabs (lead and beryllium), have also been made. Five activation reactions were retained: $^{90}Zr(n,2n)$, $^{58}Ni(n,2n)$, $^{93}Nb(n,2n)$, $^{58}Ni(n,p)$, and $^{115}In(n,n')$. The first two are activated almost entirely by 14 MeV neutrons, while the last reaction preferentially registers contribution from lower energy neutrons.

A first analysis of these experiments using the 2D discrete ordinates code DOT-3.5 has been completed. There is good agreement between the computed and measured activation rates to beyond the ThO_2 zone. Discrepancies arise however, as one penetrates deeper in the LBM. Such observations had also been made in similar assemblies without a ThO_2 zone.

These LBM experiments completed the programme worked out in cooperation with Princeton Plasma Physics Laboratory. The LBM was then dismantled and sent back to the United States.

Since there is still a need for additional information on multiplier materials, an independent series of experiments has been devoted to the study of the multiplication performance of lead and beryllium slabs. It could be shown that the ratio of In and Zr activities, averaged over the surface of a multiplier slab, is related to the (n,2n) multiplication inside the multiplier volume. Activation rates on the source end of lead and beryllium multiplier slabs of different thicknesses have therefore been measured and analysed in R-Z geometry using the 2D code DOT-3.5 together with a 46 group EFF library. Comparison of the computed and measured radial and surface-averaged activities shows good agreement for the ^{9}Be multiplier. For Pb the results are less satisfactory and analysis is continuing.

Numerical studies have also been carried out for a new type of blanket module incorporating a lithium-lead alloy. The aim of this work is to identify a realistic blanket concept which would be particularly suited for testing in the LOTUS facility.

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REACTOR PHYSICS IN THE UNITED KINGDOM 1986 - 87

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1 News Items

Approval for the 1175MW(e) PWR at Sizewell has been given and construction has begun. The extended timescale can be summarised as:-

January 1981	CEGB apply for permission to build the PWR.
January 1983 to March 1985	Sizewell B Public Inquiry (estimated cost £20M).
December 1986	Inspector's Report (3000 pages) recommending that approval should be given, is presented to the Secretary of State for Energy.
March 1987	Secretary of State for Energy gives his approval.
June 1987	Site Licence issued by the Nuclear Installations Inspectorate.
July 1987	Construction begins. Will take 7 years and will cost £1.6 x 10^9.

Some of the anti-nuclear groups are still planning to challenge the decision to go ahead in the higher courts of the land. An application for a second PWR, at Hinkley Point, was finalised during August 1987. It will be subject to a full planning inquiry, but not on the scale of that for Sizewell.

The public inquiry into the joint UKAEA/BNFL outline planning application to build the European Demonstration Reprocessing Plant at Dounreay ran from April to November 1986. The plant is designed to reprocess fuel from the next European Demonstration Fast Reactors. The first part of the inquiry report, on findings of fact has been issued for comment to the parties involved. The second part of the report which will include the Reporter's recommendations will follow later this year.

Fuel loading in the first of each pair of AGR reactors at Heysham and Torness commenced in July. The new fuel developed by BNFL has a more resilient graphite sleeve and should facilitate on-line refuelling. The tests following fuel loading should be completed to allow full power operation early next year. Construction of the second reactors at each site is about complete and substantial power should be produced in 1988. Construction of the first reactors has been delayed by about 10 months for modifications to solve a control rod oscillation, caused by an uneven flow of coolant gas at a new design of inlet ports.

The recent performance of the operating AGR power reactors has been disappointing, with the exception of the South of Scotland Electricity Board's reactors at Hunterston. The AGR and Magnox reactors at Hunterston provided more than 45% of Scottish electricity generation for 1986-87 with a load factor of about 80%.

In May, the UK Government announced that the Nuclear Industry Radioactive Waste Executive (NIREX) would abandon its plans to put low-level nuclear waste in shallow trenches at one of the four possible sites. Opposition had been encountered at all the sites. Various deep disposal systems are now being considered again, with low-level and intermediate-level waste being stored in the same facilities.

The UK Atomic Energy Authority recently issued its Annual Report for 1986-87 (and an associated video film) which covered its first year of operation under a Trading Fund, as defined in the 1986 Act. The Government has decided that the Authority should operate more like a commercial concern, earning from sales and its knowledge, endeavouring to obtain a return of more than 5% on its capital, with a limited borrowing power from the Government. Those financial objectives were just achieved during this first year.

The Annual Report highlighted a variety of interesting projects:-

a) The Joint European Torus (JET), being developed at Culham in collaboration with 11 other European countries, is approaching the levels required for the three main parameters of a plasma in an operational fusion reactor:-

	Temperature °mC	Time secs	Density 10^{12} particles/m^3
Desired	100-200	1-2	2-3
JET peak	140	0.9	1.2
JET typical	70	0.6	0.5

b) A Laser Isotope Separation technique to enrich uranium is being developed by the Authority and BNFL. Laser beams with an accurately-chosen frequency fired at uranium vapour will selectively ionise U235 which can then be separated from the un-ionised U238 by an electro-magnetic process. It has the advantage over centrifuge and diffusion methods in that it is a one-step process.

c) A new method of reprocessing uranium fuel is being studied. A unit called a vortex contactor uses fluidic principles associated with the energy in liquid. It therefore has no moving parts and negligible maintenance. The spent fuel is dissolved in nitric acid and mixed with an organic liquid. The plutonium and uranium transfer from the nitric acid to the organic liquid which can then be separated by a vortex effect in a suitable container. Several of these containers would be used in a line.

d) Now that the Windscale AGR is being decommissioned, there is a need for a facility to test AGR fuel pin and clad performance. A special 11-foot high pressure loop (PAT) is being introduced into the materials test reactor PLUTO at Harwell. An instrumented calibration rig provides temperature distributions. There is a remote handling facility nearby which includes a gas sampling rig for fission products and equipment to determine pin dimensions and examine the micro-structures of the fuel and clad. Already burnt-up pins can be used in the loop which should be operational by the end of 1987.

e) A Spinning Cylinder Test Rig in which high stresses can be produced in a thick steel cylinder has successfully completed commissioning trials at Risley. By high speed rotation (up to 3,500 rpm) and water-spray quenching, the facility can simulate complex stresses produced by simultaneous pressure and thermal shock loading in a thick-section pressure vessel. An experimental programme jointly funded by the Department of Energy and the CEGB will determine the effect on the growth of incipient cracks in PWR pressure vessels.

A new inspection validation centre was opened at Risley in June. Miniscule faults can be induced in full-scale sections of a pressure vessel and inspectors are tested in their ability to locate the defects. The centre is expected to find wider applications, for example in off-shore structures and boiler plant.

A £2M rig, ACHILLES, designed to investigate clad ballooning in PWR's has been built at Winfrith. The aim is to confirm results from calculations which indicate that following a major cooling water pipe fracture in a PWR, cooling from the emergency cooling water will be sufficient. The rig uses electrically heated model fuel assemblies of the same geometry as those for Sizewell B. They can be heated to 1100 ^{0}C and contain more than 400 thermocouples. The current programme includes experiments on two fuel assemblies - one normal and one with a flow blockage simulating clad ballooning.

3,000 houses in Devon and Cornwall are being monitored for the presence of radon gas by the National Radiological Protection Board. It is believed that some occupants are exposed to several times the radiation levels permitted for workers in the nuclear industry and that nearly 1000 deaths a year from lung cancer in Britain are attributable to radon daughters. The Building Research Establishment is investigating ways of reducing exposure in homes. These include sealing floors, increasing ventilation and installing a pump to remove radon.

2 Thermal Reactor Physics

2.1 Computing Environment

A very significant development of working environment has been the steady conversion of codes from the IBM mainframe to a local area network (LAN). This was a major undertaking mainly because of methods adopted in the past to simulate dynamic storage allocation, that are not supported by Fortran 77. The conversions to Fortran 77 do have the benefit of generating much more portable codes than we have ever had before.

The LAN is supported primarily by SUN microcomputers, but is also linked via VAXs to the mainframes for transmission of codes and data. The SUNs have proved to be a highly cost-effective way of working, offering cheap computing, advanced facilities (bit-mapped window managers and excellent debugging tools etc), enhanced user-productivity and high reliability. The machines have imposed no serious limitations on job size despite their current speed of only about one tenth that of the IBM 3084Q at Harwell.

2.2 WIMS Codes and Data

A new WIMS nuclear data library (the '1986' library) was released. The three main changes from the '1981' version were a revised normalisation of the U238 resonance integrals (based on several integral evaluations), a renormalisation of all fission product cross-sections (again based largely on integral experimental evidence), and the revision and addition of several higher actinide files based on the JEF1 library.

As mentioned above all the WIMS codes have been converted to a portable Fortran 77. This includes the old stalwart WIMSD, the more recent modular code WIMSE and the light water version LWRWIMS. Also now being included in the WIMS 'package' is the group Monte Carlo version of MONK known as MONK-5W.

All four of these codes have been tested more thoroughly than ever before (mainly due to the availability of the SUN microcomputers). More effort has been recruited into this area to rebuild a viable team capable of extending the present capabilities of the codes. In particular it is intended in the longer term that WIMSE should become the universal WIMS code with appropriate sections lifted from WIMSD and LWRWIMS, and new models added where required.

Two significant developments have been made to MONK-5W:-

(a) a resonance subgroup model has been added (based on established WIMSE methods) so that it can now be run directly from a WIMS 69 group library without preprocessing to generate effective shielded cross-sections for the resonance groups.

(b) explorations have been made into the possibilities for using MONK-5W to do burnup calculations. The code has been developed to accept tables of cross-sections (generated by WIMS) as a function of absorptions. This method relies on the WIMS depletions being carried out with a representative neutron spectrum, but the method has proved very successful for special applications.

2.3 Monte Carlo Developments for AGR Studies

A significant new development was required to model not only the axial gaps in AGR fuel stringers but also the gadolinium poison toroids. This was seen as too difficult a problem to tackle by deterministic transport methods (at least for a design calculation method) and a perturbation Monte Carlo approach was adopted.

The method known as MAX has been tested for an AGR lattice cell, calculations being carried out with an explicit representation of all the details of the cluster and moderator including the end cap, alumina pellets and fuel pins with central voids and grids. Comparisons of both radial and axial fine structure with deterministic methods (2D radial and 1D axial) and experiment (where available) have demonstrated very good agreement. Computing times are two orders of magnitude shorter than those required by conventional Monte Carlo methods.

2.4 PWR Reactor Physics

The primary objective is to co-operate with the CEGB in producing a modular code package for PWR analysis using the best data and methods available to us. The basic assembly code will remain LWRWIMS for the foreseeable future although updating of the WIMS data library, and the user image of LWRWIMS are continuing requirements.

A great deal of work has been carried out by Winfrith and the CEGB to establish the accuracy of the methods being used in both the lattice calculation and the reactor calculation. This has included detailed comparisons between LWRWIMS and Monte Carlo for single lattice cells and for groups of channels, comparisons between the coarse mesh reactor code and fine mesh LWRWIMS calculations on problems up to ¼ core in size, and the simulation of operating reactor histories.

These methods are being embodied in a modularised flexible package known as PANTHER, incorporating a good user image, comprehensive data banking facilities, and ultimately, links to operating reactor databases. The package will be capable of steady state, depletion, and rapid transient calculations, and while the first target is the PWR system, it has sufficient flexibility that AGR specific methods can, and will be, incorporated.

