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NEA COMMITEE ON REACTOR PHYSICS

REACTOR PHYSICS ACTIVITIES IN NEA MEMBER COUNTRIES

October 1988 - September 1989

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

NUCLEAR ENERGY AGENCY
38,boulevard Suchet, 75016 Paris

3. LWR In-Core Fuel Management

STUDSVIK NUCLEAR and its subsidiary STUDSVIK of America are continuing the work to develop and further improve STUDSVIK's computer codes and methods for LWR ICFM. During the year since the previous report was submitted several new companies have joined the group of users of the STUDSVIK Core Management System, STUDSVIK-CMS, in Europe, Japan, Taiwan and the US.

Efforts have been given both to the area of benchmark and validation and to improvement of the Code Package itself. Thanks to the large number of users, now about 40, a major benchmark and validation effort is carried out by the users. The appended list of references is a selection of papers that have appeared since the previous report.

Work has been done on the major codes of the CMS, CASMO and SIMULATE, but also on auxiliary codes. A code for calculating fuel temperatures, INTERPIN CS, has been introduced. Other new auxiliary codes are S3CORE, which calculates data for CECOR and INCORE, S3POST, a postprocessor for SIMULATE-3 output, and SLICK, which is a linking code for SIMULATE-3 data to core kinetics codes.

A benchmark study on application of CASMO and SIMULATE on MOX fuel has been carried out. A Shutdown Margin Model has been developed for SIMULATE and applied with good results on the Chin Shan Unit 1 SDM incident.

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This document is a compilation of national activity reports presented at the Thirty-Second Meeting of the NEA Committee on Reactor Physics, held at Argonne National Laboratory, Argonne, Illinois, USA, from 9th - 13th October 1989.

REACTOR PHYSICS ACTIVITIES IN AUSTRALIA

October 1988 - September 1989

Compiled by G.S. Robinson
Australian Nuclear Science and Technology Organisation

BACKGROUND

It remains unlikely that Australia will introduce nuclear power stations in the foreseeable future. Government policy that uranium exports are permitted from three particular mines has been the subject of further debate but remains unchanged. The only research areas closely related to nuclear energy to have substantial support are radioactive waste management (particularly SYNROC), the environmental impact of uranium mining and nuclear safeguards. Following the formation of ANSTO from the AAEC in 1987, there has been increased emphasis on the application of nuclear science and technology to Australian industry, collaboration with industry and commercial projects.

RESEARCH REACTOR STUDIES

Neutronic Modeling of HIFAR

The new code HIFUME for performing 2-group XY calculations of HIFAR with global microscopic burnup has been completed. The code makes use of standard AUS modules for neutron diffusion and burnup. However, an attempt has been made to hide AUS from the normal HIFUME user so that it can be used for routine fuel management calculations on HIFAR. The required 2-group microscopic cross sections as a function of fuel element irradiation are generated by condensing multigroup cross sections from a cell calculation over standard RZ reactor fluxes using bilinear weighting. Comparison with previous AUS calculations using macroscopic burnup (i.e. macroscopic cross sections as a function of irradiation) gave excellent agreement for reactivity at shutdown. HIFUME includes transient fission products to enable calculation of reactivity at startup. Comparison with measured reactivity loss in each reactor operating cycle was very good. All comparisons were made over about fifty cycles.

METHODS OF CALCULATION

A major effort has been made to convert most of the neutronics codes used by ANSTO to portable FORTRAN 77. This work followed from the expectation that ANSTO's IBM mainframe would be replaced in mid-1990. Almost all of the AUS system; the ENDFB processing codes XLACS, MACK and SMUG; resonance calculation codes; ANISN and ZAPP have been converted and tested either on a Pyramid or Convex super-minicomputer as well as on the IBM mainframe. In the conversion, the use of real or integer variables to store character data was generally retained. The assignment of character data to these variables is now performed by simple portable FORTRAN functions. In principle, this treatment of character data allows operation on any machines with four or more characters per word. Another major difficulty was the elimination of the ENTRY statements which had been used for variably dimensioned arrays. The coding of the system component of AUS had to be rewritten to eliminate assembler language and dependence on the MVS operating system. A version of the new AUS system has run successfully under the UNIX operating system.

REACTOR PHYSICS ACTIVITIES IN BELGIUM

October 1988 - September 1989

compiled by P. D'HONDT (SCK/CEN, MOL)

INTRODUCTION

Already quite early Belgium started some "pioneering" work within the nuclear power industry. In 1956 the first research reactor (BR1) became critical, shortly after a material testing reactor (BR2) and an industrial reactor (BR3) with a power of 11.5 MWe were builded; the latter being the first PWR of the Westinghouse-type in Europe. This reactor was shut-down in 1987. Even before the first oilcrisis the France-Belgian power reactor in Chooz was put into service in 1967. In 1969 the power reactors Doel 1, Doel 2 and Tihange 1 were ordered.

After the consecutive oilcrises of 1973 and 1979 the share of electricity within the global energy consumption was growing rapidly. For example, within the domestic sector, the total electricity consumption doubled in the period 1973-1985, while the global energy consumption almost stagnated. Within the same period the demand for electricity in the industry growed by 14 % in spite of a 30 % drop in the total energy demand.

To follow this demand, the production capacity of electricity had to grow fast. After the first oilcrisis in 1973, it was decided to fasten the execution of the nuclear programme. On the existing sites of Tihange and Doel four new units were installed, from which the last one was taken into service in 1985. A summary of the nuclear power reactors on Belgian territory is given below.

Reactors	Net installed power (MWe)	Start-up
Tihange 1	870	1975
Tihange 2	900	1983
Tihange 3	1020	1985
Doel 1	400	1974
Doel 2	400	1975
Doel 3	900	1982
Doel 4	1010	1985

The net installed power in nuclear energy amounts to 39 % of the total power available for electricity production in Belgium. However, the nuclear power reactors do guarantee about 66 % of the national electricity production which correspond to about 21 % of the global energy consumption.

Due to the better situation with respect to oil procurement and the affected public opinion with respect to nuclear power after the Chernobyl accident, this explosive growth of nuclear power in Belgium has slowed down the last years. The decision to build a new power reactor (N8) of 1400 MWe in Doel was not taken. Furthermore, with respect to fast reactor research, Belgium did not sign the cooperation agreement for development of the EFR (Europen Fast Reactor).

This situation will affect the reactor physics activities for the forthcoming years. With respect to fast reactor research, activities might slow down. With respect to thermal reactor research a stagnation is not impossible, reorientation of the reactor physics activities to more safety oriented research is plausible.

THERMAL REACTORS

VENUS-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 : 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.

The VENUS-2 core has been defined in order to examine how far major conclusions from the VENUS-1 study can be affected by significant amount 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 used experimental techniques for determination of core power distribution, axial buckling and neutron propagation were very similar to those applied in the VENUS-1 campaign.

The analysis of VENUS-1 and VENUS-2 experiments is nearly completed and final reports are practically ready for publication. Most of the results, including calculated over experimental values (C/E) were previously presented [1][2][3] and [4].

An agreement has been reached with the USNRC to pursue the experiments in VENUS in 1988 with the assistance of US laboratories. The objective was to support the PLSA (Partial Length Shielded Assembly) concept [5], aiming at a strong decrease of the neutron fluence in the circumferential welds of a vessel.

For this purpose special assemblies, in which fuel pellets at the extremities of the active height have been replaced by steel and the remainder of the fuel has a reduced enrichment, are loaded at the periphery of the core, around the azimuthal maximum of the fast flux distribution in the vessel.

Benchmarking of this methodology has been performed in VENUS through substitution of simulated PLSA assemblies to standard ones. The PLSA zones in VENUS (VENUS-3) encompass five fuel rows in two opposite branches of the cross-shaped core already designed for the VENUS-1 and VENUS-2 configurations; to build these zones fuel pins have been dismounted and the UO_2 pellets have been replaced by stainless steel rods over half the core active height.

During 1988, a full three-dimensional power map has been determined experimentally by axial scanning of numerous fuel rods taken from a quarter of the VENUS-3 core; owing to the different loadings of the branches (normal UO_2 fuel or half height UO_2 fuel + half height stainless steel) the azimuthal variations are markedly enhanced and the distorsion of the axial power distribution propagates up to the central zone of the core and in the adjacent branches. The absolute scale of the power map has also been defined using the same techniques as in VENUS-1 and -2.

Fast flux propagation outside the core has been measured in a selected number of positions; several sensors were used for this purpose ($^{58}Ni(n,p)$, $^{115}In(n,n')$, $^{27}Al(n,\alpha)$ and fission chambers). All these results have been scaled on the absolute core power.

All the measurements have been analysed and transmitted at ORNL, where the calculations are performed, starting from the experimental power map.

Preliminary comparisons of the computed results with the experimental values indicate a good agreement in most of the investigated locations, the C/E values obtained for locations around the 45° azimuth are less good, but still satisfactory ($1.0 < C/E < 1.15$).

VIP, a Reactor Physics Programme for Pu Recycle in LWRs

The operating data obtained from the power reactors, the continuing increase of the fuel performance, the new fuel characteristics, the decision recently taken by major Utilities in various countries to recycle in their power plants the plutonium obtained from reprocessing and the necessity to demonstrate to the Licensing Authority that the differences in the neutronic, safety and thermal-hydraulic features are properly taken into account show that there are still subjects of concern which require new experimental investigation for code validation. This is specially the case of MOX fuel in which the fuel composition markedly differs from the ones generally experimented up to now.

Moreover, an insufficient validation might induce a dramatic increase of the uncertainty factor with a possible reduction of the reactor power and it appears that most of the organizations concerned with MOX fuel development do not dispose of enough experimental data for their own neutronic calculation tool improvement and calibration.

Therefore, based on the experience accumulated during a 25-year collaboration, SCK/CEN together with BELGONUCLEAIRE have decided to implement a new experimental programme in the VENUS facility.

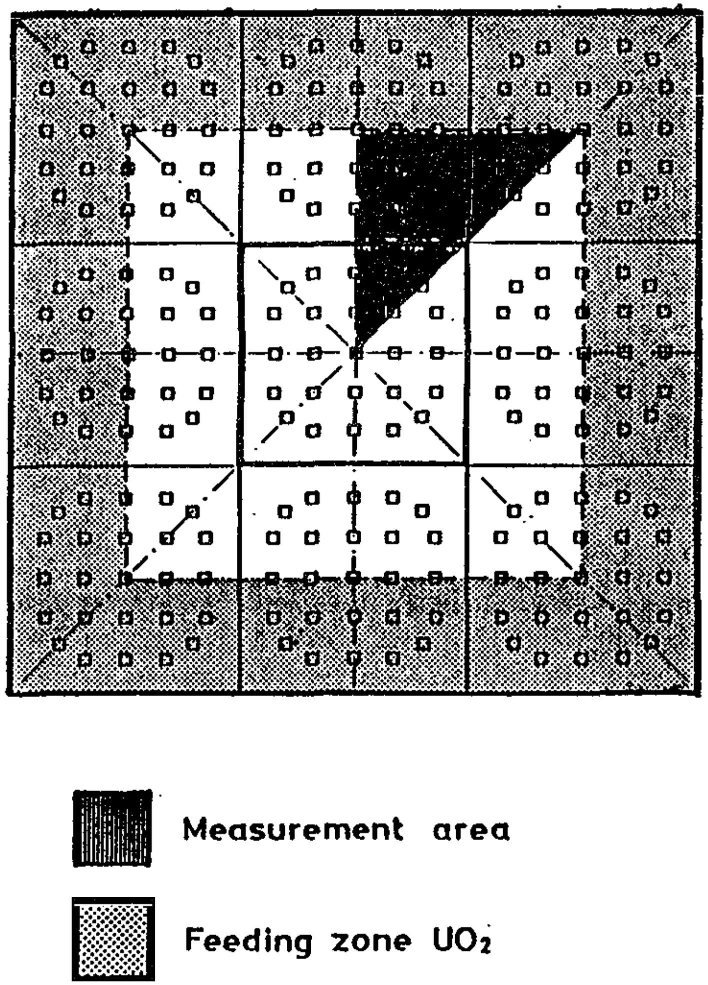
This new VENUS International Programme, called VIP, will include a complete set of experimental measurements performed with existing UO_2 and MOX fuel rods in order to provide an extensive nuclear data base for the development, the improvement and the validation of nuclear calculation methods for MOX fuels used in LWRs.

The main measurements will consist of :

- power distribution measurement on mock-ups of typical MOX assemblies,
- critical mass,
- detector response,
- reactivity effects such as moderator density effect, control rod worths, poison effect, UO_2 effect, ...

A VIP programme will be devoted to PWR (Fig. 1) and another one to BWR (Fig. 2). The orientation calculations and implementation tasks are underway. Measurements should start at beginning of 1990.

VENUS PWR MOX



1. CENTRAL UO_2 ASSEMBLY

- Critical mass
- Power distribution
- U_{235} fission detector response

2. CENTRAL MOX ASSEMBLY (3 Pu contents)

- Critical mass at nominal level
- Power distribution
- Reactivity effects (Δk)
  - critical mass at reduced level
  - Ag - In - Cd control rods
  - enriched B_4C control rods
  - boric acid
  - moderation reductions
  - Ag - In - Cd + moderation reduction
  - UO_2 central assembly
- U_{235} fission detector response

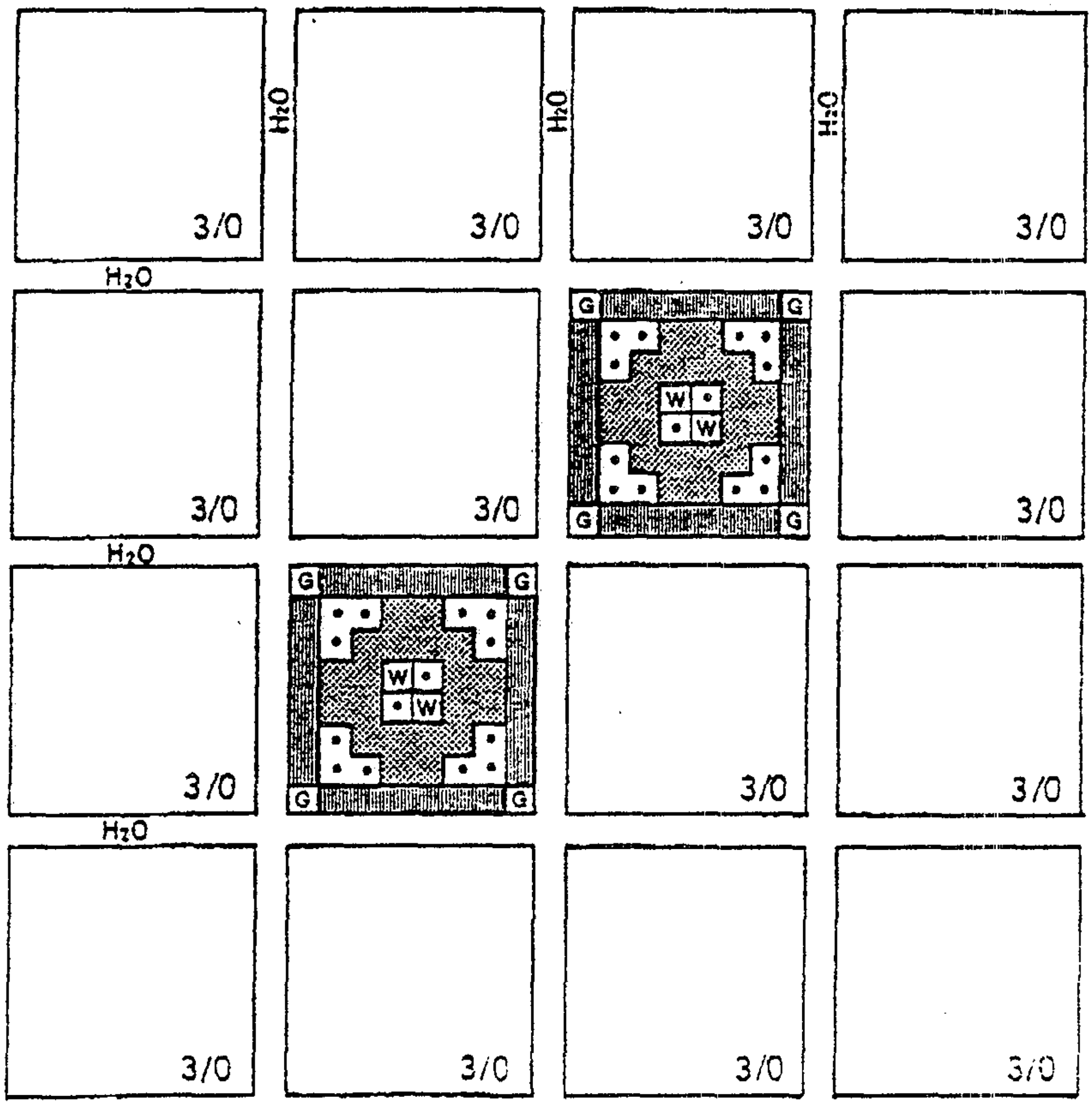
AS OPTION :

- Other MOX assembly design
- Pu vector effect on power distribution
- Am_{241} build-up effect
- β eff. measurement

Figure 1.

VENUS MOX PROGRAMME

BWR MOCK-UP



3/0 (UO_2)	Low enriched MOX
Water hole	Mid-enriched MOX
Gadolinium rod	High enriched MOX

Figure 2.

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Development and validation of Nuclear Codes

The nuclear code package used at BELGONUCLEAIRE for LWR lattices calculations is based on LWR-WIMS, an assembly spectrum code, and on MICROLUX, a three-dimensional simulator code.

The LWR-WIMS code with its 1986 updated library has been extensively validated by comparisons with experimental data from zero-power experiments carried out in several facilities, especially in the Mol VENUS critical facility and with data from mass spectrometric analyses of isotopic compositions of fuel rods irradiated in the CNA (Centre Nucléaire des Ardennes) and BR3 reactors.

In MICROLUX, a nodal formalism determines the core reactivity and the macroscopic flux distribution. With the in-currents at the node interfaces, a fine-mesh option calculates the rodwise power distribution and the power peaking factor. The MICROLUX code has been validated in experimental data such as critical boron concentration, reactivity effects, detector response map, ... obtained from measurements performed on several power reactors.

This validation was also oriented to the Pu recycle in LWRs, one of the main fuel activities at BELGONUCLEAIRE. This extensive qualification work has allowed a successful Pu recycle in the CNA reactor. In June 1989, a second batch comprising 20 % of MOX fuel assemblies was loaded in the reactor. The MOX assembly fraction in the reactor is now more than 14 %. The interpretation of physics tests at the start-up has shown a good agreement between the measured data and the theoretical predictions. No systematic deviation was observed for the MOX fuel assemblies.

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 [6]. The high accuracy of the M20 version of MERCATOR (2 D, two-group, homogenous 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 [7]. Since the beginning of 1988, MERCATOR has been adapted to read its nuclear data from the well known cell code CASMO.

ARCHIMEDE (versions A05 and A06) is a fast core simulator developed by TRACTEBEL for on-line surveillance of the core safety margins, to give expert advice to the operator about the best way to get proper core conditions. When load variations have to be planned, ARCHIMEDE predictor predetermines the best operating way, with respect to axial unbalance, effluents amounts, fuel local power variations and MTC limitations.

The simulator works from neutronic data, given by LWR-WIMS code (for A05) and CASMO code (for A06) for each fuel composition (i.e. groups of assemblies presenting same physical characteristics and history), and results of MERCATOR two-dimensional calculations for a sequence of burn up states of the core and of control rods configurations.

Average on the vertical axis give a one-dimensional two group diffusion calculation model. A synthesis is then realized to get flux and power profiles used for a composition-wise calculation of feed-backs in each axial region of the core.

ARCHIMEDE code calculates reactivity balance and flux profile, as well as derivatives and weights used for guessing the ideal operating conditions.

The fast core simulator has been installed on a digital microVAX II in Tihange-1 in December 1986. The experience shows benefits from axial unbalance control, as well as from return to critical conditions prediction after SCRAM.

ARCHIMEDE is also used in TRACTEBEL for core design studies, as soon as axial effects have to be considered. Coupling with COBRA code allows for DNB calculations.

A special module has been included to help the test engineer during startup physical tests at the beginning of each fuel cycle. This module is now used in Tihange-1, -2, -3 power plants and has also been tested in Doel-1.

A06 version will be available in the next few months. It is designed to handle CASMO neutronic tables and MERCATOR's (version MCA) results.
A30 version has been planned to include the MERCATOR 3-D nodal diffusion calculation module.
ARCHIMEDE has been presented during NEA specialists meeting in CADARACHE, FRANCE in June 1988 [8] and during the 7th power plant dynamics, control and testing symposium, Knowville, Tennessee, in May 1989 [9].

A group at University of Brussels (ULB) is developing HEXNODYN a 3D multigroup transport (or diffusion) kinetics code in hexagonal geometry, for fast reactor transient analysis. This code, made in the frame of a contract with the JRC Ispra, is a part of the european effort towards standardized software for reactor safety calculations.
HEXNODYN is a blending of the dynamics module of CASSANDRE (generalized quasistatic method) developed jointly by ULB and SCK/CEN (Mol) and the nodal transport (or diffusion) code HEXNOD developed by M. WAGNER at KWU. The diffusion version has been completed and benchmarked [10] on the wellknown NEACRP KfK rod ejection superprompt critical transient proposed by L. VATH. Results agree fairly well with those obtained with CASSANDRE; they will be presented at the PHYSOR 90 conference in Marseille (April 1990). The transport is still under development and should be completed in the coming months.
The use of response matrix in transport theory has been investigated in the frame of the use of HEXNOD.
New semi-analytical results have been obtained for two-dimensional one velocity problems in square and hexagonal cells. The transport equation is solved using a Galerkin technique with a set of basis functions which are products of harmonic polynomials and exponentials and which satisfy symmetry properties of the cell. Response matrices are evaluated using symbolic manipulation techniques.
Numerical results of model problems show a good agreement with reference solutions in those cases where the basis includes exponential functions [11].

FAST REACTORS

The analysis of the SPX1 startup experiments have been carried on at BELGONUCLEAIRE. The neutron cross-sections of the absorber rods have been adjusted to take into account the transport, fine mesh and group-collapsing effects, as well as the detailed inner structure of the rod, in a full core, coarse mesh, few-groups diffusion calculation. The use of these adjusted cross-sections results in a considerable improvement of the ratio calculation/experiment on the absorber reactivity worth, from typically 1.20 to 1.00. The prediction of the reaction rates distributions is also improved, and the remaining errors are consistent with those previously observed in the critical assemblies BIZET and RACINE.

Additional work has been carried out jointly with SCK/CEN, CEA/Cadarache and ENEA on the analysis of the gamma heat deposition experiments performed in the MASURCA facility. The results concerning BALZAC-DE2, which contained a simulated absorber rod at its centre, have been reported in [12]. The C/E on gamma doses in iron are generally less than unity in PuO_2-UO_2 mixed oxide environment suggesting considerable underestimate in the photon sources originating from plutonium. The recommendations of recent evaluations are however to decrease the fission gamma sources presently implemented in the libraries, thus leading to even greater disagreement with experiment.

As member of the INB group, BELGONUCLEAIRE took part to the conceptual design studies of the European Fast Reactor (EFR), in co-operation with the European Design and Construction companies (INB/INTERATOM, NNC, NOVATOME, ANSALDO). Comparative core physics studies were carried out on the reference conventional core and on axially heterogeneous variants. The results demonstrate that both designs are roughly equivalent with respect to most neutron physics parameters, with some advantages for the axial heterogeneous core (reactivity loss due to burnup and dpa). The decisive potential advantage of the heterogeneous version lies with reduction of clad corrosion and remains to be demonstrated in further fuel irradiation experiments at high burnup and dose.

Source Term Measurements for LMFBR Safety Evaluation

A series of seven so-called MOL 7C experiments were conducted in the Belgian BR2 reactor with a view to investigating the consequences of a severe fuel pin damage in a Liquid Metal Fast Breeder Reactor subassembly. One of the consequences could be the fission product release to the atmosphere and the corresponding hazard evaluations depend on transfer coefficients used to describe the transfer of fission products from fuel to sodium, as source term.

At present, hazard evaluations are using fractional in-vessel transfer coefficients as recommended by the Nuclear Energy Agency (13).

For some fission products, like iodine which has a great chemical affinity for sodium and which is very important for the population hazard, it is interesting to know which fraction remains fixed in the sodium bath.

For other fission products, the parametric range of the recommended transfer coefficients is quite large.
It could thus be advantageous to replace the presently recommended values by more reliable ones, based on experimental data.