2.5 Irradiated Fuel Studies

A further series of reactivity measurements on samples of fuel irradiated in a thermal reactor has been carried out under contract in the DIMPLE reactor. Measurements were made on three samples of uranium fuel which had been subjected to different levels of burnup. Additional measurements were also made on a set of calibration samples constructed from unirradiated uranium of various concentrations and absorbers such as copper, steel and boron. Measurements were made in three reactor assemblies. The first had a significant epithermal component in the neutron spectrum. In the second assembly the sample was surrounded by a tank of heavy water, inside an annular core. The heavy water provided a thermal spectrum whilst retaining sensitivity of the sample reactivity to fission product absorption. The third had a similar light water central region inside an annular core which rendered the sample reactivity only sensitive to fission production.

The sample reactivities were calculated using WIMS methods and data. A correlation scale of calculated versus measured reactivities was set up in each assembly using the calibration samples, thereby minimising any deficiencies in the calculational method. The interpretation of the irradiated sample reactivities was then carried out by comparison with this correlation. The resultant deviations of the irradiated samples from the correlation was interpreted as an error in fission product absorption. The deviations are still being evaluated but suggest significant variations with burnup.

3 Fast Reactors

3.1 Prototype Fast Reactor (PFR)

During the period July 1986 to August 1987 runs 12 and 13 were completed involving a total of about 130 effective full power days (EFPD). Run 12 began on 20 August 1986 with a full complement of three secondary circuits. On 25 October after 61 EFPD had been accumulated a leak was detected on the inlet side of one of the superheaters. Operation continued to the end of the run on 2 December 1986 on two secondary circuits while this superheater unit was replaced.

Run 13 commenced on 22 December on two secondary circuits until 3 January when the third circuit was reinstated. A period of three circuit operation followed until 27 February 1987 when the plant tripped as a result of a leak in one of the original superheaters. Operation on two circuits then resumed while this unit was replaced. The replaced superheater was commissioned in July 1987 and the third superheater is now being replaced as a precautionary measure. The reactor was shut down for refuelling on 16 August 1987.

3.2 Subcritical Monitoring of PFR

Further comparisons have been made of calculated and measured subcritical count rates during three reloads of PFR. Three low-power monitoring chambers located in the radial neutron shield were used. Counts were only taken with subassemblies present in all positions to avoid problems encountered previously with sodium channels.

6-group source-mode diffusion-theory calculations are based on individual subassembly centre-plane material number densities and source strengths from the routine operational support calculations. Two-dimensional triangular geometry is used. A refinement to the calculations in the adjustment of the axial buckling as irradiated fuel is replaced by fresh fuel. This adjustment was based on separate studies. The total fluxes at the meshes at the edge of the core reflector closest to the instrument were assumed to be proportional to the reaction rates at the instrument positions.

Comparison of calculation and measurement showed agreement within a few percent for the count rate on each detector throughout the sequence of moves. For each individual subassembly exchange, the agreement was within about 2% for each detector and about 1% (corresponding to 18 cents in reactivity) for the average of the three detectors.

3.3 Fuel Reprocessing

The present plant at Dounreay has processed all the PFR fuel to date, a total of over 10 tonnes. A new plant, the Marshall Laboratory, costing £8m has recently been opened. It contains what is reputed to be the largest glove box in the world. This houses a pulsed column test rig, also the largest of its kind. There are a total of six pulsed columns examining different stages of the fuel reprocessing - eg the separation of plutonium and uranium from the fission products, and the separation of the plutonium and uranium.

3.4 The CADENZA Benchmarks

The final report on the International Comparison of Calculations for the pin and plate CADENZA assemblies is being presented to this meeting. All eight solutions submitted showed a positive pin-plate reactivity discrepancy for the normal cores, ranging from 0.0015 to 0.0082 $d\rho$. The mean discrepancy and standard deviation about the mean for the six solutions which effectively have 3D models for both cell types are 0.0047±0.0019. The experimental uncertainty on the difference in k-values for the pin and plate cores is ±0.0013 (1σ).

Extra calculations in simple geometries allowed a study of the contributions to the discrepancies and their variations from solution to solution. Significant variations were found for the reactivity effects of the difference in the homogenised compositions of the pin and plate cells, and spatial and streaming heterogeneities of both cell types.

Consideration of the variations in these calculated reactivities from heterogeneity and the composition change and the experimental reactivity uncertainties gives a random standard error of 0.0015 $d\rho$ on the mean discrepancy of 0.0047 $d\rho$. There may be systematic errors from the cross-section data and in the attempts to represent the plate cell in 3D.

Further results for the k-values of the voided assemblies, element replacements etc are given in the final report.

While the reasons for the differences and discrepancies are not fully understood, the following (1σ) uncertainties are suggested for heterogeneity effects alone.

Plate-cell with a plutonium-metal fuel plate (3D or modified - 1D cell treatment for spatial heterogeneity)	±0.004 $d\rho$
Plate-cell with a mixed-metal or mixed-oxide fuel plate	±0.002 $d\rho$
Pin cell with mixed-oxide fuel	±0.002 $d\rho$

3.5 Development of a Common European Fast Reactor Cell Code

Work has continued on the new cell code, ECCO, which is being developed under the European Fast Reactor Collaboration. The major part of the functional specification for the code has now been completed. The data library for the code will be derived from the Joint Evaluated File JEF-1 (and eventually JEF-2), but for test purposes a fine group library has been generated by converting the FGL5 library into the ECCO format. This has been done as a practical check of the proposed format and to enable comparisons to be made between ECCO and the MURAL code with consistent data. A routine for calculating collision probabilities in cylindrical geometry has been produced. This is based on existing routines but incorporates a number of improvements, particularly in connection with the evaluation of Bickley functions and with the treatment of regions containing material with very small or very large cross-sections.

4 Criticality

4.1 MONK Development

MONK6 has been placed in the Winfrith ANSWERS service with entirely new and high quality user documentation. It is available for lease through the ANSWERS service which offers full facilities for mounting and running the code successfully on a variety of computers including SUN and VAX workstations. Training seminars are provided together with trouble-shooting and a comprehensive advice service on criticality issues.

The option which allows geometry/composition perturbations is now fully operational. Validation checks for both types of perturbation have been carried out. A paper will be presented at the 1987 Tokyo Conference on Criticality in October.

4.2 Analysis of the ZEBRA-8 Test Cells

ZEBRA-8 contained a series of seven k-infinity equals unity cells in plate geometry with fast/intermediate neutron spectra of the type not normally encountered in criticality benchmarks. Detailed representation of the plate cells, including the plate cans and the sheath have been input into the latest version of MONK. The associated DICE cross-section data which is based on UKNDL was used. The k-values and C8/F9 and F8/F9 reaction-rate ratios were compared with the experimental results. Poor agreement was found, the (C-E) differences for k_∞ being -0.02 to +0.10, much greater than differences found for the usual criticality systems with fast or thermal neutron spectra. Generally, both reaction-rate ratios were underestimated, by up to nearly 20%.

This poor agreement contrasted with the excellent agreement for calculated k_∞ and the reaction-rate ratios using 1D and 3D cell representations in MURAL with the adjusted FGL5 data.

Extra calculations in 1D geometry using MONK with 37-group region data from MURAL/FGL5 and with the DICE data confirmed that the poor agreement arose from the DICE data rather than from the MONK code. The MONK custodians are aware of the deficiencies of the DICE data and hope that the situation will improve when JEF-1 data can be used.

4.3 DIMPLE

Analysis of the two critical assemblies simulating loading errors in the 20-compartment boron-steel walled transport/ storage skip is complete. 7%-enriched UO_2 pins were introduced into an edge or one of the centre compartments instead of the usual 3%-enriched pins. This produced significant peaking in the perturbed compartments and asymmetric flux distributions.

k-values were calculated using LWRWIMS-TWOTRAN and MONK6.3 by Winfrith and using the French Monte-Carlo Code MORET by

Dr Poullot of CEA, Fontenay-aux-Roses. MONK6.3 represented each pin explicitly; for MORET cell-average cross-sections for the pin cells were produced; for TWOTRAN some preliminary preparation of cross-sections was necessary to represent the 7% pin clusters in the XY geometry. The reference assembly with 3%-enriched pins in all compartments was also calculated.

Each method gave good predictions of the k-values, agreeing to within about 0.01 with experiment, with a general tendency for the predictions for the perturbed assembly to be higher than for the reference.

In contrast, however, comparison of measured reaction-rate distributions across the skip with predictions from TWOTRAN showed different effects in the two perturbed assemblies. While there is reasonable agreement between calculated and experimental reaction-rate distributions with the perturbation at the centre of the skip, calculation overestimates the fission rates in the 7% compartment at the edge of the skip by about 20%. In fact consideration of the neutron balance for this assembly suggests that decreasing <u>all</u> the calculated reaction rates in the perturbed compartment by 20% would change the total production and absorption in such a way that the overall k-value of the whole assembly would not be altered significantly. This is an example of an accurate calculated k-value not necessarily being an indication that all other parameters of the assembly are well predicted.

Six organisations involved with criticality assessments in the UK carried out calculations for the blind-benchmark subcritical assembly in the skip. Each compartment contained 52 3%-enriched UO_2 pins and the k-value was measured to be 0.711 ± 0.0015. Most participants used MONK6.3, one used MONK5.3 with a correction based on previous analyses and one also used KENO. All the calculated results showed satisfactory agreement with experiment.

5 Shielding

5.1 Benchmarks for Neutron Data Testing

The series of Winfrith data testing benchmarks has been expanded by the completion of a stainless steel penetration benchmark mounted in the ASPIS shielding facility located on the NESTOR Reactor. The experimental data, including integral reaction-rate and differential spectrum measurements are currently being analysed using the Winfrith Monte Carlo code McBEND, the three dimensional finite difference diffusion code SNAPSH and the discrete ordinates code DOT.

The stainless steel benchmark is the starting point for a fully integrated three year programme, JANUS, for fast reactor shielding. The programme will follow the step by step approach, so successfully used in the UK thermal reactor programme, to go from single material data testing benchmarks through to more specific problems such as mixed steel/B_4C shields and local shielding for intermediate heat exchangers. The work is being carried out as part of the European Fast Reactor Collaboration and several shield components which have already been used in the JASON programme in HARMONIE have been borrowed from Cadarache.

5.2 The Winfrith Transportable Neutron Spectrometer (TNS)

Winfrith have developed what is believed to be the first truly transportable neutron spectrometry system. Twenty years of experience with spectra measurement have been incorporated into the design which is based on proven laboratory techniques. The TNS comprises:

1) A probe unit containing the detectors:

    i) An organic liquid scintillator for the detection of neutrons of energy greater than 1MeV.

    ii) Three spherical proportional counters filled with hydrogen to different pressures to cover the energy range from approximately 50KeV to 1.1MeV.

    iii) Bare and cadmium covered BF_3 chambers to provide a measure of the thermal and epithermal flux.

2) A main unit housing the main nucleonics (main amplifiers, ADC, multi-channel buffer etc).

3) A computer-based control and analysis unit for performing on-line analysis of the proton recoil data to provide spectra and an assessment of neutron dose-rate. It is noted that any change in the neutron quality factor (ie dose effectiveness) can be readily accommodated.

The system has been configured such that it can be used by staff without specialist skills in spectroscopy. It has application in research environments where a laboratory spectrometry capability is required; in health physics situations where it can be used to calibrate integral neutron dosimeters and provide an independent measurement of dose-rate.

The TNS has been used successfully during six separate measurement campaigns on operating nuclear plant under a wide range of operating conditions.

AEE Winfrith
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September 1987

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

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U. S. Department of Energy
Washington, D.C. 20545

Introduction

Reactor physics activities in the U. S. have continued to support the design of advanced liquid metal reactors emphasizing improved economics, high reliability, and passive safety. Major efforts have been focused on the General Electric (PRISM) and Rockwell International (SAFR) modular reactor concepts which utilize metal core designs. Additional studies at Westinghouse AESD have designed backup oxide cores for each reactor. Various large core design options have been studied at ANL and WAESD.