Under-water high resolution gamma spectrometry was implemented during the last two MOL 7C experiments carried out in BR2.
As shown in Fig. 3, the detector device is positioned in the reactor pool, close to the reactor cover, "looking" to the sodium expansion tank of the loop.

As for the MOL 7C experiments themselves, this programme is initiated by KfK Karlsruhe (D), executed by SCK/CEN Mol (B) and supported by the CEC, JRC Ispra (I).

Due to the experimental conditions the experimental data are concerned with the transfer fuel-to-sodium, fission products collected in the gas plenum being excluded.

A first estimation of some of these transfer coefficients is obtained by comparison between experimental data and the fission product inventory as predicted by ORIGEN-2 calculations. In a first step, all the fuel inventory is assumed as melt, in order to get final results, the actual amount of destroyed fuel has to be obtained by the planned post-irradiation examination of the experienced bundles.

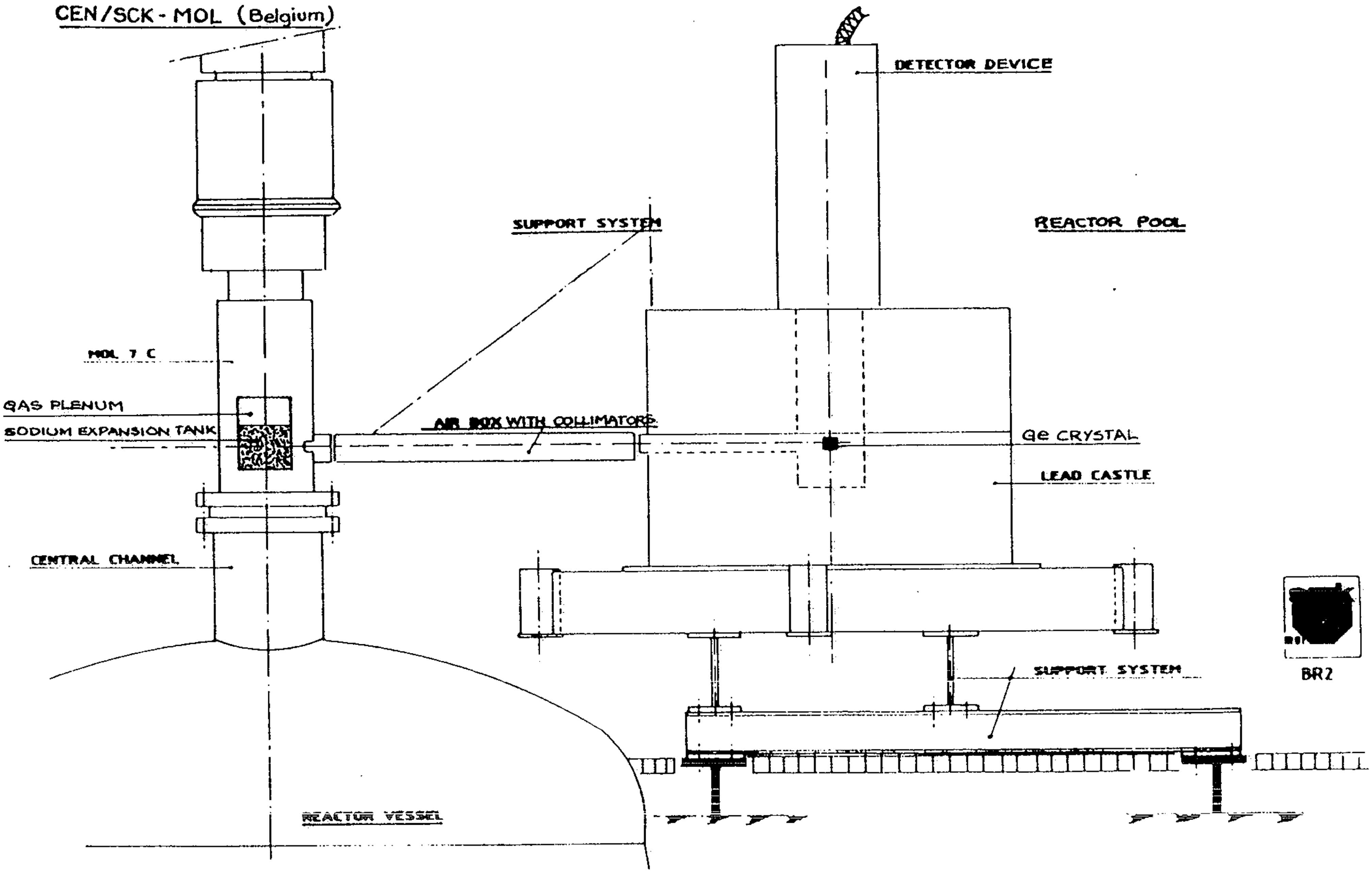

FIG. 3 : MOL 7C EXPERIMENT IN REACTOR BR2
Lay-out of the measurement device for the Source Term Measurement

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REACTOR PHYSICS ACTIVITIES IN CANADA 1988/1989

Reactor Physics Branch
Atomic Energy of Canada Limited

Compiled by R.T. Jones

This report summarizes activities carried out by various organizations during the period 1988 October to 1989 September.

1. UTILITIES

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

1.1 Ontario Hydro - Toronto Canada

1.1.1 SMOKIN Development and Validation

SMOKIN is a three dimensional space-time kinetics neutronics code based on a modal expansion method. Ontario Hydro has modified the calculation of the modal coupling coefficients in the program to provide a cross-coupling coefficient for every pair of modes (referred to as the "full matrix" version). The original program calculated and used only the self-coupling coefficients and the cross-coupling with the fundamental mode.

Several other enhancements of the code have been made to model the reactor regulating system in CANDU reactors and make improvements in other areas. A testing and validation program is underway on the revised program.

1.1.2 Benchmark Problem for CANDU In-Core Fuel Management

A benchmark problem for in-core fuel management in CANDU reactors has been developed as part of a Coordinated Research Program under the sponsorship of the IAEA. Ontario Hydro is applying its fuel management programs to the benchmark problem in order to provide results for comparison with codes available in other countries.

2. UNIVERSITY ACTIVITY

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

2.1 Royal Military College (RMC), Kingston, Ontario

RMC owns and operates a 20 kW SLOWPOKE research reactor. Research topics that are being pursued include: reactivity calculations for the SLOWPOKE reactor [1], core life extension using thorium fuels in SLOWPOKE heating reactors, optimization of fuel management in a marine PWR, and optimization studies of CANDU reactors. The latter include fuel bundle designs and fuel management for a possible thorium fuelled 600 MW CANDU reactor.

2.2 McMaster University, Hamilton, Ontario

McMaster University owns and operates a 10 MW MTR type research reactor. Other research topics covered are:

2.2.1 Nonlinear Fission Reactor Dynamics

Several nonlinear time-dependent formulations for a fission reactor cell have been investigated by analytical and numerical means [2,3]. It has been found that non-regular trajectories, more complex than limit-cycles, are commonly encountered.

2.2.2 Fission-Decay Nuclear Batteries

The combination of fission-breeding in Cf-251 and spontaneous fission of Cf-252 have been shown to provide for an intrinsically stable nuclear energy release. It appears that unattended operation for periods of 10 to 100 years is feasible for this type of fission-and-decay nuclear battery [4].

2.3 Ecole Polytechnique de Montreal, Quebec

R&D activities in the field of reactor physics have been carried out by the Groupe d'Analyse Nucleaire (GAN) at the Institut de genie energetique, in the areas of neutron transport theory and reactor calculations using diffusion theory.

2.3.1 Transport Theory

Transport calculation activities were concentrated on the following three subjects.

1. Three-dimensional transport calculations:

   The development of EXCELL [5] was pursued. EXCELL is the three-dimensional collision probability module of the transport code DRAGON [6]. Precision problems were investigated in the numerical quadrature for the transmission and collision probabilities, including an improved treatment for tracks intersecting corners and the use of integration lines that preserve the symmetry of the probabilities. New collision probability renormalization techniques were implemented in EXCELL. Analytic reductions obtained for the probabilities associated with homogeneous tubes or hexahedrea were also implemented [7].

2. Two-dimensional transport calculations:

   A two-dimensional version of the three-dimensional collision probability transport code EXCELL was written. This version uses numerical quadrature and collision probability renormalization techniques similar to those of the original EXCELL. The ADI technique that has been successfully applied for solving the three-dimensional Interface Currents flux equations has also been extended to the two dimensional version of

EXCELL.

3. Self-shielding Calculations:

A new resonance self-shielding calculation module was implemented into DRAGON, using an equivalence method based on an extension of the Bondarenko method for interactions between heavy nuclei [8].

2.3.2 Diffusion Theory

Diffusion theory activities were concentrated in the area of Generalized Perturbation Theory (GPT) applications to CANDU fuel management and reactor design. The code OPTEX-4 was developed to simultaneously optimize in 3-D the fuel management and control absorber distribution at equilibrium refuelling. The gradient of the characteristic functionals are obtained using two independent approaches, requiring the solution of fixed source eigenvalue problems (direct for the explicit approach, adjoint for the implicit approach) [9]. These solutions, as well as the solution of the diffusion problem, are obtained in 3-D by calling the diffusion module TRIVAC-2. A new multigroup version of TRIVAC is under development, incorporating vectorization.

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 Whiteshell Nuclear Research Establishment, Pinawa, Manitoba

The Systems Analysis Branch at WNRE includes two 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 Research Physics Group and the Small Reactor Concepts Group. The current scope of reactor Physics work for each group is described below.

3.1.1 Reactor Physics Group

The Reactor Physics Group provides radiation shielding and nuclear criticality safety assessment services, and reactor operations physicist support to internal AECL development projects as well as to external clients. The majority of the group's activities during the past year involved development support for the following projects: MAPLE Research Reactor, SLOWPOKE Heating Reactor, used fuel handling and dry canister storage, Waste Management Used Fuel Disposal Concept, and physicist support for SDR commissioning. Also included is analytical procedure development and upgrading.

Radiation shielding assessment support continued for the 10 MWt SES-10 Heating Reactor, the 10 MWt MAPLE-X10 Research Reactor, and several different versions of the MAPLE-10 Generic Reactor intended for specific external applications. Reactor analysis capability was enhanced with special attention to the creation, transport, and shielding of all mobile activation products in the light water coolant (^{13}N, ^{16}N, ^{17}N, ^{18}F, ^{19}O, and ^{41}Ar) in both of these reactors, and in the heavy water reflector (^{3}H, ^{16}N, ^{17}N, ^{17}F, ^{19}O, and ^{41}Ar) in the MAPLE reactors. Extra effort was also directed at studying the neutronic coupling between the core and in-pool spent fuel racks, and at assessing the ability of standard S_n libraries to properly transport neutrons through the heavy water reflector. Finally, development was started on a simple method of estimating the time behaviour of the detector signal from photoneutrons, in reactors that contain heavy water or beryllium, that does not require experimental input. The method is being tested on SDR for use in MAPLE design work.

Support also continued for AECL's spent fuel dry canister storage program with shielding assessments of the fuel baskets, shielded work station (including windows), transfer flask, and concrete storage canister. In addition to standard bulk shield calculations, detailed scattering and streaming analyses were conducted for the flask hoist mechanism and door assembly, flask/canister connection system, and the canister-contents verification tube. This work demonstrated the need within AECL of a code for handling these calculations that is more accurate than hand methods but cheaper and faster than Monte Carlo.

Extensive comparisons were completed, as part of the concept assessment of the Waste Management Used Fuel Disposal Vault Concept, between the used fuel characterization codes LATREP/CANIGEN, ORIGEN, ORIGEN2, ORIGEN-S, and WIMS. ORIGEN-S using multicycled working libraries has been chosen as the reference method. Planned experimental analysis of used fuel is now required to further refine these fuel characterization methods for use on CANDU fuel.

Criticality safety assessments were performed for the canister storage of NPD experimental fuel, the used fuel container for the Disposal Vault Concept, concrete canister storage of consolidated PWR fuel for an external client, and re-licensing of Whiteshell's hot-cell facility. KENO-5a used with the SCALIAS 3.1 System was approved at Whiteshell for criticality safety analysis of LEU fuel ($\leq$5 wt%)

The group continued to provide reactor physics support to the SDR commissioning program including experimental test approval and evaluation, in-pool gamma ray flux measurements, power reactivity coefficient measurements.