Tests in EBR-II and FFTF are continuing to demonstrate LMR passive safety characteristics and to validate calculational methods utilized in the design of passive safety features for advanced LMRs.

A series of critical measurements on large axially heterogeneous cores is in progress at ZPPR in a cooperative program with PNC/Japan. A joint DOE/PNC program of shielding benchmark measurements also is being carried out at the ORNL Tower Shielding Facility to improve the accuracy of advanced LMR shield designs.

Critical Experiments

Two measurement programs have been carried out at ZPPR in the past year. A benchmark space reactor program in the fall of 1986 was followed by the third cooperative JUPITER program between the USDOE and PNC of Japan. Several advances in measurement techniques and calculational capability were also made during the year.

The space reactor program, designated ZPPR-16, tested three critical configurations. Each core was built from enriched uranium, graphite, sodium and stainless steel. The first assembly had 13 internal control rods, with 7 rods designated as operating rods and 6 as safety rods. Withdrawn rods were replaced with BeO followers. A thin BeO reflector surrounded the core. The second configuration had only six internal safety rod positions and a thicker BeO reflector, modeling a design capable of external control. The third assembly was a safety test in which the core was "flooded" with polyethylene and the internal B_4C rods were fully inserted. The ZPPR-16 program was described in the open sessions of the 4th Symposium on Space Nuclear Power Systems, January 12-16, 1987, Albuquerque, NM.

The JUPITER-III international cooperative program between the USDOE and PNC of Japan began in January 1987, with experiments scheduled throughout calendar year 1987. Tests have been completed on the first assembly (ZPPR-17), an intermediate-size, axially heterogeneous LMFBR. The second assembly, a conventional design of 8500 liter core volume, is presently being loaded.

New experimental apparatus has been designed and built during the year, largely for the purpose of dynamic reactivity measurements. The apparatus has emphasized reactivity due to mechanical motion such as expansion or bowing, although the concept is applicable to changes in material composition or density. The apparatus has been successfully employed in sodium coefficient measurements.

The principal calculational capability advancement is the development of an automated reactor modeling code that utilizes a newly employed materials data base. This new capability potentially saves hundreds of lines of input and makes resolution of the model and the actual reactor loading much easier. Input decks have been generated with this system (NIPPER) for the ZPPR-17 cores. A more complex version of the code (called BUILDVIM) is capable of producing detailed VIM Monte Carlo input down to the cladding and gap level.

A gamma heating analysis capability is just about ready for routine application. In this area, experiments have advanced much more rapidly than analysis capability, and the latter is just starting to catch up. A new 3D nodal perturbation capability is also in place.

Passive Safety Tests

Analyses continued for three tests conducted with the FFTF during FY 1986. These tests were part of the Passive Safety Testing program at WHC whose purpose was to demonstrate and accurately quantify the characteristics of the FFTF which make it respond automatically to mitigate the consequences of off-normal events. These data will eventually be used to upgrade existing system models of liquid metal reactors so that similar characteristics of future, advanced reactor designs can be predicted with confidence.

The first of the three tests resulted in data for 198 reactor steady state conditions. The thermal-hydraulic data were normalized to a thermal power assessment at each state. Control rod position changes were converted to reactivity based on accurate calibrations of the rods at zero power, with corrections to the positions for burnup reactivity losses. The states were selected so that the reactivity feedbacks due to fuel temperature changes could be easily extracted.

A preliminary analysis indicated that the fuel reactivity effect (Doppler and axial fuel expansion) could be separated from the structural effects. In particular, it was found that the fuel axial expansion effect in the burned core was less than for a new core (as expected). It was also learned that the core inlet temperature coefficient (core expansion) was larger than earlier anticipated. Finally, it is tentatively concluded that bowing is monotonic for power-to-flow ratios less than unity in FFTF. Analysis of the highest power Loss-of-Flow-Without Scram (LOFWOS) dynamic tests using the Gas Expansion modules (GEM) was performed using the SAS4A/SASSYS core and systems analysis code. Use of the SAS4A/SASSYS code provides the mechanism for validating the reactivity feedback models used for the safety analysis of the SAFR and PRISM innovative reactor designs. An effort was also initiated to use the General Electric ARIES-P systems code to analyze the LOFWOS tests.

The second test consisted of a rapid reduction in flow rate without compensating control rod movements. Initial evaluations of the data indicate the negative reactivity feedbacks appear faster than current models would predict. This causes the reactor power to drop faster during the initial few seconds of such an incident and noticeably reduces the amount of heat that must be removed.

Loss-of-flow tests from power levels up to 50% full power were conducted as the third test series. This series included tests down to pony motor flow and tests all the way down to natural circulation. Special reflector assemblies (GEMs), which contain sodium when the pumps are running and only gas when the inlet pressure drops due to cessation of pump flow, were in the reactor for this test series.

Core Demonstration Experiment

For the Core Demonstration Experiment (CDE) core, an exhaustive design approach was taken because of the increased uncertainty of refueling Rows 1 through 3 and part of 4 with a heterogeneous arrangement of fuel blankets. This approach included analysis, using the latest ENDF/B cross sections, of reaction rates measured in early FFTF operations. Adjustments were made to the neutron cross sections, based upon measured reaction rate data, to reduce differences between measured and calculated parameters. These adjustments, primarily in U-238 capture, were particularly important for predicting reactivity changes with burnup in heterogeneous configurations like the present CDE core.

Space dependent, energy independent axial bucklings were used for the first time to account for axial neutron leakage in 2DB calculations. Previously, these bucklings were also space independent. Three-dimensional calculations of the core were used to calculate the axial buckling for each assembly in the core. This increased complexity was required for the axially blanketed heterogeneous nature of the CDE core.

Criticality and fission rate biases were computed for the CDE core using the ENDF/B Version 5.2 adjusted cross sections. These biases were particularly important since the total reactivity changeout (positive fuel and negative blanket contributions) was five times greater than for any previous cycle reload. The objective was to balance shutdown margin and cycle length to give a good assurance of adequate shutdown margin and sufficiently long (100-150 EFPD) cycles.

The CDE core reload design was accomplished and FFTF was brought to power without the aid of critical assembly experiments. The actual measured secondary control rod shutdown margin of 7.70$ was 82 cents less than predicted. The actual cycle length was 137.7 days at 291MW (EFPD) (approximately 8 EFPD added by a driver replacing a breached test 30 days into the cycle) while the prediction was 125 EFPD (291MW).

The methods described are still being used at FFTF. Good calculation to measurement agreement has been observed for subsequent cycles.

Isotope Production

The first FFTF isotope production test assembly was removed from the reactor after one cycle of irradiation, and the test pins were recovered and examined. The purposes of the test were (1) to validate physics calculations of Co-60 production, and (2) to produce a large quantity of high quality Gd-153, an isotope used for early detection of the bone-thinning disease, osteoporosis. The test assembly contained isotope production pins (thirty two cobalt pins and four europium oxide pins) interspersed with yttrium hydride pins, and was located in the inner row of radial reflectors (row 7). The hydride material moderates high energy core neutrons so that the neutron capture response for all isotopes is predominantly in the epithermal region (1 eV to 1000 eV). Continuous energy Monte Carlo calculations with the MCNP code were used to perform detailed neutronic analyses of the test assembly and to predict final isotopic concentrations. Preliminary results of Co-60 and Gd-153 concentration measurements agree with calculational predictions within 20%, which is encouraging given the complexity of the calculational problem. Interest has been expressed in several other isotopes; and in almost all cases,it is believed that useful material can be produced by activation in a hydride assembly in FFTF. The limiting prediction uncertainty in most cases is likely to be due to uncertainties associated with target isotope capture and fission cross sections. Although the epithermal spectrum is advantageous for production of many isotopes, uncertainty associated with target material resonance shielding effects has in the past complicated making credible estimates of isotope production capability. As a result of the cobalt test, much of this uncertainty has been removed.

During conceptual design of the test, it was recognized that thermalized neutrons exiting the test assembly would cause power peaking in adjacent fuel pins. Experimental support was deemed mandatory to confirm design methods, and ANL-West/ZPPR physicists designed and successfully performed experiments on fuel fission rate peaking. A coordinated analysis effort between ZPPR and WHC personnel resulted in validation of the MCNP design calculations, which in turn enabled irradiation of the isotope production test assembly with high confidence that adjacent fuel pin power peaking would be within FFTF technical specifications.

Physics Calculations for Metal Fuel Core

Neutronic calculations for an FFTF full metal fuel core at the beginning and end of an equilibrium cycle were made using a set of neutron cross sections self-shielded for the appropriate environment. In addition to the two-dimensional reload design type calculations to specify fuel enrichments, this work included three-dimensional calculations of control and safety rod bank worths, individual rod worths, burnup reactivity loss rate, and power defect. A joint effort by Argonne National Laboratory (ANL) and Westinghouse Hanford Company (WHC) is underway to determine the magnitude of biases in calculated values of physics parameters for a metal fuel core for FFTF.

Decay Heat Monitoring

A detailed calculational monitoring of decay heat for each assembly in the FFTF plant is now realized by utilizing a decay heat data base and a user friendly SYMPHONY computer program to access the data base on an IBM-PC. The output is tailored to the day-to-day needs of operations and fuel management, and includes the time-dependent decay heat for an assembly or a specified set of assemblies. The SYMPHONY decay heat data base is updated periodically, including certification, at the end of each irradiation cycle using output from neutronics calculations on a mainframe computer.

Shielding

Measurements for the first experiment in the U.S./Japan collaborative shielding research program (JASPER) were completed in October 1986 and analysis of the experiment is in progress at ORNL and in Japan. The experiment, which consisted of steel, graphite, and boron carbide slabs arranged in either benchmark configurations or design-representative mockups, provided important verification of the latest evaluations of the relevant cross sections. Specifically, a recently prepared iron evaluation was found to give improved agreement with the measurements above 4 MeV relative to ENDF/B-V (Mod. 3) data. More significantly, the latest boron-11 evaluation yielded considerably better results compared to the ENDF/B-V data due primarily to changes in the total cross section at 1 MeV. In both cases, the newer data helped to reduce the nonconservative underprediction of neutron transmission through the shield mockups.

The second JASPER experiment was completed in March 1987 and consisted of homogeneous and heterogeneous mockups of the fission gas plenum region in an LMR fuel assembly. The purpose of the experiment was to verify earlier analyses, which predicted neutron streaming factors of 1.3 and 2 for the two different L/D ratios studied in the experiment. Measurements for these two L/D ratios, however, showed no observable streaming enhancement for the heterogeneous mockups relative to the homogeneous mockups. Detailed postanalysis is in progress in order to understand the current discrepancy.

Test components for the third JASPER experiment were fabricated, but measurements will not begin until the Tower Shielding Facility is restarted. The experiment will provide data on the relative effectiveness of different materials and different geometries for axial shield designs in an LMR.

Shielding analysis supporting the design of advanced LMRs and MHTGRs continued with emphasis on determining design margins and flux monitoring options. Trade-off studies indicated a potential savings of more than $1M for a typical 300 MW(e) LMR module due to the weight and size advantage of using boron carbide to replace some of the steel shields within the primary vessel.

Nuclear Data

Development of the ENDF/B-6 nuclear data file is continuing through the CSEWG evaluation activity. Preliminary data testing of the standards evaluations has been completed. The results indicate some improved agreement with integral experiments. Significantly improved evaluations are expected for the major fission and fusion materials, Be-9; B-10,11; Cr-52,53; Li-6,7; Fe-56; Ni-58, 60; U-235, 238 and Pu-239, 240, 241. In order to meet the goal of a mid-1989 issue date for ENDF/B-VI, it will be necessary to reduce some of the original ENDF/B-VI objectives for evaluations of fission product cross sections, decay spectra, refractory materials, and some of the higher actinides.