The final area of interest is the upgrading of the computer software inventory. The SCALIAS 3.1 Code System, implemented last year on the VAX/VMS computer cluster, was returned to RSIC and is now being distributed as CCC-47-5B/SCALIAS 3.1. Problems encountered when the VAX Math Library was upgraded from V4.7 to V5.1 (different results from KENO-5a) are still under investigation. The group has acquired and implemented the latest version of MCNP (3B). The repeating/lattice geometry capability will be valuable in upcoming benchmarks with WIMS/3DDT.

3.1.2 Small Reactor Concepts Group

Work on the Nuclear Battery [10,11], a very small, heat-pipe-cooled reactor power supply for the generation of electricity in remote locations, was suspended as a result of changed priorities within AECL. Instead, reactor physics effort has been transferred to examine the feasibility of an advanced MAPLE research reactor as a possible future replacement for the aging NRU research reactor at the Chalk River Nuclear Laboratories (CRNL).

The main reactor physics calculations performed for the Nuclear Battery during the past year addressed its safety response to large, fast, hypothetical reactivity transients [3]. In the Nuclear Battery, four control rods are used to compensate for 52 mk of reactivity due to temperature, xenon and burnup effects. The rapid removal of a single control rod could add a maximum of about +19 mk, bringing the reactor to a super-prompt-critical state. The only mechanism available in the Nuclear Battery to safely terminate such a rapid excursion is prompt fuel temperature feedback, mainly through Doppler broadening of capture resonances in ^{238}U. Using a point kinetics model it was determined that, for rapid reactivity additions of up to +22 mk, there would be no unacceptable damage of the TRISO coated fuel particles and, thus, the Nuclear Battery reactor would be safe following the hypothetical ejection of a single control rod.

Because the calculation of fuel temperature reactivity feedback is very important to the safety of the Nuclear Battery following rapid reactivity additions, the accuracy of these calculations was evaluated. Comparisons were made between calculations and fuel temperature feedback measurements in the SPERT-UO_2 fuel, pulsed-reactor experiments. The analyses indicate that the calculated values of fuel temperature feedback were about $7 \pm 3\%$ (about 0.60 mk) lower than the SPERT experiments. Therefore, the calculated fuel temperatures required to terminate rapid reactivity excursions in the Nuclear Battery will be reliable for safety analyses.

Several physics-related studies were performed in support of various external MAPLE research reactor projects:

1. Assistance was provided to ININ attached staff from Mexico evaluating the use of either two MAPLE-10 reactors or one MAPLE-20 reactor.

2. The fluxes in the various irradiation sites were evaluated for the proposed MAPLE-MNR at McMaster University.

3. A study was completed of the effects of placing a UO_2-fuelled 37-element CANDU bundle in the LH1 site of the KMRR reactor in Korea.

4. A physics scoping study is underway to determine the operating characteristics of a MAPLE reactor operating at 12 or 20 MW to meet the requirements of the Egyptian Atomic Energy Authority (AEA) for a research reactor to be built at Inchas.

The coolant temperature reactivity coefficient was evaluated for the MAPLE-X10 core to be built at CRNL and a comprehensive reactivity coefficient analysis was completed for the MAPLE Generic 10-MW research reactor [13]. The early stages of operation of the MAPLE-X10 core were investigated with and without

Mo targets to determine its capability to be used as a standby producer of ^{99}Mo. It was found that the use of 18- element fuel was necessary to prevent excessive consumption of driver fuel.

MAPLE reactor statics calculations are based on the WIMS-3DDT-FULMGR code system [14]. The FULMGR code was improved to include region fluxes with the flux output printout. The integration of FULMGR with the SPORTS-M thermal-hydraulics code was achieved through the new TANKLINK module. This module can be used to feed back the parameters of greatest effect on reactivity (fuel temperature and void) to determine core reactivity (from 3DDT) as a function of power or coolant flow. This feedback uses a multidimensional interpolation algorithm that was recently incorporated into MAPDDT and uses the multi-dimensional parametrization made possible by the cross-section utilities, FRAND16P and RCONVASCIIP. A mesh-conversion algorithm was written in the TANKLINK caller, CDETA, to transform from a general mesh used in 3DDT to a series of up to two hydraulic core modules per channel, each with a uniform mesh interval for SPORTS-M.

The two-dimensional reactor kinetics code TANK [15] was enhanced and tested. The TANK code has an improved method for considering reactivity feedback effects of fuel temperature, coolant void/density and poison build-up. A new thermalhydraulic section models the forced circulation of the coolant through all the fuel channels, and recalculates the coolant inlet plenum temperature. TANK was used to analyze twelve postulated accident scenarios for the MAPLE-X10 PSAR (Preliminary Safety Analysis Report). TANK simulations revealed that, in the original design, the reactor could become supercritical if the control rods continue to withdraw after two of the three shutdown rods are deployed. In a benchmark study [16], TANK was used to simulate selected SPERT-1B(24/32) transients from low power with high subcooling. The agreement with the experimental data is good.

3.2 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.

3.2.1 CANDU Neutronics and Passive Safety

The general objective is to enhance the passive safety of CANDU by modification of the lattice parameters. In particular:

(a) reduction of coolant void reactivity, and,

(b) increase in negative feedback reactivity due to loss of cooling.

3.2.2 Code Development

(a) Coupled Neutronic/Thermalhydraulic multidimensional transient codes are now used routinely in analysis.

(b) Routine transient analysis is now carried out with space-dependant kinetic codes that explicitly describe the variation over the core of the kinetic properties of the lattice.

(c) AI is being applied to arrive at optimal refuelling strategies.

3.2.3 Reactor Regulation and Safety Systems

High speed computation of 3-D diffusion theory problems is being studied for on-line power mapping and shut-down.

3.2.4 Advanced Plutonium Burner

Enhancement of the conversion ratio, by making use of subcritical multiplication, has been studied for CANDU (natural uranium lattices). Results show that plutonium contribution to fuel utilization can be raised to several times that achieved in a normal CANDU.

3.2.5

Use of unrefabricated spent LWR fuel in CANDU has been studied. Additional burnup of 35000 MWd/te(V) is indicated. Such an LWR/CANDU tandem fuel cycle raises fuel utilization by 70%.

3.3 Chalk River Nuclear Laboratories, Chalk River, Ontario

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

(a) reactor assessment and design,

(b) code development and nuclear data,

(c) research reactor support, and

(d) experimental activity.

3.3.1 Reactor Assessment and Design

3.3.1.1 Advanced CANDU Reactors

Work continued on assessing fuel management options for slightly enriched uranium (SEU) in CANDU. For the optimum enrichment of 1.2% SEU, giving a burnup of about 21 MWd/kg, an axial shuffling scheme was devised which gave excellent power profiles, from the viewpoint of fuel performance (declining power history with burnup), and low peak bundle powers. When combined with the advanced CANFLEX fuel bundle, element ratings were substantially reduced from current levels with natural uranium fuel. Axial shuffling involves removing the entire fuel string from the channel during refuelling, rearranging bundle pairs, and reinserting the bundle pairs back into the channel in different positions, along with fresh fuel. Time-average, and time-dependent refuelling simulations were performed for axial shuffling, and two other fuel management options [17,18].

Physics and safety studies are continuing in support of the advanced CANFLEX fuel bundle. This bundle has greater subdivision (43-elements), and 2 pin-sizes, which reduces peak ratings by 20% compared to the 37-element bundle. Optimization of the number and location of bundle appendages improves the critical heat flux. The lower ratings, as well as optimization of the pellet design, facilitates the achievement of extended burnup with SEU, and power maneuvering. The lower ratings offer significant safety benefits [19].

3.3.1.2 Small Reactors

The SLOWPOKE Demonstration Reactor (SDR) is a 2-MWt prototype of the larger 10-MWt SLOWPOKE Energy System Reactor being designed at AECL. SDR located at WNRE, has operated at powers up to 1.2 MW for short times under manual control. The measured negative reactivity change with increasing power is in good agreement with calculation. Calculated reactivities were obtained from four-group reactor calculations using the CITATION finite difference diffusion theory code. The four-group cross sections for each material region in the reactor simulation were obtained from multigroup WIMS pin-cell or supercell calculations. The calculated k-effective is between 1.002 and 1.003 for two critical configurations with different absorber plate/control rod positions. Experimental and calculated copper activation distributions through the core agree within about $\pm$ 2%.

Reactor physics calculations, using the WIMS and CITATION computer codes, for design studies of the 10 MW heating reactor, SES-10, are continuing. Reactor physics aspects regarding reactor safety, control and economics are under investigation.

3.3.2 Code Development and Nuclear Data

Efforts have continued to validate WIMS-AECL against experimental data using the ENDF/B-5 based library.

Nuclide compositions as a function of burnup for natural uranium and plutonium/aluminum alloy fuel have been compared with WIMS-AECL results. For the natural uranium fuel there is a tendency for WIMS-AECL to predict a too high value for the Pu^{239}/U^{238} ratio, the error increasing with burnup to a value of about $+3\%$ at a burnup of about 11000 MWd/t. For the plutonium alloy fuel the ratio of Pu^{239} to total plutonium is predicted to within experimental error. The results for the Pu^{240} to total plutonium ratio show some sensitivity to the number of WIMS-AECL main transport groups.

Detailed lattice parameters and $k_{effective}$ for a lattice of Bruce fuel, measured in the ZED-2 reactor, have been compared with WIMS-AECL. The reactivity change on voiding the coolant is too high by 3.5 ± 1.1 mk, where the uncertainty comes from the experimental error. Within experimental error none of the detailed lattice parameters show any effects attributable to coolant voiding.

3.3.3 Research Reactor Support

3.3.3.1 MAPLE-X10

Work has continued on the design of this 19 site, 10 MWt reactor. Simulations have been run, from initial core loading through to equilibrium refuelling, to ensure the refuelling scheme is viable and the regulation and safety system meets reactivity specification throughout the initial operational period. Reactivity coefficients were calculated. Work was also started on the development of the commissioning program.

3.3.3.2 Core Development

The TRIAD3 code system is being developed for simulating the very heterogeneous NRU research reactor. The diffusion code uses discontinuity factors, with cell parameters derived from WIMS. The code was run in a core-following mode for a year as part of the commissioning process. Results appear to be reasonable when compared with (somewhat inaccurate) power distribution measurements [20] and the previous simulation code. Further validation is planned.

3.3.3.3 NRU Cold Source

Various conceptual designs for a cold source for the NRU research reactor were evaluated using the MCNP code.

3.3.3.4 Neutron Radiography

A large neutron radiography facility was designed and installed adjacent to the NRU reactor thermal column. The facility, now fully operational, produces a thermal flux of $4 \times 10^7 n.cm^{-2}.s^{-1}$ at the film plane, with a cadmium ratio of 1000.

3.3.3.5 LEU Conversion

A batch of 30 Low-Enriched Uranium (LEU) driver fuel rods for the NRU research reactor were manufactured and irradiated to 80% burnup. The rods, each containing pins of U-Si-Al dispersion fuel, with 20% U^{235} in uranium, behaved just as well as the regular U/Al HEU (93% U^{235}) rods.

3.3.3.6 CANDU Fast Flux Bundle

A conceptual design for a fast flux facility, for irradiating metallurgical specimens in a CANDU reactor, was evaluated. The concept uses a regular CANDU 37-element bundle with the centre element removed to provide space for the specimens. It was shown that fast fluxes up to twice the peak pressure tube value could be achieved.

3.3.4 Experimental Activity

Work related to various advanced fuel cycles for CANDU reactors has occupied the experimental section for most of the past year.

3.3.4.1 $(U^{233},Th)O_2$ Fuel Program

Seven rods of this fuel, each containing five bundles are available. The measurements being performed in the ZED-2 critical facility include:

(a) the reactivity effect of central substitutions of 1,3,5 and 7 rods with various coolants, in a natural uranium fuelled core,

(b) The reactivity effect of heating the fuel and coolant of the seven substituted rods from 20°C to 300°C, and

(c) the distribution of various reaction rates including fission of U^{233}, Pu^{239} and U^{235}, and capture of Th^{232}, Cu^{63}, Mn^{55}, Lu^{176}, In^{115} and Au^{197} in and about the bundle located at the centre of the substituted region.

The experimental measurements are about 80% complete and some have already been reported [21].

3.3.4.2 Measurements with $(Pu,U)O_2$ Fuel

Measurements similar to those described above, but using five rods of $(Pu,U)O_2$ bundles have been completed and are being analyzed. The results obtained to date [22] indicate generally satisfactory agreement with calculations performed with the cell-code WIMS-AECL and the core-code CONIFERS, except for the reactivity effect of heating the fuel and D_2O coolant to 300°C. For this the calculation underestimates the temperature at which the reactivity coefficient switches from negative to positive.