Methods Development

Studies of nodal transport-theory methods are continuing. It has been found in some cases that nodal transport computations have not been sufficiently accurate. For example, computational results have sometimes worsened when the mesh was refined. Recent ANL work remedied a weakness in the differencing method which contributed to this phenomenon. The changes in the nodal equations have improved the behavior of the transport solutions so that accuracy improves as the mesh is refined. Another important weakness in nodal transport codes lies in the coarse treatment of the angle dependence of the transverse leakage. In the past, when more rigorous transverse leakage treatments have been put in nodal transport codes, it was often found that the iterative solution process diverged. Previous changes in iterative strategies could force the solution to converge but were slow. A new iterative process under study with rigorous transverse leakage treatments appears, in test computations, to be considerably faster, though more tests are needed for confirmation.

Work on the COMMIX thermal-hydraulic code has focused on the numerical methods by which structure-fluid interaction is treated in the energy equation. It now seems that attempts in the original code to make this treatment implicit were not successful, and an improved implicit approach is now being implemented. Very early tests suggest that this change may cut running times considerably. In a separate effort, an automatic time-step control algorithm was added to the code during the past year.

A rigorous pole representation of the multilevel cross sections which is particularly amenable to reactor physics applications has been developed and generalized to accommodate all ℓ-states of practical interest. The method requires a one-time conversion of the R-matrix parameters to pole and residue parameters in momentum space, where the collision matrix is single-valued and meromorphic. The WHOPPER code developed for this purpose has been completed and tested for cross sections at a wide range of energy and temperature. Test calculations have demonstrated that the rigor of the Reich-Moore formalism is, indeed, preserved for both heavy and intermediate nuclides. Work is underway to include this option in the MC^2-2 code system.

Refinements have been made to an alternative method for Doppler broadening of point-wise cross sections based on the solution of a finite-differenced heat equation; this method is under consideration for the processing of data for the VIM Monte Carlo code. A recurrence relation for the pertinent coefficients required at a given temperature step has been derived to improve the efficiency. In addition, the computational efficiency is further enhanced by a newly developed analytical expression for these coefficients in the case of extreme broadening. The advantage of the proposed method over the widely used Crank-Nicholson method has also been demonstrated by numerical tests.

After a study of several methods of computing the worth of fuel assembly motion, a technique which uses a special number-density weighting scheme was chosen for use in design studies. Other methods, which generally were forms of boundary perturbation theory, gave good results but were more costly to implement. As part of the work, studies were made of the effect of batch-averaging on fuel motion worth analysis and the effect of streaming on core-expansion and fuel-motion worths.

Work began to convert some ANL production codes to the VAX/VMS operating system. The DIF3D code was brought up on an Alliant multiprocessing computer, and an IBM, fully-Fortran-77 version of the Los Alamos TWODANT transport code was made available through code centers. Work began to extend ANL perturbation-theory calculational tools to two-dimensional S_n transport theory and to fuel-cycle analysis.

REACTOR PHYSICS ACTIVITIES AT THE JRC-ISPRA
(September 86 - August 87)

by H.RIEF

Commission of the European Communities
JOINT RESEARCH CENTRE
Ispra Establishment

REACTOR CORE PHYSICS

Perturbation Analysis with KENOEUR

The intensive testing of the Monte Carlo perturbation schemes realized in KENEUR has led to further improvements of biasing schemes used in correlated tracking.
About 30 sample cases dealing with a large variety of possible perturbations have now been prepared. They will be part of the code package to be distributed after approval by the JRC authorities.

A good example for the potential of KENEUR is the analysis of a fast reactor loop experiment which has been performed in a light water cooled driver assembly called SCARABEE at Cadarache. Fig. 1 shows a horizontal cross-section of the Monte Carlo geometry mock-up as it has been used in the calculations. The design was generated by PLOTGEOM.

In a measurement the difference of power generation in the test loop was determined as a function of the material of the hex-can surrounding the 37 pin fuel bundle. In one case the hex-can consisted of a sheet of 1 mm titanium in the other case of one of 2 mm stainless steel. In a KENEUR material replacement calculation the differences of the elements of the eigenvector (of the fission matrix) were determined. (As is well known they are proportional to the spatial neutron power distribution).

Despite the fact that only less than 0.7% of the system's power is generated in the test loop, the relative difference between the two configurations (materials) could be evaluated with a low statistical uncertainty. A comparison between calculated and measured values is shown in the following table.

Coupling factor Essai	n+γ power in test bundle / th.core power	n power in test bundle / th.core power	n power in test bundle / th.core power	idem, JRC calcul.
APL 3	7,35 E-3	6,33 E-3	6,70 E-3	6,75 E-3
APL 2	6,4 E-3	5,38 E-3	5,70 E-3	5,47 E-3

In the frame of a joint international safety study to which also the JRC participates KENEUR is now being used to interprete the neutronics of loss of coolant and melt-down experiments performed in SCARABEE.

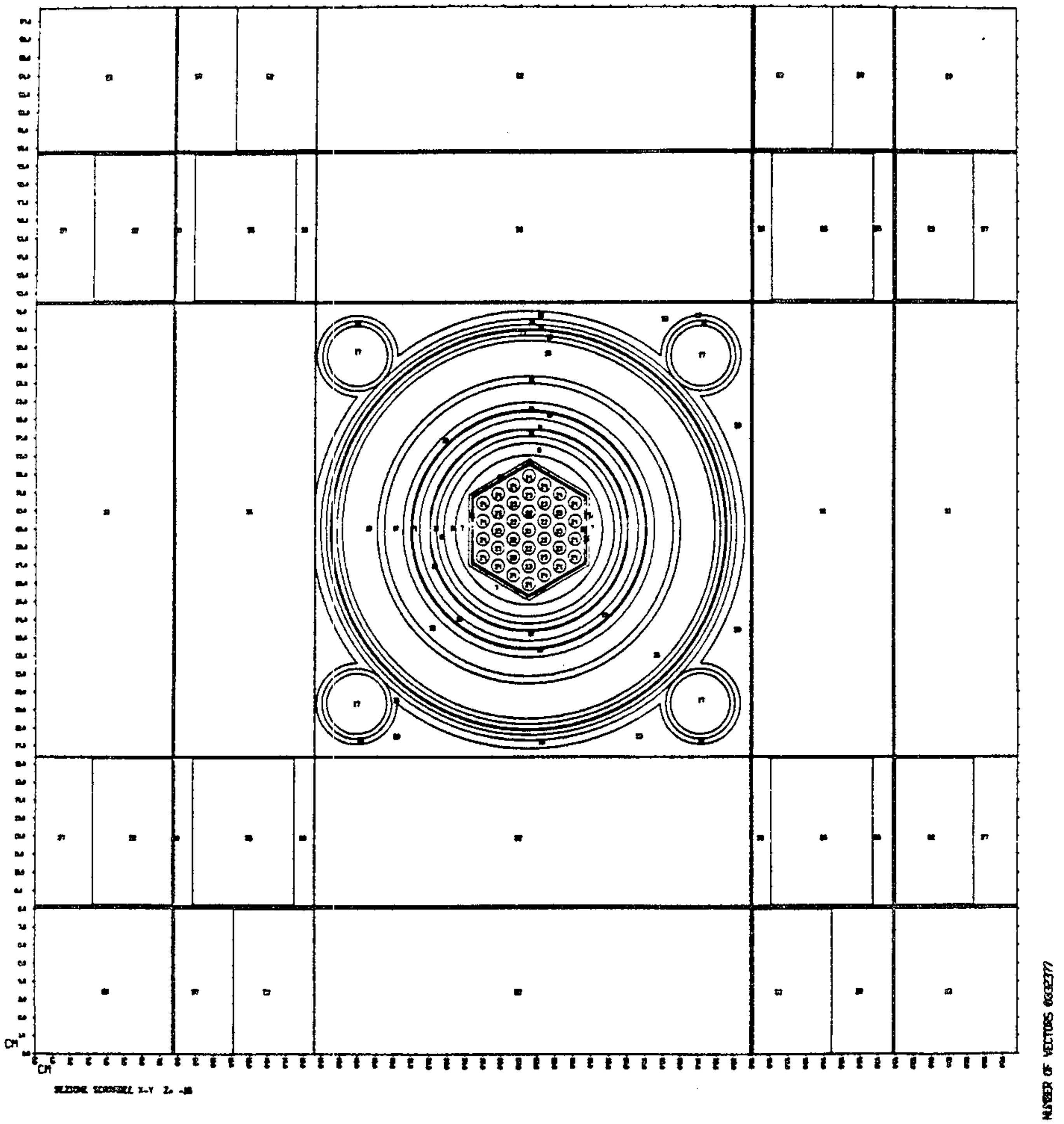

Fig. 1

PLOTKENO : A program for the graphical display of KENO-IV geometries

Experience has shown that the input specifications of complex 3D geometry descriptions as they are used in Monte Carlo codes can only be validated efficiently by graphical display programs. To plot the geometry input of KENO-IV a code, called PLOTKENO, has been developed and tested. It plots the lines of intersection with planes specified by the user. As an extra-option the media specifications of different geometrical regions can be printed onto the design.

The program is composed of the geometry part of KENO-IV [1] and the program PLOTGEOM[2] developed by R.Jaarsma in 1972.

The input specifications for PLOTKENO consist of the Parameter Input and the Geometry Input Data of KENO-IV and a set of Input Data for PLOTGEOM.

Due to the "ARRAY' structure of KENO-IV drawings have to refer to a superimposed coordinate system. An ARRAY consists of a threedimensional composition of BOXes. The content of a BOX is described in terms of an individual coordinate system. The box type arrangement is governed by an "ARRAY DESCRIPTION" in which the lower left corner of box (1,1,1) coincides with the origin (0,0,0) of the coordinate system of the drawing. PLOTKENO uses the geometry section of KENO-IV in a manner similar to a random walk problem. It determines the points of intersection between a geometrical structure and a grid of lines of flight. From these intersections a picture, representing a cross-section of the geometry is generated.

The plot routines are standard "CALCOMP" system routines.

Development of a New Neutronics Module for EAC

The 3-D hex-z nodal transport and diffusion code HEXNOD, which is needed for the European Accident Code, has been further developed and benchmarked by KWU. A review of this work is presented by KWU in the German Progress Report.

The reactor dynamics code Cassandre, from Mol, Belgium, is presently being modified by the Universite Libre de Bruxelles, so that it can be coupled with the nodal HEXNOD code. This work is supported by a JRC contract. The coupled system will then be linked to the European Accident code which requires a 3-D neutronics for taking azimuthal incoherencies into account and a dynamics treatment because molten fuel motion causes considerable flux shape changes.