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Reactor Physics Activities in Denmark

Compiled by Erik Nonbøl
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1. Introduction

Denmark has no commercial nuclear reactors to-day and no nuclear power is foreseen for the near future.

The Energy Department at Risø National Laboratory still maintains a research and development effort in reactor physics, because this subject field is seen as fundamental to a general understanding of reactor technology, and it is also the basis for reactor physical services for the two research reactors and the fuel work at Risø National Laboratory. Furthermore, the department maintains the educational activity for universities and high schools at the Danish reactor no. 1.

2. Development of the BWR Core Simulator Programme, COSIMA

Due to our limited staff we have chosen to concentrate on developing an advanced core simulator program, COSIMA. Some of this development has been reported at the latest NEACRP-meeting, but during the past year several improvements have been introduced.

A Ph.D. work performed some years ago at Risø (1) produced a true multigroup nodal program based on the Nodal Expansion Method, NEM.

This program was modified and substituted for the TRILUX routine (POLKA) in our old nodal code and thus constitutes the neutronic module of COSIMA (2).

The NEM method is "self-contained", i.e. it does not depend on any dubious constants, but requires only regular multigroup data like diffusion coefficients, scattering matrices, fission cross sections etc. The method allows the use of cross sections modified with the socalled "discontinuity" factors, which to some extent will compensate for using homogeneous cross sections in large nodes containing strong inhomogeneities.

Previously, the cross sections used in the simulator had been calculated by the LEWARD programme (3) in the way that one set was produced assuming a control rod inserted during the whole burnup history, and another set was produced assuming no control rod inserted at any time.

The burnup distribution within the assembly is, of course, dependent on the time in which the control rod has been in, and so are the nuclear properties of the assembly. An exact simulation of the CR history is not possible unless the assembly burnup calculations are made currently for each node during the overall simulation of the core performance.

This seems presently to require quite prohibitive calculational efforts, and thus some approximative method had to be developed.

The method chosen assumes that as to the dependence of burn up, the nuclear properties of an assembly depend on:

1) The total burnup
2) The part of the burnup in which the control rod has been inserted

but not on the detailed time variation of the control rod movements.

The principles are illustrated in Fig. 1, where the actual control rod movements are shown at the bottom, and the assumed distribution of burnup with and without control rod indicated in the diagram.

It is seen that burnup-wise the control rod presence is pushed from the actual time to the beginning of life of the fuel.

For most of the nuclear parameters there is quite a dramatic variation to be seen, when the control rod history is taken into account (4). Being the most integral quantity, k_{inf} is shown in Fig. 2 for a Quad Cities assembly with 3 gadolinium rods.

The effect is most pronounced at low burnup, but is clearly seen also at higher burnups.

In the near future we plan to perform verification of COSIMA on the reactor Quad Cities.

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FIGURE 2.

KINF AS FUNCTION OF BURNUP, WITH AND WITHOUT CONTROL

PARAMETER IS THE BURNUP, WHICH HAS TAKEN PLACE WITH THE CONTROL ROD IN

FUEL ASSEMBLY: QUAD CITIES, 3 GD PINS, 50% VOID

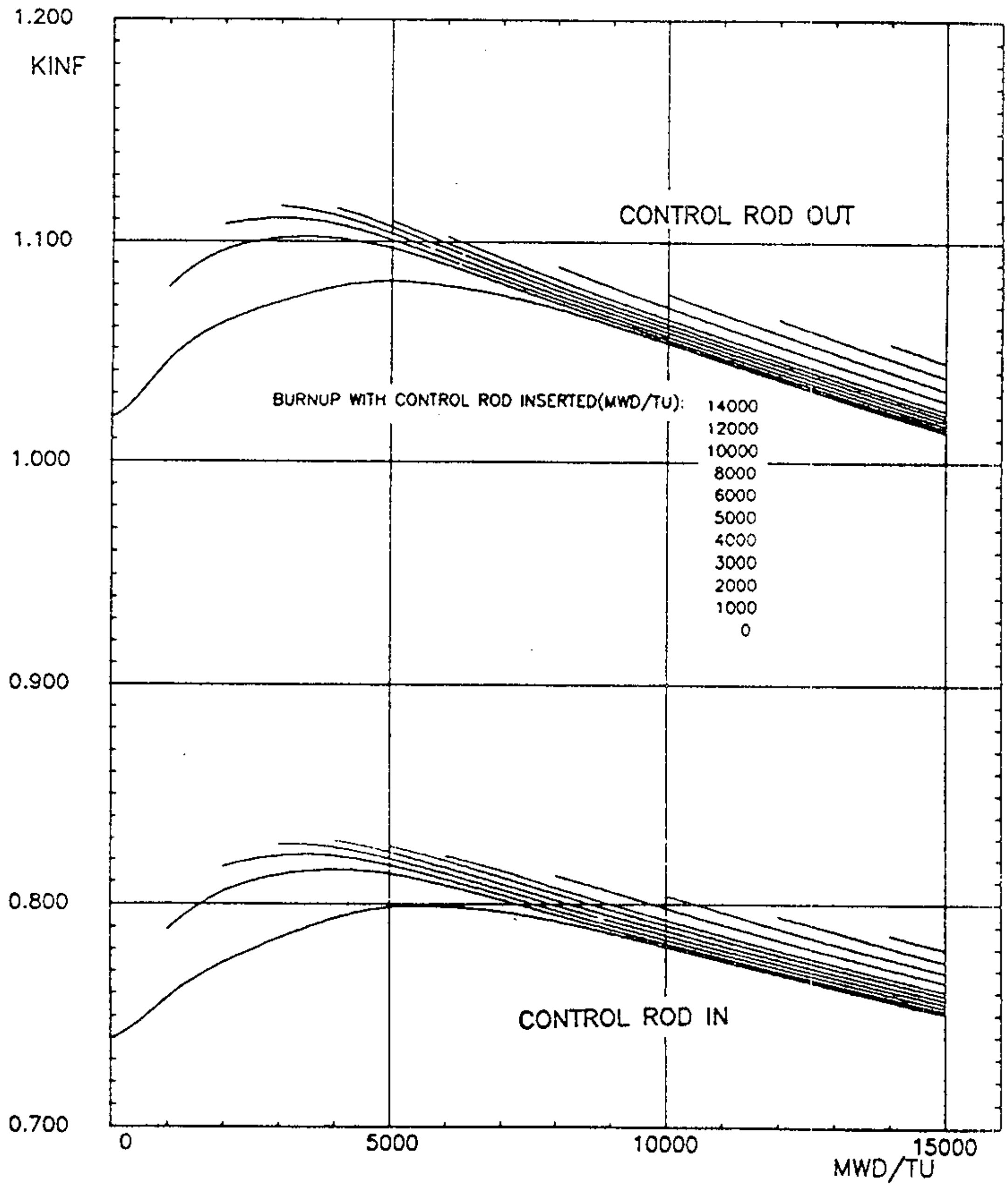

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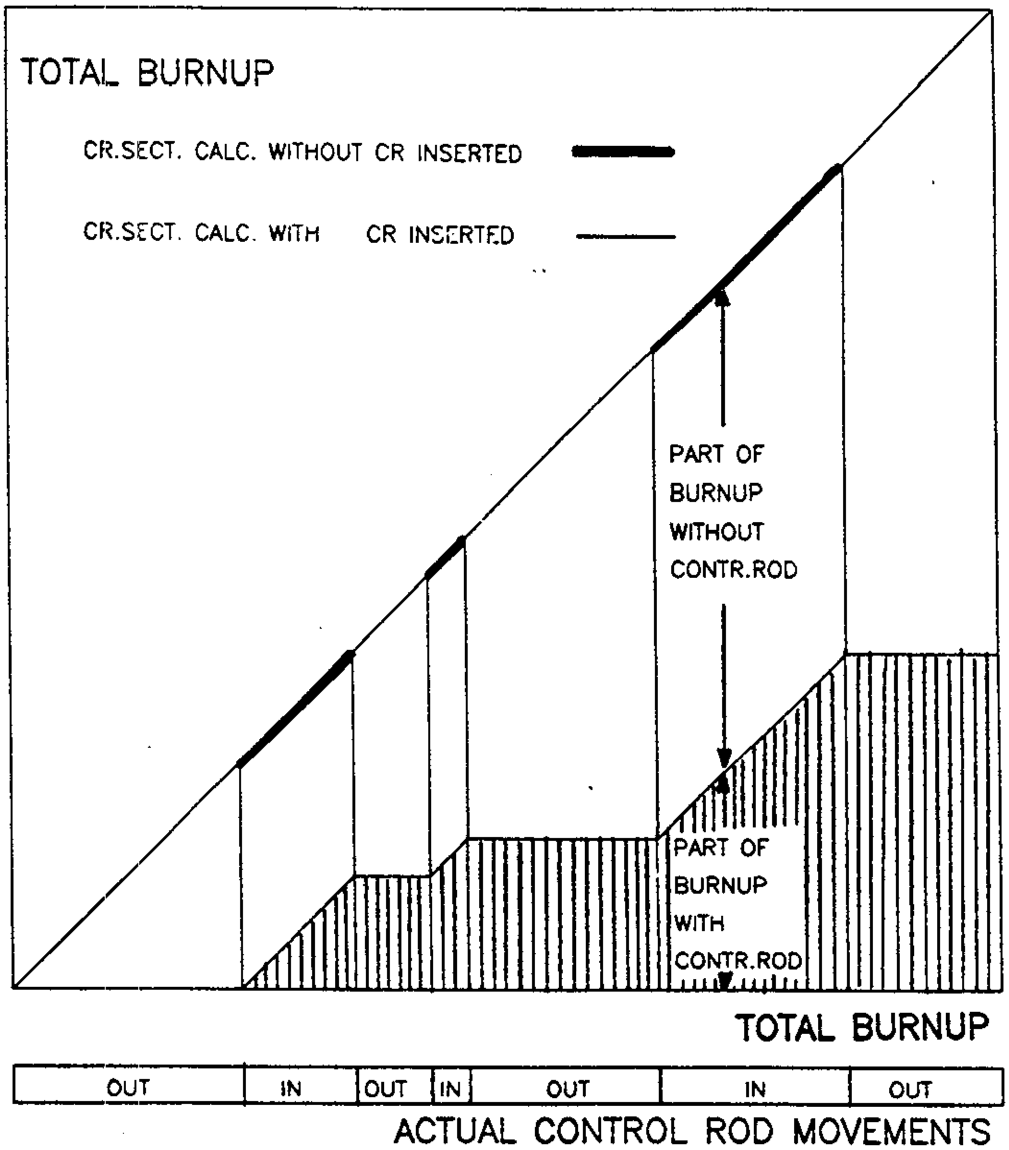
FIGURE 1.

CONTROL ROD HISTORY APPROXIMATION



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REACTOR PHYSICS IN FINLAND

STATUS REPORT TO THE NEACRP 1989

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1 NUCLEAR POWER IN FINLAND

From 1980 to 1988, the consumption of electric energy in Finland has risen from 39.9 to 58.7 TWh, or 5 % a year on an average. In 1988, there were still four nuclear reactors (two BWRs and two PWRs) which, operating at a mean capacity factor of 91 %, produced 18.4 TWh or 31 % of the total.

Although net import of electricity is already equivalent to the output of a 1000 MWe reactor, no decision on whether or not to build any new nuclear power plants is expected for a couple of years at least.

2 CELL CALCULATIONS

Some test calculations have been carried out with the fuel assembly burnup program CASMO-HEX.

The dependence of the results of the program on the data library was studied in the case of VVER-440 fuel assemblies. After some improvements CASMO-HEX can at present use three different data libraries based on UKNDL, ENDF/B-III or ENDF/B-IV (all have been delivered by Studsvik Energiteknik AB). The calculations indicate, that the choice of the data library may have a noticeable effect on the results (e.g. on the reactivity and its burnup dependence).

The capability of CASMO-HEX to predict correctly the effect of burnable absorbers has been and will be tested in connection with the Coordinated Research Project on Safe Core Management with Burnable Absorbers in VVERs arranged by IAEA. A benchmark problem has already been calculated.

3 CORE CALCULATIONS

The nodal code HEXBU-3D was used to analyse the reactor physical characteristics of the three-batch fuel cycle of the VVER-1000 reactor. The analysis consisted of a simulation of the first five cycles and the calculation of the reactivity coefficients and control rod efficiencies for the first and stationary cycles. The cross section data were evaluated with the cell burnup code CASMO-HEX. The results will later be compared with calculations performed in the Soviet Union and with measurement data from operating reactors.