RADIATION SHIELDING

Iron and Sodium Deep Penetration Benchmarks

The interpretation of the EURACOS shielding benchmarks has been completed by a detailed Monte Carlo analysis using an almost exact 3D geometry model and point cross sections. The calculations were carried out by MCNP in conjunction with an

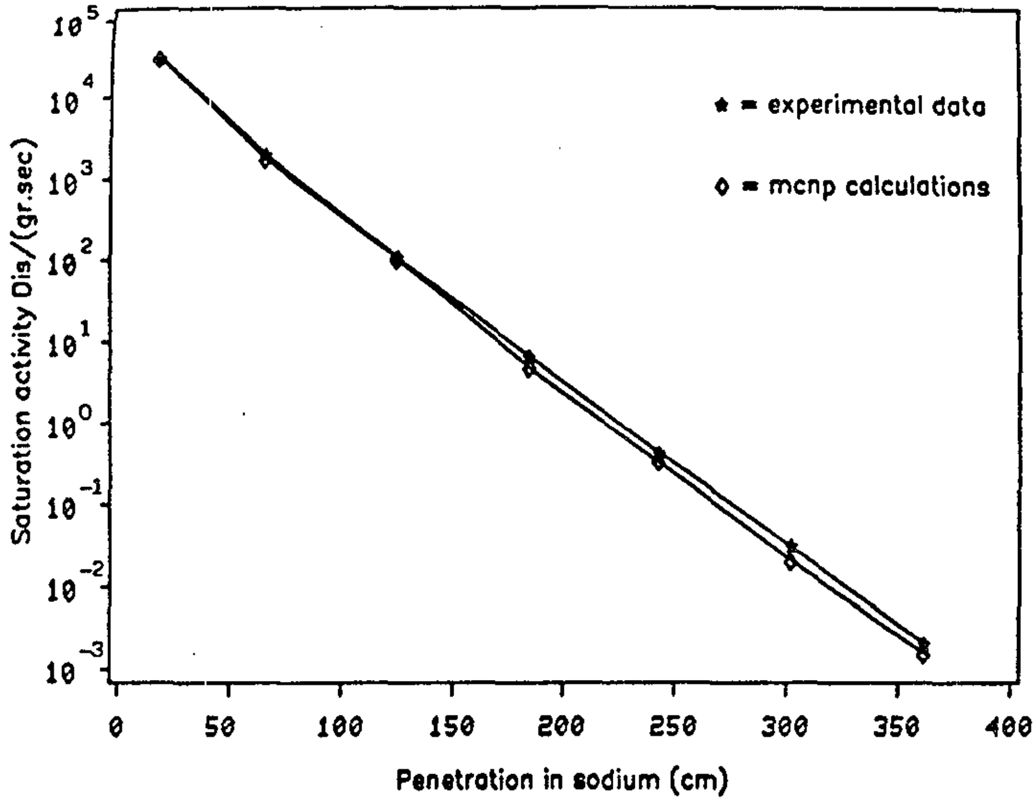

FIGURE 3 : $^{32}S(n,p)^{32}P$ longitudinal activation profile in the sodium assembly.

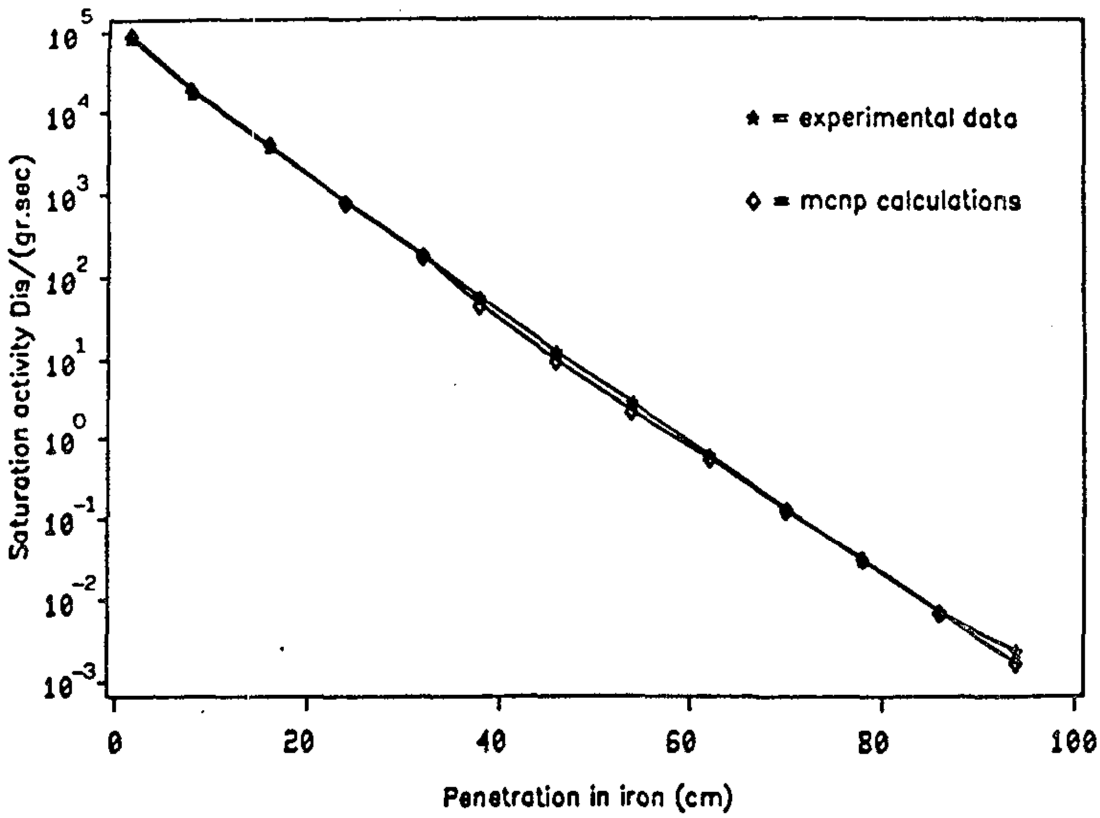

FIGURE 2 : $^{32}S(n,p)^{32}P$ longitudinal activation profile in the iron blocks.

optimized splitting procedure developed in behalf of ENEA by A.Dubi (Ben Gurion University, Beersheva; Israel). Two cross section libraries were applied: the original MCNP-library and JEF-1.

In both experiments, iron (100 cm penetration depth) and sodium (400 cm penetration depth) the attenuation factors for fast neutrons measured by the S(n,p)P reaction are of the order of 10^{-8}.
The comparison with calculated values shows for both experiments and both cross section sets an excellent agreement. The difference between calculations and measurements remains even at large penetration depths below 30%, as can be seen in Fig.s 2 and 3.

SAFETY OF FUSION TECHNOLOGY

1. Guidelines for Shallow Land Burial

This sutdy aims at determining the acceptability of natural elements as constituent for fusion reactor material from the point of view of limited long-lived radioactivity, so that the material could be recycled, or disposed of by near surface burial.

Previous studies (1) have shown that the "structural materials contain the largest fraction of the radioactive inventory of a Tokamak reactor. The reduction of this fraction is the goal for the developoment of low-activation materials.

From the point of view of the potential environmental impact and cost, it is highly desirable that the induced radioactivity decays in a relatively short time, so that the disassembled parts may be withdrawn from the intermediate storage for recycling and reuse within not more than 100 years and that - when this is not possible - the final disposal could be performed by Shallow Land Burial (SLB) and not in other, more sophisticated and expensive repositories.

Norms concerning final repositories are different from one country to another, rather incomplete and focused on fission wastes. Reference will be made in the following to the US code which is the best known. The disposal of radioactive material in SLB in the USA is controlled by the Nuclear Regulatory Commission's document known as 10 CFR 61 (2).

The acceptability of rad-waste for SLB is based on the concentration of radionuclides: first the long-living isotopes C14, Ni59, Nb94 and Tc99, are considered (actinides are not included here); then the relatively short-lived isotopes Ni63, Sr90, Cs137 are taken into account.

These norms do not fix any limit for radioactive nuclides not explicitly mentioned in the table. According to them, there would be no limit, for instance, to the use of Ag in fusion reactor materials since the long-lived nuclides produced by silver are not mentioned in those tables. The same could be said for Bi, Pb, Al and many other elements. Actually, these norms are focused on fission wastes and may not be simply extended to fusion reactor materials. In order to assess whether fusion wastes would be suitable for SLB, it is necessary to determine the radioactivity concentration limits for may other nuclides not mentioned in [2]. For this purpose, an approximate procedure has been devised to determine the concentration limits of long living nuclides in SLB, taking into account three main parameters [3]

- the gamma source density S
- the biological hazard Y
- the half-life T.

2. *A new activation cross section library*

From the knowledge of the activation cross sections of the isotopes and of the decay properties of the radioactive nuclides, it is possible to predict the long-term behaviour of the radioactivity of an element that has been irradiated in a component of a fusion reactor.
In particular, it is possible to determine the maximum weight fraction (MWF) of each element in a mixture, so that the post-service activity does not exceed a given value.

In order to perform these calculations it was necessary to develop a large data base containing activation cross section of all nuclides of interest.
This work, carried out as part of a research contract (4), has led to the creation of a new file, called REAC-ECN from the REAC library (5), with a number of additions and changes. The file contains cross section data for the reactions of all stable nuclides and all unstable nuclides with T greater than one day.

3. *Element acceptability for Recycling and SLB*

Table I reports the results of calculations, which have been performed for all the stable elements. It gives for each element the maximum weight fraction (MWF) which might be present in the new component, so that - after service in the first wall for an equivalent life of 10 MWa/m^2 - the conditions for recycling (R) or shallow land burial (SLB) are satisfied, respectively.

The data listed in Table I should be considered as a first approximation which is useful for making selections and orienting the research on new materials in the direction of the low activation materials. For nearly all the elements, the conditions for SLB are more relaxed than those for R.

The data of Table I hold for a first wall which has suffered an integrated wall loading of 10 MWa/m^2. They can be extrapolated to other values of the integral wall loading or to other components.

As the distance from the first wall increases, the total neutron flux decreases rapidly and the conditions for R and SLB become more relaxed.
The results reported in the table indicate that the recycling after operation in the assumed conditions, will not be allowed until the presence of critical impurity elements (Ag, Eu, Tb,

Maximum weight fraction allowed in the FW to fulfil the requirements for recycling and SLB, after 10 MWa/m²

Element	Recycling	SLB	Element	Recycling	SLB	Element	Recycling	SLB
1 H	NR	NR	25 Mn	NR	NR	62 Sm	10	0.2%
4 Be	NR	NR	26 Fe	NR	NR	63 Eu (0.2)	0.01	0.1%
5 B	NR"	NR	27 Co	5	5%	64 Gd	1	10-50
6 C	NR	NR	28 Ni	0.1%	2%	65 Tb (0.5)	0.01	2
7 N	NR	0.1%	29 Cu	0.1%	0.5%	66 Dy	10-50	0.1-0.5%
8 O	NR	NR	30 Zn	50 %	10%	67 Ho (0.01)	0.05	1
12 Mg	NR	NR	39 Y	NR	NR	68 Er	1-10	100
13 Al	100-500	0.5%	40 Zr	1-5%	10%	69 Tm	100	0.3 %
14 Si	20 %	NR	41 Nb	1	10	71 Lu	1	10 %
15 P	NR	NR	42 Mo	50	500	72 Hf	0.1	0.1 %
16 S	NR	20%	46 Pd	10-50	0.5%	73 Ta	50	NR
19 K	100	0.2%	47 Ag (0.5)	0.01	2	74 W	0.5%	5-10 %
20 Ca	0.1-1 %	1-5%	48 Cd	20	0.2%	75 Re	500	0.1 %
22 Ti	5-10 %	NR	50 Sn	10-50	1-5%	76 Os	1-5	50-100
23 V	NR	NR	55 Cs	100-500	NR	77 Ir (0.1)	0.1	10
24 Cr	NR	NR	56 Ba	10-50	50%	78 Pt	50	1 %
						83 Bi	1	100

Units are ppm or weight percent (%)

NR: no restriction apply to this element

Values in parentheses are the minimum level of impurity attainable by present industrial methods

Ho, Ir) will be reduced below the levels that are attainable by present industrial methods.

On the other hand, SLB does not seem to be heavily affected by the presence of these impurities.

4. General remarks and conclusions

The two concepts of recycling and SLB have been applied as indicators of the long-term radioactive behaviour of fusion reactor materials. There are important qualitative differences between the two concepts which should be stressed.

The capability of recycling components of fusion reactors is essentially an economic issue. It is not clear at present whether it will be more expensive to recycle old material or to produce new material. Recycling should be more attractive and more feasible for the outer components such as the shield and the toroidal field coils, which have larger size, mass and cost, and lower activation level. There is no reason to consider recycling of the first wall as a critical issue. The results of this study show that the presence of impurities will very probably not allow recycling for any steel in the first wall and blanket. The inner shield material could be recycled if the composition is optimized, while the outer shield material could also be recycled in the case of AISI 316. It seems that there is very little interest in developing elementary tailored steels for the purpose of recycling.

A reliable solution of the waste problem would increase the safety, reduce the environmental impact and improve public accpetance of fusion technology. This goal should be pursued for all components of the reactor, including the first wall. AISI 316 as a first wall material would not be suitable for SLB, but an optimized material could be.

The development of low-activation materials for future fusion reactor should rely on the use of the elements that allow SLB.

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ACTINIDE MONITORING

Non destructive assay (NDA) methods are being developed to monitor U and Pu at the various stages of the nuclear fuel cycle including the radioactive waste.
The instruments serve to support the nuclear safeguard inspection of the IAEA and the EC. The present activity covers:

1. Development of instruments and interpretation models
2. Setting up of a laboratory to test the performance of instruments with a large number of different UO_2 and PuO_2 samples occuring at the various stages of the nuclear fuel cycle
3. Training of inspectors of IAEA and EC on individual NDA instruments and in a simulated physical inventory (PIV) of nuclear fuel cycle facilities.

1. Development of NDA Instruments

Following NDA techniques are being developed and tested at the JRC Ispra.

1. Pu isotopic composition determination by high resolution γ-spectroscopy (Pu-meter)

2. U isotropic composition determination by large resolution γ-spectroscopy (U unrichment meter)

3. Pu-mass determination by a determination of the spontaneous fission rate via neutron correlation counting techniques (Swift register, True Correlation Analyzer, Fast time of flight multiplot analysis)

4. Pu-mass determination by calorimetry

5. U-mass determination by active neutron interrogation with an Sb-Be (γ-n) source (PHONID-instruments) counting prompt neutrons

6. U-mass determination by active neutron interrogation with a Cf252 neutron source counting delayed neutrons (DUCA).
7. Automatised waste barrel counter applying neutron correlation counting techniques.

8. Passive γ-scanning (fuel rod scanner, MTR-fuel element scanner, Segmented waste barrel scanner).

Most of these instruments are controlled via personal computers which are equipped with a software for data acquisition and interpretation to get the isotope composition of the fuel mass using stored instrument and nuclear data.

2. Setting up of the Performance laboratory PERLA

During 1987 most of the U and Pu-standards for this laboratory will arrive and become available for the calibration of NDA assay instruments, and for testing their performance with

respect to hard-and software.
The laboratory will be made accessible to none JRC-users either via collaboration or customer contracts.
Samples consist mainly of material existing in fuel fabrication facilities and are characterized by destructive analysis performed in different laboratories:

1. A large family of homogeneous PuO_2 powder batches MOX powders, pellets and pins for low and high burn-up with various seizes

2. A reduced family of MOX powder of various size

3. A family with existing fuel pins of reactors

4. A family of samples with high enriched U (powder, μ-spheres, metal plates, fuel elements)

5. A family with intermediate enrichment

6. A family with low enrichment

3. *Training*

Training courses are being performed at the JRC on the various instruments listed under heading 1. These courses can be performed with small quantities of fissile material (up to 350g U or 200 g Pu) in the laboratories or for larger quantities in the PERLA laboratory.

During July 1987 a "physical monitoring verification" exercise of a Uranium fuel facturation plant for IAEA and EC safeguard inspectors has been arranged. This course covered:

1. Instrument formalisation and calibration
2. Verification planning and verification measurements
3. Data processing to evaluate operator-inspector differences, material balance evaluation and completion of inspection report.

and took place in the PERLA laboratory.

A REVIEW OF ACTIVITIES IN THE FIELD OF FAST REACTOR PHYSICS IN 1986-1987

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I. BN-600 and BN-350 Reactors

More than 6-year -old operational experience of the BN-600 fast reactor atomic power station on the whole is considered as positive. The BN-600 power unit equipment has been mastered and is operating reliably, that confirms the validity of design principles on the pool-type reactor and modular concept of steam-generator. For the time of precommissioning work and nuclear power plant operation the need of modifying certain elements and equipment was brought out with the aim of their updating. This experience is realized at the development of future reactor designs.

By the beginning of 1987 the reactor has generated 21.5 milliard kw•hr of electric energy, for a number of years the BN-600 atomic power plant has been operating with the installed capacity plant utilization factor 72-74% and is shut-down basically for fuel handling. The maximum core fuel burn-up is 7%, in individual test assemblies the burn-up reached 9%. Efforts are attempted to increase core fuel burn-up. Since the middle of 1986 the transition of the reactor to the new up-dated core was initiated. The highest fuel rod thermal load $q\ell_{max} \approx$ 540 w/cm is known to be realized in the BN-600 reactor. Recently it has been settled to reduce fuel rod thermal load by about 15% so as to eusure higher reliability of fuel rods from the standpoint of further fuel burn-up increase and fuel cycle economics improvement. It was obtained as a result of increasing the core height from 75 cm to 100 cm with the same height overall dimensions of fuel assemblies. For reactor transition to a new core a special reload program was developed which was completed in summer 1987 under transient conditions the reactor operates in fact under the rated power and temperaturs conditions.

In the course of the BN-600 reactor transition to the up-dated core with three radial enrichment zones its basic neutronics characteristics were measured. Data of control and safety system efficiency are given in Table 1. It should be noted that in the reactor there were used control and safety system (CSS) rods with various type absorbers-europium oxide, enriched and natural boron carbide, as well as of a "trap" type with zirconium hydride and enriched boron. Calibration characteristics of CSS rods also changed appreciably.

Experimental data obtained at power reactivity factor measurement have demonstrated a good agreement with the values measured at earlier stages. As to the evaluation of contribution into reactivity balance with regard to thermal elongation of CSS drive bars, the resultant data are indicative of the conservation of this earlief-measured effect nature and value (0.1 mm/°C). The result was the continuation of activities associated with the search of the ways to advance and to reduce the price of the existing clad leak-tightuess control systems (CLTC) and the development of new techniques. In this case the simplest models were utilized which reflect the processes associated with delayed-neutron carrier transfer in the reactor primary circuit coolant (delivery time coolant flow rate etc.); evaluation experiments on neutron field perturbation by CSS members were conducted on various power levels, neutron noise diagnostics is being developed with the use of special methods of mathematical processing (correlational, statistical etc.).

Investigations on plutonium build-up in reactor lateral screen fuel assemblies, plutonium build-up in waste core fuel assemblies in the non-destructive method, fissile isotope content control in primary circuit coolant were conducted. The analysis has shown, that the calculated data yield a fair prediction of burnt-up uranium isotope composition and plutonium build-up in lateral screen fuel assemblies. Plutonium content in fuel rods of innev fuel assemblies series is proportional to fuel burn-up fraction and increases by (5÷7)% in fuel rods neighbouring with fuel assembly wrapper tube.

Table 1

Relative efficiencies of various rod groups of the BN-600 control and safety system

№	Rod (group)	April 1986	January 1987	June 1987
1.	Shim rods	1.00	0.90±0.02	0.85±0.02
2.	Automatic control rods 1 and 2	1.00	0.83±0.02	0.79±0.02
3.	Total safety system rods efficiencies	1.00	0.87±0.02	0.83±0.02
4.	Total inner annular row shim rod efficiencies	1.00	0.89±0.01	1.12±0.01
5.	Total outer annular row shim rod efficiencies	1.00	1.21±0.01	1.24±0.01

Notes: 1. In January 1987 practically half the fuel elements in the core were of the advanced type.

2. The result was not corrected for β_{eff} variation.

The results presented in the table show that in the process of transition to the new core significant redistributions in rod efficiencies have occureed, i.e., the central rods became of less efficiency and the peripheral ones- of a higher one.

Development and experimental verification was made for the technique of uranium quantity examination in primary circuit sodium (employing the solid-body track detector method-SBTD), its sensitivity is $\sim 10^{-10}$ g U^{235}/ g Na. The method employed enables us also to evaluate isotope composition of uranium contained in coolant, which in its tur makes it possible to identify the area of uranium ingress into coolant.

The BN-350 reactor is operating steadily with power output 750 MW utilized for electric power production and sea water desalting.

Next year will be 15 years since the day of the plant commissioning. The sodium circuit equipment essentially exceeded the design life-time and its effective operation is in progress. Experimental analyses on structural and fuel materials of the core for the reactors being developed are under way on the reactor. Standard fuel assemblies provide for the maximum burn-up 9%.

Experimental-calculated investigations of energy release fields in periphery fuel assemblies of the BN-350 reactor were completed in 1986. The aim of the investigations was to increase the reliallity of predicting the performance of inpile storage and lateral screen regions.

The measurement was made for reactor radius distribution of $U^{235}(n,f)$; $U^{238}(n,\gamma)$; $Pu^{239}(n,f)$; $U^{nat.}(n,f)$ reaction rates in low enrichment region cells, external screen and in-pile storage. The maitioned reactions except $Pu^{239}(n,f)$ were measured by fuel assembly height in the external screen cell.The measurement were made in the method of activation detectors irradiated in the corresponding fuel assemblies. The calculations were performed with the use of the TRIGEX complex.

The difference of calculated and experimental values of main reaction rates in low enrichment region cells and the screen was as high as 5% except for $Pu^{239}(n,f)$, where the difference is 14% (in the screen).

The disagreements in storage cells are within the limit of 5÷37%, in this case the calculated model requires improvement.

Axial distributions of reaction rates in the screen cell in the central part of fuel assembly (by the core height) are predicted in the calculation with an accuracy not less, than 5%.Disagreements of calculation and experiment in and faces reach 20%.

2. Investigation at Critical Facilities

Investigations of the BN-800 reactor mock-up with axial heterogeneity went on at the BFS-2 critical facilities. For the states with semi-inserted and totally inserted reactivity compensation rods the critical parameters were determined, the core radius- and height- distributions of fission reaction rates were measured, the data on efficiency of shut-off rods mocks-up and reactivity compensators were obtained.

The two-dimensional diffusive low-group (r,z) routine RADAR provides the value K_{eff}= 0.993 ÷ 0.995 taking into account the heterogeneous correction estimate + 0.8% $\Delta K/K$, in this case the error in K_{ef} due to assembly material concentration error does not exceed 0.2% K/K.

The experimental height distributions of uranium-235 fissions in the semi-inserted compensators model have a skewness up to 15% in the radius of compensators location and 6% along the central core axis. The calculation for the simplified (r, z)-model underrestimates this skewness evaluation. Late in the year the uranium fuel in the core centre was substituted by plutonium.

Neutronics characteristics of simple models with plutonium fuel (series BFS-49) were examined on the BFS-1 facility. In the third version the diluent was hydrogen, in the fourth one- sodium.

The core radius and height-distribution of plutonium-239, uranium-235, 238 fission reaction rates are also evaluated with fair accuracy 2-4% by the RADAR routine calculation. The disagreement of experimental and calculated data in lateral and face breeder blankets is conventionally abserved.

Experimental value of sodium reactivity void effect in the BFS-49-4 centre is 30% lower, than the calculated value.

Experimental analysis of the effect of various hydrogen amounts on the criticality value and the degree of neutron field distortion was carried out. Experimental data preevaluation shows, that nowadays the heterogeneous design of BFS critical facilities with plutonium pellets does not allow the experimental results on hydrogen introduction to be transfered on the core with real fuel assemblies with reasonable accuracy. Investigations with fast power reactor fuel rods are required so as to obtain reliable data.