A further verification of the HEXBU-3D code was carried out against criticality measurements from an experimental reactor at the Kurchatov Institute in Moscow /1/. The reactor was a full-scale mockup of a fresh core of a VVER-440 reactor that had the same initial loading as the Loviisa reactors. The experimental data included critical states with different concentrations of soluble boron and different positions of control rods. The average of the effective multiplication factors calculated with HEXBU-3D was 0.9990 for the 25 critical states measured at the reactor. The arithmetic mean of the deviations from unity was 0.24 % and the maximum deviations were about 0.5 %. These results can be considered rather good, since the moderator height varied in the experiment from 25 cm to 250 cm while the active height of the core was 244 cm.

The measured efficiency of all control rods in the experimental reactor was predicted by HEXBU-3D within 0.7 % in reactivity. This agreement is much better than observed between calculated and measured efficiencies of control rods in the Loviisa reactors. The latter values are obtained by dynamic reactivity measurements together with the rod drop technique and obviously these values cannot be accurately produced by simple calculations of static reactivity. A research project is going on to investigate the measurement using a prompt jump approximation. The idea is to calculate the prompt behaviour of the neutron flux at the points outside the reactor core where the ionization chambers for the measurements are located.

4 RELOAD DESIGN

Certain modules to be included in a BWR reload design planning code are being tested. They perform the tasks of creating an initial loading pattern (to be used as a starting-point for later improvements) and suggesting favourable bundle moves after the results of a simulation of the cycle are available. The moves are intended to improve the thermal margins of the core, i. e. to reduce the maximum heat flux and raise the dryout and shutdown margins by increasing the burnup at and around any problem positions. Not only the average burnup of the fuel bundles is taken into consideration, but also their vertical burnup distribution.

5 SHIELDING AND ACTIVATION CALCULATIONS

The activity inventories of the decommissioning waste components have been calculated and reported for all the

Finnish nuclear reactors /2, 3/. The required flux calculations were performed using the REPVICS program system, in practice ANISN with the BUGLE-80 cross section library and some auxiliary programs of our own and the activation and subsequent cooling of the components were studied with the ORIGEN-S computer program.

The REPVICS program system has been utilized also in a study concerning the pressure vessel irradiation and gamma heating in the Loviisa reactors.

For these studies some auxiliary programs of REPVICS had to be modified. In particular, applying the S_N branch of REPVICS to the Olkiluoto reactors with their square core geometry made it necessary to alter the PVIS program, which prepares a distributed source for ANISN. In the original hexagonal-core version of PVIS the range of fission spectra has been widened.

The cross section preparation program BUGLER has been modified so that the cross sections taken from the BUGLE-80 library can be multiplied by an arbitrary factor given for any energy group(s) and any nuclide(s). This has been used to correct the cross sections for resonance shielding in U-238. In addition, an option to replace the photoelectric absorption cross section for gamma rays in BUGLE-80 by energy deposition cross sections has been included. Activity cross sections can now be taken from the BUGLE-80 cross section library itself (e. g. neutron absorption or gamma energy deposition cross sections) and multiplied by suitable input factors, in addition to the old option of taking the activity cross sections from a separate file.

The calculations concerning the dose rates inside and outside final disposal canisters of spent fuel have been continued.

6 DEPENDENCE OF STEAM MOISTURE ON THE POWER DISTRIBUTION IN A BWR

Moisture in the steam of a nuclear power plant causes trouble with enhanced radioactivity, erosion processes and reduced efficiency. Measurements have shown that the moisture content of the steam in a boiling water reactor may be quite different during different cycles of operation and also change drastically during a single cycle, although the operating conditions remain largely the same. The variations were thought to have something to do with the changing power distribution in the reactor's core and, using the available experimental data, correlations were developed to calculate the moisture level for a known power distribution.

The correlations are based upon the idea that high moisture values are most likely to occur if the steam separators situated above the reactor core become overloaded in certain regions. A computer program containing these correlations seems to predict the steam moisture quite satisfactorily /4/. It was found that the power density of the core's central region must be high enough in order to keep the moisture low. Thus, the reactor should preferably be loaded in a way that makes it possible to avoid the use of too many deep control rods near the centre of the core. This may occasionally lead to a need for a few fresh fuel bundles extra.

7 ACTIVITIES RELATED TO THE CHERNOBYL ACCIDENT

Although no essentially new work has been done concerning the Chernobyl accident, a couple of papers have been published on this subject /5, 6/.

8 INSTABILITY STUDIES

In the year 1987 some power oscillations occurred during an abnormal situation in the TVO I BWR-reactor. Preliminarily, it was thought that this event was contrary to the existent predictions. The Finnish Centre for Radiation and Nuclear Safety (STUK) has studied the event and its variations with the three-dimensional BWR-dynamics code RAMONA III and with the one-dimensional code TRAB. The three-dimensional results have shown that the event can be acceptably postcalculated. Also the one-dimensional calculations have yielded correct instability behaviour when the reactor power has been raised by some 20 per cent. It has furthermore been observed that the instability behaviour is greatly affected by such an uncertainly modeled phenomenon as the heat conduction in the gas gap.

9 THREE-DIMENSIONAL DYNAMIC CODE FOR VVER REACTORS

The three-dimensional dynamic code HEXTRAN for VVER reactors is in a testing phase. It is based on the three-dimensional stationary code HEXBU-3D and the one-dimensional dynamic code TRAB. The time integration methods of TRAB are used. The radial heat transfer in a fuel rod is calculated for each fuel bundle. The thermohydraulics is calculated axially in separate hydraulic channels, which connect to one or several fuel bundles. For the calculation of the moving control rod followers in the VVER-440 type reactors a special treatment is developed.

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REACTOR PHYSICS ACTIVITIES IN FRANCE
October 1988 - October 1989

M. Darrouzet & M. Salvatores, CEA/CEN Cadarache

1 - GENERAL

In the end of 1989, France will have 55 reactors commissionned with a capacity of 57 GWe. The share of nuclear power in the total French electricity production is about 70 %.

At present, the growth is slower. During the next four years only 9.5 GWe will be added.

A renewal in nuclear plant construction is expected in the beginning of 21st century, in particular to replace any existing plants nearing end-of-life and decommissioning.

The national electric power utility has given notice of the decommissioning of the four GCR in the near future.

Concerning the pressurized water reactors which are the bulk of installated capacity, the last reactors of the 900 MWe series have been commissionned in 1988. The current series is now P'4 (1300 MWe PWR).

The connection to the grid of the first of French design (N4 serie) is expected at the earliest in 1991.

The national electric power utility, in collaboration with the constructor FRAMATOME and with the CEA is carring out the studies of a future reactor called REP 2000.

In other respects, an industrial agreement has been announced between FRAMATOME and KWU to answer the call for international bids of nuclear reactors.

SUPER PHENIX is back in operation. The reactor was restarted on 15/01/89 and went to full power on 19/06/89 and after that date it has been operated normally.

The reactor will be shut down during the fall for an already foreseen period of maintenance. At that time, a new configuration of the core to reach $\sim$ 400 FPD (from the present $\sim$ 150 FPD), will be implemented.

The reactor physics activities concern :

- the improvement and optimization of the reactors in operation : increase of fuel burn-up, recycling of uranium or plutonium (loading by quarter core and burn-up up to 42000 MWd, loading with 30 % of MOX assemblies : St Laurent B1 plant has now 32 MOX assemblies at St Laurent B2 plans 16 MOX assemblies),

- the studies of undermoderated plutonium lattices and of core loaded with 100 % MOX assemblies,

- the studies of parameters important for the safety (temperature coefficient...),

- the support to the European Fast Reactor (EFR) design.

2 - <u>CORE PHYSICS</u>

2.1 - <u>Thermal reactor experiments</u>

Experiments are performed at the Cadarache center in the frame of cooperation with the utility (EdF) and the industry (FRAMATOME).

2.1.1 - <u>Tight lattice core experiments</u>

The ERASME program, described in previous documents, has been completed, both for the core parameters (in the EOLE reactor) and the fission product experiments (in the MINERVE reactor). A short survey of recent results is given in a separate paper for this meeting [1]. A detailed analysis is underway, to provide to the designers a full set of bias factors and uncertainties to be associated to the design studies in a large range of HCLWR concepts. A specific theoretical approach has been developed to that aim [2]. The fission product experiments were of relevance to validate current FP libraries, and in particular the performance of the JEF-1 data has been assessed and indications obtained for possible data revisions in the frame of the version 2 of the JEF file (JEF-2).

2.1.2 - Pu recycling

The significant Pu recycling program underway and planned for the future years, has raised a number of specific requests in the physics area, to reduce uncertainties on the major neutronic parameters. An experimental program has been agreed among CEA, EdF and FRAMATOME, to be performed on the EOLE reactor in Cadarache, between mid-1989 and early 1992. The first phase of the program (called EPICURE), to be completed before the end of 1989, covers the following 3 steps :

a) Full UO_2 core (3.7 % enrichment) clean configuration (no water-holes),
b) configuration with water holes,
c) a "zoned" MOX simulated assembly is introduced at the center of configuration b.

This final configuration is represented in Figure 1. Successive steps in the program will cover different moderating ratios of the uranium core, full MOX cores with different moderating ratios and a simulation of a checker-board core configuration (up to 5 MOX assemblies in the U core).

For each configuration, power distributions, buckling, spectrum indexes and absorber cluster reactivities will be measured. Some temperature coefficient and β_{eff} experimental values will also be obtained.

2.1.3 - Spent fuel analysis

Spent fuel analysis has been traditionally an important part of the validation data base for the neutronic methods and data. This program has been continued during this year. The data are directly used to assess bias and uncertainties to be associated to fuel cycle code systems. Moreover, the data, together with sensitivity analysis, are used to improve the basic data, and that use is foreseen specifically for the JEF-2 file validation.

A typical example is the measurement of the γ-activity ratio of Cs-134 and Cs-137, which is used to determine the input burn-up of irradiated fuel which is processed at LA HAGUE plant.

The "on-line" analysis of this measurement needs the use of software to which basic neutronic data (such as capture cross-sections) are given. These data are physically validated, e.g. with the comparison of the Cs calculated and measured (mass spectrometry) abundances for irradiated fuel samples (e.g. from the Bugey and Fessenheim power plants), dissolved at the COMIR installation in Cadarache.

C/E values are given in Table 1. Good agreement is observed for Cs-133 and Cs-137, but a discrepancy for Cs-134. This discrepancy has to be accounted for (by adjustment coefficients) by the "on-line" analysis equipment.

2.1.4 - Data banks

In agreement with FRAMATOME, the CEA has decided to organise the large experimental data base (critical experiments, irradiated fuel analysis) in a data bank, for easy storage and retrival of data. This data bank will be compatible both with design calculation routes and with experimental analysis, more sophisticated, calculation routes. A first pilote version will be available by the end of 1989.

2.2 - Thermal Reactor Data and Method Development

2.2.1 - Multigroup structures and basic data

The use of the JEF-2 data, foreseen for the near future, will be coupled to a generalisation of the present multigroup structure, adopted together with the APOLLO cell code.

Test are being made for a convenient representation of the major physical caracteristics of, e.g., Pu isotope cross-sections.

In the frame of the JEF project, the thermal and epithermal data of U-235 have been revised, accounting for the new experimental data on η and σ_f.

2.2.2 - The new cell code APOLLO-2

Work on the code is in progress. The burn-up modules have been tested. Self-shielding algorithms are being validated [3].

2.2.3 - <u>Other code and method developments</u>

a) The 3D on-line codes, based on fast 3D core power distribution calculations, are still being developed both at CEA (the RITME code previously reported) and at EdF (the CAROLINE code [4]). Both codes are verified on operational reactor data with very encouraging results.

b) A specific work related to the treatment of the double heterogeneity for fuel dissolver criticality problems, has been performed. Significant results have been obtained to understand discrepancies previously found in the NEACRP benchmark exercise, and are reported at this meeting [5].

c) Sensitivity methods have been extended to the analysis of thermal reactors [2], and are applied to a systematic re-analysis of spent fuel analysis results.

d) Full 3D analysis of rod ejection configurations are performed at EdF with the new developed COCCINELLE code [6].

e) Validation of the methods for the fine power distribution reconstruction starting from the internal instrumentation in a PWR, with mixed UO_2/MOX fuel (EdF and CEA collaborative program).

f) Uncertainties reduction studies for the fluence assessment on the PWR tanks with better prediction of incident neutron spectra (EdF and CEA collaborative program).