In late 1986 at the BFS-1 facility the BFS-53 assembly was assembled which was an advanced reactor mock-up using metallic fuel in the lower part of the core and oxide fuel in the upper one. The core had the uranium metal-based central axial blanket which in subsequent experiments was replaced by a zone consisting of steel and sodium.

At the COBRA facility the CBR-13 assembly was studied the central part of which was made up of 16%-enrichment uranium dioxide and chromium (and had $K_\infty \simeq 1$).

3. Theoretical and Methodical Analysis of Activities in the Field of Nuclear Data

Theoretical studies on improving the method of group constants preparation for reactivity factor calculation have been carried on. The method of these constants preparation for the correct calculation of reactivity factors as a result of local material density perturbation was developed in the traditional concept of linear averaging of group cross-sections. The essence of the developed approach consists in revealing spectral perturbation group constants.

To realize this approach a complex of programmes GRUKON-SPEKTR was developed and is being modified. This complex of programmes adopts maximum of the detailed information contained in evaluated nuclear data files for the purpose of obtaining spectral perturbations in group constants. It permits the consideration of resolved and unresolved resonance structure of cross-sections directly in neutron energy distributions, gives a detailed description of both scattering anisotropy and in elastic scattering processes. At present the calculations on the evaluation of refined method influence on an accuracy of neutron distribution functionals prediction is under way.

In the field of constant:

- evaluation was carried out for the uncertainties in resonance shielding and their Doppler increment in the range of unresol-

ved resonances U-238, U-235 and Pu-239;
- processing of experimental data on spherical transmission for structural materials was completed with an aim to determine the cross-section of neutron drift under the U-238 fission threshold;
- the first version of the library of reference macroscopic experiments on fast critical assemblies was formed and the system of constants BNAB-78 was verified;
- development of application program pack for the calculation of group constants from evaluated data files - GRUKON-6 was accomplished.

SURVEY OF EXPERIMENTAL STUDIES
OF PHYSICAL CHARACTERISTICS OF
BN-600 REACTOR

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Experimental Studies of Fundamental Reactivity Balance Components

The fundamental reactivity balance components were repeatebly varied in the course of the BN-600 reactor operation in 1985-86.

A number of effects were ouly refined or reevalyated, which do not make essential contribution into reactivity balance with respect to life-time or determined at earlier stages with high accuracy (hydrodynamic and neptunium effects, isothermal temperature effect, prompt component of power effect and a series of others).

The efficiency of the BN-600 control and safety rods was measured affter each overload in the rod-tripping method on the basis of rod-drop inversed kinetics (RDIK). Analysis of the experimental data on control and safety rods effeciency measurements in the course of reactor operation has demonstrated, that the effect of various versions of delayed neutron constants (taking into account burnup) employed by the RDIK technique and dependent on the specific isotope core composition on the results of control and safety rods efficiency does not exceed $\sim$ 4%.

Efficiency of the compensation rods system changes insignificantly (in fractions of β_{ef}-it slightly increases ($\sim$ 5%), in terms of $\Delta K/K$ - it is actually constant). The neutron field increase with the extraction of the internal ring compensation rods in the core centre where the control rods are located was observed result in the control rods efficiency increase and by contrast, the insertion of internal ring compensation rods up to the lower limit switch(LLS) reduces the efficiency of rods located in the core centre. In other words the efficiency of control rods as dependent on the position of internal and external rings compensation rods varies more than 30%.

Calibration characteristics of individual control and safety rods vary rather noticeably as the compensation rods system mores over the reactor run. Comparison of CSR calibration characteristics obtained experimentally with the calculations (TRIGEX) has shawn, /1/, that calibration characteristics of single rods of internal and external rings compensations rods (I and 8) essentially differ in the vicinity of the upper limit switch (ULS).This may well indicate the variation in the nature of interference between the compensation system rods as they are inserted into the core. The compensation system calibration characteristic obtained through calibration characteristics averaging for compensation rods of the internal ring is a better approximation than an averaged calibration characteristic on individual compensation rods. The systematic error of the means of compensation system efficiency measurement by measuring the efficiency of the lower section of its travel and the subsequent recalculation by the calibratien characteristic may be not more than 3÷ 5%. Systematic error of reactivity margin (or reactivity effect compensated by compensation rods movement over a practically linear region of travel (300÷600 mm by rod-position indicator) does not exceed 2.5%.

The shut-off rods officiency measurement have shawn, that it slightly reduced as compared to the starting area (~ 7%), efficiency of the shut-off rod-5 (neutron-trap type) appeared lower (by ~ 15%) than efficiency of the remaining conventional shut--off rods.

The power reactivity factor (PRF) was basically determined for the initial period of reactor runs at the moment of BN-600 reactor's going critical (power increase) after vefuelling. This is due to the fact, that these are just the periods, when the compensation system rods efficiencies were measured. Comparative analysis of PRF values for the latest few runs has demonstrated an agreement within the PRF error limits at rated output and levels close to it.

The BN-600 reactor transition to the modified core was initiated in August 1986 during the regular refuelling with the aim to improve specifications (the increase of fuel burnup fraction, the decrease of energy strength over the zone, longer times of reactor runs etc.)

The transition is carried out successively in several steps, by way of partial substitution of conventional-type fuel assemblies in each fuel handling by the modified ones.

By the present time two steps have already been passed upon which the BN-600 reactor core consists in half of fuel assemblies of this or that type. The basic neutron-physical reactor characteristics were measured upon the completion of each step. Table I presents the measurement results of the BN-600 CSS (control and safety System) efficiency for various steps of transition to a new zone. A trend to sufficiently noticeable variation of CSS efficiency is well-defined as the modified zone is being developed.

Efficiency of CSS rods located in the core centre (shut-off rods 1,2; part of shut-off rods, internal ring of compensation rods system 1÷6 ; central compensation rods) decreased as compared to the data obtained when measuring at an "old" zone (on an average by ~6%). At the same time the efficiency of compensation rods system internal ring 7÷ 18 increased on an average by 23%. It should be noted that in the external ring there is a number of compensation rods members with boron absorber, whose efficiency is ~ 25% higher than that for traditional ones on the europium oxide basis.

The power effect of reactivity was measured in the course of the BN-600 reactor's going into power after each reload by recording the reactor state after a period (1.0-2.0) hr. and the subsequent recalculation, and by introducing small reactivity perturbations (1.0- 2.0) ¢ at various power levels. The obtained data are given in Table 2.

Table 2

Power level,%	17	32	47	58	70	80	90
Power factor of reactivity,¢/%	1.52	1.22	1.06	0.88	0.82	0.78	0.66

Table 1

Control and Safety System Efficiency

Type and number of rods	April 1986 (before transition) % K/K	¢	August 1986 step 1 % K/K	¢	January 1987 step 2 % K/K	¢
АР-I	0.248	34.6	0.239	34.I	0.2I9	3I.3
АР-2	0.26I	37.3	0.242	34.5	0.220	3I.4
АЗ-I	0.527	75.3	0.5I0	72.9	0.467	66.7
АЗ-2	0.546	78.0	0.530	75.7	0.477	68.I
АЗ-3	0.644	92.0	0.587	83.9	0.483	69.0
АЗ-4	0.585	83.6	0.53I	75.8	0.526	75.I
АЗ-5	0.467	66.7	0.448	64.0	0.457	65.3
АЗ-II	0.265	37.9	0.267	38.2	0.270	38.5
КП-ТКI	0.380	54.3	0.374	53.4	0.345	49.3
КП-ТК2	0.386	55.2	0.374	53.4	0.349	49.9
КП-ТК3	0.372	53.I	0.354	50.6	0.349	49.8
КП-ТК4	0.363	5I.9	0.360	5I.5	0.336	48.0
КП-ТК5	0.359	5I.3	0.353	50.3	0.323	46.2
КП-ТК6	0.3II	44.4	0.309	44.2	0.302	43.I
КП-ТК7	0.277	39.6	0.298	42.5	0.37I	53.3
КП-ТК8	0.275	39.3	0.296	42.3	0.36I	5I.6
КП-ТК9	0.286	40.8	0.286	40.8	0.367	52.4
КП-ТКI0	0.297	42.4	0.295	42.I	0.294	42.0
КП-ТКII	0.285	40.7	0.285	40.7	0.382	54.6
КП-ТКI2	0.267	38.I	0.266	38.0	0.370	52.8
КП-ТКI3	0.257	36.7	0.25I	35.9	0.273	39.0
КП-ТКI4	0.234	33.4	0.239	34.2	0.340	48.5
КП-ТКI5	0.246	35.2	0.256	36.6	0.380	54.3
КП-ТКI6	0.270	38.5	0.283	40.4	0.277	39.5
КП-ТКI7	0.264	37.7	0.27I	38.7	0.352	50.3
КП-ТКI8	0.253	36.2	0.273	39.0	0.300	42.8
КП-ТКЦ	0.40I	57.3	0.386	55.I	0.37I	53.0

Note: Gross error of CSS efficiency determination is $\sim$ 2% + 4% (without error of β_{ef})

The mean value of power factor of reactivity in the range (0÷ 97)% $N_{nom.}$ was ~ 1.2 ¢/% , that is in a good agreement with the PRF data obtained in the measurement at previons steps. To evaluate the contribution of CSS drive bars elongation into reactivity balance (displacement of the true position of CSS members with respect to the rod-position indicator) the members shut-off rods 1,2 and compensation rods-1 were adjusted to holding-down stiff clip (HSC) in the course of heat-up and going into power. The obtained data are shown in Table 3.

Table 3

Parameters detected with HSC measurements

Reactor output %	H (mm) - HSC shut-off rods-1	H (mm) - HSC shut-off rods-2	H (mm) - HSC compensation rods-1	Averaged outlet temperature °C	Averaged inlet temperature °C
0	0	5	5	230	230
2.0	4	11	15	280	303
11.0	7	13	25	313	338
32.5	11	18	25	326	420
60.0	20	22	-	342	503
90	23	27	-	344	519

The mean value of relative thermal elongation was (0.08÷0.1) mm/°C, which is indicative of the preservation of the nature and value of this effect measured at earlier stages of reactor operation.

Experiments on Validation of Long-Term CLTC Systems (Clad-Leak-Tightuess Control)

Taking into account an outlook for the further development of fast reactors, which provides for the increase of nuclear fuel burnup (up to 12÷15%) it is essential to search ways for advancing and reducing the price of the existing CLTC systems and for developing new techniques, in this case the principal efforts are focused on the enhancement of sensitivity in these systems and on their utilization for operational preliminary localization (prelocalization) of faulty fuel assemblies in the reactor core.

These ways can be the following:

- utilization of simple models reflecting the basic processes associated with neutron-nuclide transfer in the reactor primary coolant (delirery time, coolant consumption etc.);
- conducting experiments on neutron field perturbation by CSS members at a certain level of reactor output;
- conducting neutron noise diagnostics employing special techniques of mathematical processing (correlational, statistical and other methods).

Simple models of primary circuit coolant motion based on experimental data of sector sodium flow in the upper mixing chamber of pool-type reactors (e.g. BN-600 and BN-800) may appear fairly useful. As opposed to alternative methods they, on the one hand, provide core state monitoring at any reactor operation conditions (power increase, constant level), and on the other hand,- due to common mathematical calculations, this method of localization task solution can be included in the computer real-time operation mode and provide operational monitoring of the core state.