2.2.3 - <u>Core design</u>

The main works concern :

- improvement of the 30 % MOX fuel management,
- studies of 100 % MOX fuel management,
- analysis of the technical feasibility of tight lattice core with or without spectrum shift with or without axial or radial blanckets [7].

These studies require the development of reference calculation methods, parametric work accident analysis (control rod ejection accident, steamline break analysis...), which are underway.

2.3 - Fast Reactor Experiments

2.3.1 - Power Reactor Experiments

PHENIX

In the frame of the European cooperation on Fast Reactors, and in support of EFR axial heterogeneous core option, it is presently discussed the possibility to introduce a significant number (19) of axially heterogeneous fuel subassemblies in the core, in particular to validate fuel performances. A final decision will be taken before the end of 1989.

SUPER PHENIX

After the reactor start-up, a physics experiment program has been performed, mainly devoted to control rod worth verification and to a new handling error simulation.

These experiments, as the initial series of start-up experiments, are being analysed by the European "ad hoc" Task Force. Summaries are expected in 1990. However, already now, new design uncertainties have been derived for the major design parameters. As an example, for control rod worth there is a generally agreed figure ($\pm$ 12 % at 2σ) by all the partners, which is presently used for EFR design studies.

Recent analysis indicates that little C/E spatial variation is observed on control rods worths (for three inner ring rods C/E = 0.92, and for three outer ring rods C/E = 0.95, with CEA present methods and data).

2.3.2 - Integral experiments in MASURCA

The CONRAD-AX (first phase of the European program CONRAD) has started in June 1989. The first configuration is a clean (no rod positions) axial heterogeneous core, with a fertile slice of 20 cm at the core mid plane. In table 2, the C/E value on the critical mass is given. Axial and radial flux distributions have been performed. In the next phases, the axial heterogeneous configurations will have fertile slices of different thickness, position and composition (e.g. simulation of Pu build-up).

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An extensive program of β_{eff} measurements is foreseen after the CONRAD-AX phase. A tentative program, for the proposed NEACRP experimental benchmark, is given in a separate paper at this meeting [8], for comments of the potential participants. These experiments should take place in the second half of 1991.

2.4 - Data and Method Development

Most of the activity has been devoted to the development of a modified version of the CARNAVAL-IV formulaire, to account for the results at the SUPER PHENIX start-up. This new formulaire (called CARNAVAL-IV-89) is now available and its performances are summarized in table 3 for some important parameters. Work has been devoted to the support of the EFR project (in particular in view of the introduction in PHENIX of axial heterogeneous subassemblies), to advanced concepts (see paper at this meeting [9]), to Na void reactivity coefficient uncertainties assessment and reduction [10], and to the analysis of the reactivity swing during the cycle [11].

The development of a new neutronics code system has been decided in the frame of the European collaboration on fast reactors. This code system (called ERANOS), is based on the CEA present code system CCRR. This major development should be completed by 1993 and mostly performed in Cadarache. The new common cell code ECCO is being validated extensively. An operational version will be available by the end of 1989. Improved self-shielding algorithms and collision probabilities routines have been developed [12, 13].

In the frame of the JEF-2 file assessment, work on the new Pu-239 evaluation has been completed. The work on Pu-241 resonance data (in cooperation with ORNL), is also almost completed.

Decay data and fission yield related data validation has been continued in the frame of the JEF project.

3 - EXPERIMENTAL TECHNIQUES

The β_{eff} measurement techniques have been continuously tested at MASURCA. The proposal for a NEACRP experimental benchmark in this domain, is indicated in a paper presented at this meeting.

Active nuclear measurements techniques (neutron interrogation or modulated gamma transmission), together with classical passive methods, have been developed for fuel cycle related purposes (e.g. process control, criticality measurements etc). A paper presented at this meeting, gives details on the techniques [14].

4 - SHIELDING STUDIES

Shielding experiments are performed at WINFRITH on the NESTOR reactor, in the frame of the European cooperation on fast reactors (the JANUS program). These common experiments are also analyzed at Cadarache, with the BISTRO S_N-1D code and the PROPANE shielding formulaire. Past results have indicated the need for improved experiments to better understand flux attenuation in B_4C layers. These experiments are presently been performed in the frame of the JANUS program.

5 - FUSION NEUTRONICS STUDIES

Fusion neutronics studies are presently being devoted to data uncertainties impact evaluation on the shielding design of the NET machine. This activity is done in support to the NET team design work. The EFF/JEF data are being used, together with the sensitivity capabilities of the CEA neutronic code system CCRR.

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TABLE 1

Samples defined by the number of irradiation cycles		1	1.5	2	3	4
Initial enrichment %		2.10 %	3.10 %	3.10 %	3.10 %	3.14 %
Burn-up (GWD/b)		21.5	20.9	25.2	39.2	49.8
$(\frac{C-E}{E})\%$	Cs 133/238 U	- 1.6	+ 0.3	- 1.1	- 2.6	+ 3.2
	Cs 134/238 U	- 12.8	- 12.9	- 13.7	- 2.7	- 1.5
	Cs 137/238 U	- 4.9	- 0.3	- 2.1	- 2.1	- 1.0
	Cs 134/Cs 137	- 8.3±1.5	-12.6±1.5	- 11.8±1.5	- 0.7±1.5	- 0.5±1.5

TABLE 2

	CONRAD-AX	RACINE-1E[a]
Calculated reactivity (Diffusion XYZ 25 groups)	- 1221 [b]	-
Mesh correction	+ 256	-
Transport correction	+ 647	(+ 600)
Rodlet end caps correction	+ 71	-
E-C	+ 247	+ 457

(a) Radial heterogeneous configuration with the same core enrichment and fuel/structure/Na volume ratios, and approximately the same core volume and fissile/fertile ratio.

(b) Values in 10^{-5} $\Delta K/K$.

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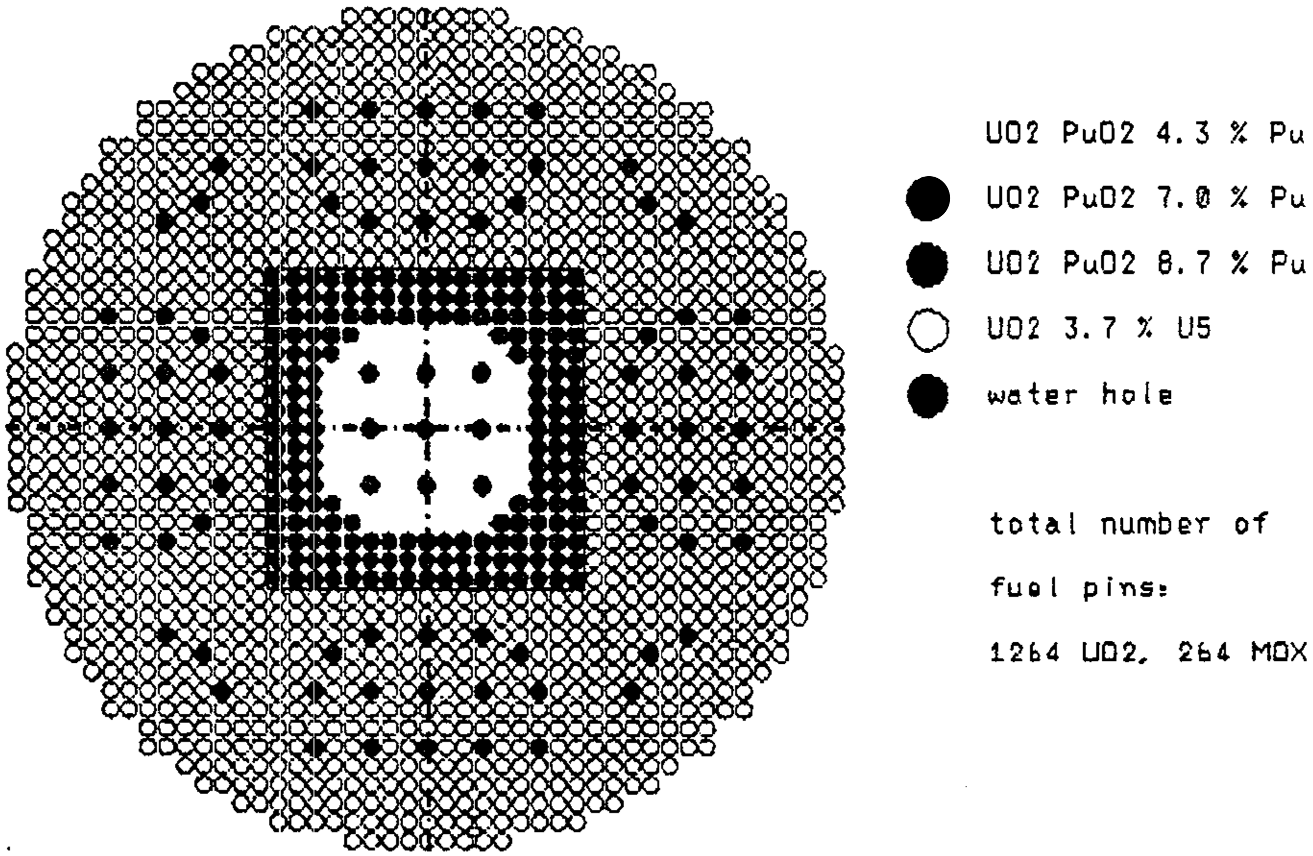
TABLE 3

	CARNAVAL IV Standard	CARNAVAL IV 1989
REACTIVITY OF CRITICAL CONFIGURATION		
[E-C p.c.m.]	- 137	- 2
- First critical Core	+ 41	+ 111
- Full power Core		
ROD WORTH (C/E)		
Complementary shutdown system	1.10	1.03
Main control rod system	1.08	1.01

INFLUENCE OF CARNAVAL IV CROSS-SECTION MODIFICATIONS

EPICURE: 3 ZONES MOX ASSEMBLY IN UO2 CORE



F.R. Germany

REACTOR PHYSICS ACTIVITIES IN THE FEDERAL REPUBLIC OF GERMANY

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GENERAL:

With the 1314 MWe pressurized water reactor GKN-II at Neckarwestheim in Baden-Württemberg the last, for the time being, SIEMENS/KWU Konvoi reactor was commercialized on January 3, 1989. At present about 41 % of the electricity production in Germany is covered by nuclear energy.
The fast reactor SNR-300 is still waiting for the operation-licence. Financial support by the government, the manufacturers and the user-companies is guaranteed until the end of 1991.
On February 16, 1989, the contracts for the Fast Reactor Development in an European frame (EFR) were signed. These contracts cover

(a) European R&D activities in the field of fast reactors at the Nuclear Research Centers at Karlsruhe (KfK), CEA and UKAEA as well as research activities at INTERATOM; Netherland's ECN-Petten cooperates as a Research Participant Institute.
(b) The cooperation of the manufacturers, i.e. INB (Interatom), Belgonucléaire Neratom, Ansaldo, Novatome and NNC.
(c) The SERENA-FASTEC contract regulates the know-how exchange between the partners as well as the financial share in placing licences to third parties.

In the High Temperatur Reactor with Thorium fuel, THTR-300, 31 from 2500 bolts are broken; these bolts fix the heat-isolation of the hot-gas channels. Each isolation plate is fixed by 5 bolts, no more than 1 bolt per plate is broken. Providently the utility HKG has asked the government for a permanent shut-down of the plant. The licence for operating THTR ends mid of 1992; a new licencing

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procedure is necessary for the time after 1992. It is expected that new conditions will be placed on the safety measures for THTR-300; the plant is still shut-down.
A common development of a modular HTR with 200 MW_{th} has been agreed by the USSR-State Committee for the Use of Nuclear Energy and the companies ABB/Siemens (ABB = Asea Brown Boveri). On June 13, 1989, a "General Contract" was signed between the USSR State Committee, V/O Technab Export and the German company HTR-GmbH for a detailed planning of this HTR-reactor.
The largest change in Germany's nuclear power program is coming from the decision of the government in June 1989 to abandon reprocessing activities on a large scale in Germany: The Wackersdorf plant in Bavaria was given-up. According to an agreement between Cogéma and Veba, reprocessing of burnt nuclear fuel can be performed in La Hagne. This decision, which was prepared first by industry, has also implications on the activities of the research center KfK.
On June 6, 1989 the German Federal Government has approved a basic ruling for the development and use of nuclear energy within a European Strategy, which in consequence leads to a cooperation in Europe with a division of workload.