This approach for the solution of the set task can be formulated in the general form as the determination of leak-untight fuel assemblies position by the indications of a few delayed neutron sensors (DN) which monitor coolant flows over individual regions (sectors) of the reactor core. Within this approach simple mathematical expressions can be obtained, which associate count rates of sensor pulses with the region of leak-untight fuel assembly occurrence. In fact the task is reduced to solving the set of K-- equations of the form:

$$\frac{dN_k/dt}{dN_1/dt} = \frac{S_k(\vec{R})}{S_1(\vec{R})} \cdot \frac{\mathcal{E}_k}{\mathcal{E}_1} \cdot \frac{B_k(t)}{B_1(t)}$$

where dN_k/dt - is count rate variation of the K^{th} detector;

$\mathcal{E}_k$ - is efficiency of recording with the DN-detector;

$B_k(t)$ - is a function taking into account neutron--nuclides decay for the time of their delivery to the K^{th}- detector.

$S_k(\vec{R})$ - is a function defining the relative fraction of neutron-nuclides hitting the K^{th}-detector from the neutron-nuclide source (leak-untight fuel assembly) with the coordinates $\vec{R}$.

The approach based on introducing neutron field local perturbation in various core regions (with constant integral power) and the analysis of delayed neutron sensor indications seems to be promising as compared with the alternative approaches to the solution of faild fuel assemblies localization task. The above-mentioned approach(further over compensation technique) takes into account common physical considerations-proportionalety of delayed-neutron precursor yield to the local energy release in the failed fuel assembly location.

The overcompensation technique advantages are due to the fact, that:

- no reactor design variations and special-purpose equipment development are required for realization of the technique;
- the location of failed fuel assemblies is determined with the operating reactor and does not result in the decrease of the plant availability factor;
- the accirate determination of failed fuel assemblies coordinates is feasible theoretically.

However, the practical possibilities of realization of the mentioned advantages is defined by a great number of dissimilar factors and requires a number of scientific and engineering problems to be solved.

The activities in developing the technique of failed fuel assemblies location detection for the BN-600 and BN-800 reactors based on the analysis of delayed-neutron sensor responses to local neutron field perturbations, have been currently initiated.

The calculated and experimental works being under way are focused on the solution of the following problems:

1. Development of the methods of detecting, isolating and evaluating the effective signal in delayed-neutron sensor indications.

2. Development of algorithms for faulty fuel assemblies location determination.

3. Optimization of the technique characteristics.

Experimental part of the activities was carried out at the BN-600 reactor with the use of a standard delayed-neutron sensor of the clad leak-tightuess control and compensation rods of control and safety system.

Calculated part of the activities was carried out at the computer SM-4 with the use of a specially designed library APL - the program of data analysis. The results of the three above- mentioned tasks in short are reduced to the following.

The first task.

Development has been made for the procedure of verification of statistical hypotheses on delayed-neutron sensor signal variation as local neutron field perturbation are introduced. The procedure is based on employing the bootstrap technique, which allows refusal of the detailed assumptions on statistical properties of the signal and realization of a wide spectrum of various signal characteristics for analysis. Fig. 1 represents histograms of mean-signal level difference distribution prior to (X_I) and upon (X_2) introduction of local neutron-field perturbations obtained in the bootstrap method. Case a) corresponds to introduction of perturbations in the region without failed fuel assemblies. In Case b) perturbation is introduced in the vicinity of a failed fuel assembly, that results in statistically significant increase of delayed-neutron sensor indications ($\sim 97\%$ $\Delta > 0$).

Suggestions will be made on the noise filtration methods with the isolation of the component corresponding to local neutron field variations in the delayed-neutron sensor indications. Hewhard's control charts and non-recursive numerical filter based on time series adjustment by parabolas over a various number of points (5+11).

To evaluate the sensor response at the introduction of various neutron field perturbations the algorithm of linear regressive analysis with factor significance evaluation in the boothstrap method was used. This approach allowed the normalized estimates of effects to be obtained and influence of introduced perturbation non-equivalence to be excluded.

The second task

When solving the localization task the reactor core is first divided into M non-intersecting regions. The set of N equations to the series of N experiments each reduced to neutron field deformation and delayed-neutron sensor indication measurements

$$
\begin{aligned}
\Delta n_1 &= \Delta \Psi_{11} q_1 + \Delta \Psi_{12} \cdot q_2 + \ldots + \Delta \Psi_{1M} \cdot q_M \\
\Delta n_2 &= \Delta \Psi_{21} q_1 + \Delta \Psi_{22} \cdot q_2 + \ldots + \Delta \Psi_{2M} \cdot q_M \\
&\vdots \\
\Delta n_N &= \Delta \Psi_{N1} q_1 + \Delta \Psi_{N2} \cdot q_2 + \ldots + \Delta \Psi_{NM} \cdot q_M
\end{aligned} \qquad (I)
$$

where Δn_i - is the measurement of delayed-neutron count at introduction of i^{th}- neutron-field perturbation as compared to the initial level (prior to the experiments);

$\Delta \Psi_{ij}$ - is the measurement of neutron field in the j^{th}-core region at introduction of i^{th}- perturbation;

q_i - is the value specifying the availability of failed fuel rods in the i^{th}- core region.

The problem associated with the solution of system (I) is, on the one hand, the need to increase the number of separated core regions M so as to increase an accuracy of localization and on the other hand, the restriction for the number of experiments N.

The proposed way of solving this problem is based on employment of group arguments control method for the solution of system (I). In this case instructing and testing parts are isolated from system (I). The instructing part is responsible for specially organized sorting of the solutions of various complicacy (by the number of areas containing failed fuel assemblies), and the selection of a solution optimum in accury takes place in the testing part of the system.

This way enables us, first, to evade the requirement $M \leq N$ and, second, to obtaine ($\Delta n, \Delta \Psi$) solutions, which are data input perturbation-resistant.

Another approach to localization task solution being currently verified is an algorithm based on the representation of relations (I) in the form of an unclear ratio. Advantages of this approach are the following:

- simplicity of calculation;
- reduction in requirements for the completeness of the assigned information, in particular, on neutron field variation;
- feasibility of realizing common heuristics considerations.

Fruitfulness of the approach based on nuclear logic is confirmed at the analysis of data on the experiments on overcompensation at the BN-600 reactor.

The Third task.

The reference directions of technique characteristics optimization is minimization of introduced neutron field perturbation amplitude and minimization of time required for measurements.

In the first direction the technique based on the periodic variation of CSS rod position and the analysis of intercorrelational rod position function and delayed-neutron sensor indications is being developed. Nowadays the dependences of sensitivity of the technique under consideration on the signal/noise ration have been obtained, which permit the optimum amplitudes of the introduced variations to be selected.

In the second direction the activities on emplaying methods of experiment planning to minimize the number of experiments N and to select optimum forms of the introduced perturbations $\Delta\Psi$ (from the standpoint of accuracy of evaluating q).

I. Determination of Plutonium Build-up and Uranium Burn-up in Lateral Screen Fuel Assemblies

To evaluate the variations of energy release in the reactor screen due to fission of plutonium being built-up and ^{235}U burn-up the measurements were made for plutonium content and fuel burn-up of waste lateral screen fuel-assembly.

Investigations were carried out with the use the specimen-witness technique. A specimen representing a capillary tube filled with depleted uranium oxide of the same composition as in fuel assembly screens was irradiated in one of the screen fuel assemblies during the whole life-time. Upon onloading fuel assemblies from the reactor the specimen was extracted from it and dispatched to radiochemical analysis.

Uranium and plutonium were reparated and their isotopic composition was determined with the use of mass-spectrometer, α-spectrometer and γ-spectrometer techniques.

Burn-up of uranium and accumulated plutonium was determined by the build-up of Cs-137 and Ce-144 with the use of the γ-spectrometer technique, as well as by the build-up of Nd-148 with the use of mass-spectrometry and isotopic dilution method. Calculated analysis of the experimental data was performed with the results of the three-dimensional geometry reactor calculation in the code TRIGEX. Results of the experimental and calculated data comporison are given in Table 4.

Table 4

Comparison of Experimental and Calculated Results of Uranium Isotopic Composition and Plutonium Build-up in Waste Fuel Assembly of BN-600 Reactor Lateral Screen

Uranium	235	236	238
Calculation/experiment	0.965 ± 0.048	1.00 ± 0.17	1.0000 ± 0.0002
Plutonium-239 Calculation/experiment	1.0000 ± 0.0008		

Burn-up obtained through averaging of the results for Cs-137, Ce-144 and Nd-148 is 0.395 ± 0.010% t.a. As can be seen from Table 4, the calculation adequately predicts isotopic composition of uranium and plutonium build-up.

Determination of Plutonium Build-up in Waste Fuel Assemblies of BN-600 Reactor Core in Non-Destructive Method

Gamma-X-ray-spectrometric analyses of high-enrichment waste fuel assembly region were conducted in the BN-600 reactor with an aim to determine fuel burn-up and plutonium build-up.

Elementary plutonium-uranium ratio in individual fuel rods of fuel assemblies was determined. The technique of this ratio determination is based on measuring uranium and plutonium X-radiation excited by fission product γ-quanta.

Relative fuel burn-up fraction was determined by the measurement results of selenium-144, ruthenium-106, cesium-137, strontium-95 fission product γ-radiation intensity, as well as Mn-54 in fuel rod cladding /2/.

In the course of measurements fuel assemblies or individual fuel rods moved in frout of the collimator in the hot cell wall and were immobilized in the prescribed position with a special device.

The results of relative burn-up fraction determination indicate essential variation of neutron flux in fuel assembly cross-section.

Plutonium content in fuel rods of inner fuel assembly series is proportional to fuel burn-up fraction and increases by (5+7)% in fuel rods next to fuel assembly wrapper. This increase may well be explained by spectrum softening in the vicinity of fuel assembly wrapper, which is confirmed by the decrease of $^{54}Fe(n,p)$ reaction rate (threshold $\sim$ 3 MeV) in the vicinity of the wrapper tubes (by $\sim$ 10%).

Relative axial distributions of selenium-144 and strontium-95 fission product concentrations are in a good agreement with plutonium concentration distribution since these products migrate insignificantly.

The value Pu/U was determined over the whole fuel assembly making use of the measurement results of Pu/U space distributions over the fuel assembly.

The value (Pu/U) of the fuel assembly determined in this way agrees with the calculated value:

$$M/C = 0.95 \pm 0.04$$

Fissile Isotope Content Monitoring in I-circuit Coolant of BN-600 Reactor

The technique of fissile isotope (U^{235}, U^{238}) content measurement in I circuit coolant was introduced at the BN-600 reactor using the solid-body track detector method (SBTD) (sensibility $10^{-10} \frac{g\ U^{235}}{g\ Na}$). The measurement procedure consists in the following. The sample of I circuit sodium is taken to sodium nitrate and is moulded into pellet later adopted as SBTD radiators. Side-by-side the reference pellet-radiators with the know uranium content are fabricated in the same technique.

SBTD with radiators of the sample under analysis and the reference one are irradiated in a high-density neutron flux. The fissile isotope content in coolant is determined by the results of comparing the number of fissions of the sample under analysis and the reference one.

In the investigations immediately following the low power physics test of the reactor the values $(3 \div 3.5) \cdot 10^{-9} \frac{g}{} U^{235}/g$ Na were obtained, that corresponds to the amount of uranium in the core due to fuel assembly surface contamination. Uranium surface contamination of fuel rods was also determined employing the SBTD method. For this purpose the fuel rods selected at fuel assembly fabrication were irradiated together with quartz SBTD fixed on them in the thermal neutron flux with the known characteristics. As a result of the conducted measurements the values of fuel rod surface contamination were obtained within the limits $(2.5 \div 5.5) \cdot 10^{-9}$ gU^{235}/cm^2.

The method employed enables also the in-coolant uranium isotopic composition to be evaluated. This is attained by irradiation of samples in thermal and fast neutron flux with the know value of ratio σ^8/σ_f^5. The data on isotopic composition permit the region, uranium flows into coolant from, to be determined.

(The figures mentioned in this report have been omitted).

References

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