I. REACTOR PHYSICS ACTIVITIES AT THE NUCLEAR RESEARCH CENTER KARLSRUHE

1. Evaluation an processing of nuclear data

Th evaluation for ^{238}U for the JEF-2 file was finished, that for ^{241}Pu is near completion /1/. The processing code NJOY-87.0 was implemented on the KfK-computer and was improved /2/.

2. Reactor physics analysis for fast reactors

(a) The fast reactor cell code KAPER4 /3/

KAPER4 is a broad-group cell code which solves the integral transport equation with isotropic scattering in a collision probability formalism. The buckling method is used to separate the macroscopic reactor flux, and the microscopic cell flux. The code performs zero-dimensional eigenvalue calculations with given group- and direction dependent bucklings. This calculation also provides fluxes or adjoint fluxes, and reaction rates in the cell. In addition, KAPER4 produces cell-averaged cross sections, as well as anisotropic diffusion coefficients obtained by the method of Benoist. These cross sections can be stored in the SIGMN block in the KAPROS system, and used in whole-core calculations.

The code has its own cross section preparation routine. It reads the data from a broad-group library, which also contains tabulated self-shielding factors ("f-factors"). The most frequently used cross section set is KFKINR, which has 26 energy groups. Self-shielding is then calculated by means of a procedure originally due to Wintzer /4/, which can be considered as an extension of Wigner's equivalence principle.

The flux calculation uses exclusively macroscopic cross sections obtained by the KAPER cross section preparation routine. Input of macroscopic cross sections is not possible.

(b) In voided channels of a fast reactor lattice the determination of the streaming-reactivity is of great importance. During an accident the fuel pins breakup, fuel and cladding material are filling the channels: these parts of the core are homogenized in the model. The associated reactivity change is positive because of reduced leakage. The method, realized in the code ARIADNE /5/, is applied in accordance with the dynamic behaviour of clad breach and gives an upper limit of the streaming reactivity effect.

(c) Experimental and theoretical efforts to better understand severe accidents in fast reactors are mainly concentrated on core melt-down situations together with an improvement of the theoretical description of fluid-dynamics using three velocity fields (AFDM). Measurements are being performed to investigate the penetration of a core melt into the core-structure. The melt is simulated by a thermite reaction on Fe- and Al-oxide. Up to now deep penetrations were obtained in these out-of-pile measurements, in contrast to the results from in-pile investigations. It could be shown that this different behaviour is due to the presence of solid particles in the out-of-pile produced melt.

3. Advanced Pressurized Water Reactors (APWR)

The investigations for this reactor are done in a tripartite cooperation between KfK/Siemens-KWU/PSI-Würenlingen.

The theoretical tools were improved by (a) implementing a detailed determination of the neutron distribution in resolved resonances, which is based on the code RESAB /7/. Furtheron the three-dimensional nodal diffusion and transport theory code in hexagonal geometry HEXNOD /8/ is used in design calculations very effectively. To keep $k_{eff} = 1$, control of the reactor is automatically taken into account in the code system. To describe fuel elements with heterogeneities as boron rods or water holes, a supercell code can be applied /9/.

The calculational procedures at KfK were critically reviewed with respect to the interpretation of critical experiments performed in PROTEUS at PSI /10/.
Design calculations are concentrating on relative wide lattices with low pin power ratings to achieve a high burnup of approximately 70 GWd/t. A preliminary analysis shows that this goal can be reached with a throughout negative void reactivity.
Experiments are being performed at PROTEUS and are covered in the Swiss Progress Report.
Experiments and their theoretical modelling are performed at KfK on the reflooding behaviour in tight lattice rod bundles after a loss of coolant accident (LOCA for the blanket region of an APWR with p/d = 1.06) /11/. A comparison of PWR and the tight lattice APWR-geometry shows that during forced feed flooding the reflood heat transfer is quite different.
A blind code prediction of an experiment with a wider lattice (p/d = 1.24) under forced feed flooding shows substantial deviations from the test results. The experiments hopefully will help to resolve the deficiencies in the theoretical models.

4. Some Aspects of Fuel Cycle- and Safety Analyses for PWRs

The formation of volatile fission products from spontaneous fission during dissolution of burnt fuel elements has been critically discussed /12/. Work is in progress on the investigation of high-burnup (60 GWd/t) LWR uranium-, MOX-, and recycled uranium fuel.
In thermal hydraulics, good experience was gained by vectorizing the computer code COMMIX-2. The CPU-time could be reduced by a factor of about 20, so that now routine calculations and parametric studies can be performed, which previously hardly were possible /13/.

5. Fusion Neutronics

Two design concepts are studied at KfK: a helium cooled ceramic blanket and a blanket with Pb/Li eutectic as a breeder material and coolant. Both concepts should be DEMO-relevant, and design studies are being conducted in parallel also for NET.
A general anisotropic neutron transport code system has been developed at KfK during the past years. While the one-dimensional transport modul ANTRA-1 is frequently used at KfK for analyzing integral benchmark experiments, the 2-dimensional code ANTRA-2 is presently being checked. The ANTRA-development is based on a rigorous treatment of anisotropic neutron scattering.

For a spherical shell of beryllium with 5 cm thickness the measured neutron leakage spectrum has been compared with calculations using EFF-1 cross section data. The results are shown in Fig. 1, in general giving good agreement except for energies between 0.35 and 0.7 MeV, where the experimental uncertainty is rather high. ANTRA-1 results agree well with Monte Carlo calculations (MCNP) and conventional S_N/P_N approximations, using the same basic nuclear data in a completely different way.

At KfK the results of the KANT-experiment on spherical beryllium shells also showed good agreement with calculational results, using the MCNP code to describe the leaking neutrons from the surface /15/.

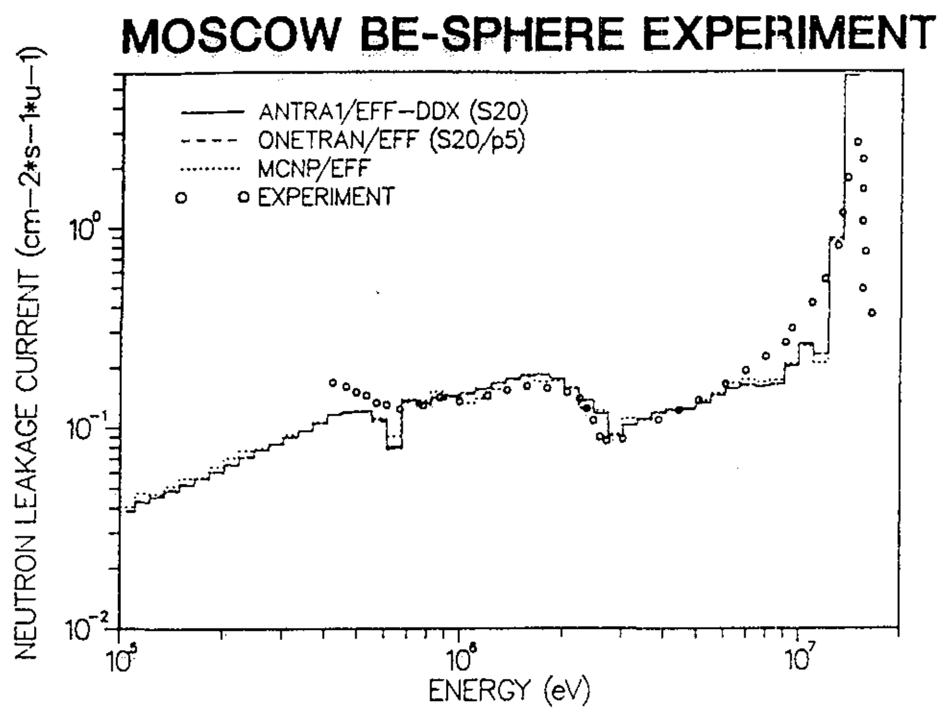

Fig. 1 Energy spectrum of neutrons leaking a 5 cm thick spherical Be-shell (central 14 MeV neutron source)

As a major conclusion, the good agreement of calculational with experimental results shows especially the reliability of the EFF neutron data for beryllium.

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II. REACTOR PHYSICS ACTIVITIES AT SIEMENS / UB KWU

1. Status of Design and Experience with Recycling for SIEMENS/KWU Type LWRs.

The main interest is presently devoted to Pu recycling in PWRs and has reached an advanced status in design, licensing and insertion of MOX fuel assemblies on a commercial basis. Licensing of Pu recycling in BWRs is under preparation. The technical feasibility of recycling of enriched reprocessed uranium (ERU) has been clearly proven by the insertion of lead test assemblies.

Optimized designs of MOX fuel assemblies for PWRs presently use up to 3 types of MOX fuel rods of different Pu-content on the basis of natural or tails uranium as carrier material. Power flattening is improved by water rods. High flexibility exists in the number of MOX fuel assemblies to be loaded in PWR cores; licenses allow in some cases the insertion of up to 50% MOX fuel assemblies and include modern low leakage loading patterns for the cores. The design of all MOX cores offers attractive possibilities to future Pu utilization.

The MOX fuel assembly designs submitted for licensing of Pu recycling in BWRs comply with U fuel assemblies in use at the given reactors. An internal water channel instead of some water rods has especially advantageous features to reduce undermoderation in MOX fuel assemblies and leads to only 4 different MOX fuel rods; the poisoning is done by insertion of a number of U-Gd rods in the MOX fuel assemblies. The quantity of MOX fuel assemblies to be inserted in the BWR cores follows the concept for self-generation Pu recycling.

For U recycling, reprocessed uranium was reenriched to a somewhat higher U235 content to compensate the parasitic neutron absorption of U236. No adverse effects have been found during reactor insertion of such ERU fuel assemblies. The U236 content of RU from high burnup fuel gives a technical limit to U recycling as long as no very highly separative enrichment (laser) is available.

The SIEMENS/KWU experience with MOX and ERU fuel assemblies is based on the insertion of up to now 244 MOX fuel assemblies in 8 PWRs (48 MOX-FAs loaded during shut downs in 1989), 168 MOX-FAs in BWRs and PHWR (until 1974), and 9 ERU-FAs in PWR power plants of Obrigheim (KWO) and Neckarwestheim (GKN-1).

Primary operating results include information on cycle length, power distribution, reactivity coefficients and control rod worth of cores with recycle fuel assemblies. Postirradiation examinations on standard and test fuel rods (partly ramped in the HFR Petten) have been performed with regard to dimensional behaviour and fission gas release under stationary and transient conditions and to mass spectrometric isotopic composition measurements. Reprocessing of burnt MOX fuel was successfully tested, too, as well as reactor irradiation of 2nd generation MOX fuel.

All results show comparable operational behaviour to U fuel assemblies up to the maximum achieved average fuel rod burnup of 47 MWd/kg (HM); i.e. thermal recycling is feasible on an industrial scale. Normal levels of reliability and safety can be maintained in the reactor operation at increased recycle fuel fractions of higher fissile content and at higher burnup.

2. Space-Time Kinetics Code HEXTIME for the Analysis of High Converter Reactor Transients.

With the advent of the high converter reactor (HCR) new calculation requirements have to be met due to its particular design features. The geometric layout of the fuel assemblies with their tighter hexagonal fuel rod lattice constitutes the basic difference in comparison with the PWR. It necessitates suitable adaptations of nodal methods available for Cartesian geometry. Besides, the resulting neutron spectrum is shifted towards higher energies and into the epithermal energy range. In order to correctly evaluate this effect, the neutronic equations must be treated allowing for more than two energy groups. With regard to thermal hydraulics, and apart from the need for hexagonal application of the corresponding equations, the narrow fuel rod pitch introduces the need for more detailed safety-related analysis. Using an open-channel thermal-hydraulic model yields an improved description of core behaviour at far-off-nominal conditions with highly non-uniform power density distributions and/or low mass flow through the core, e.g. under steam line break conditions.

To meet these requirements the coupled neutron kinetics/thermal hydraulics code system HEXTIME is now under development.

The neutronic part of HEXTIME is based on the hexagonal code HEXNOD / 1 / allowing for up to four neutron energy groups. However, a time-dependent treatment of the nodal equations had to be developed. The resulting steady-state and transient nodal methods applied in hexagonal geometry are similar to the corresponding Cartesian methods / 2 /. One of the basic differences is that for a hexagonal prism (hexagonal node) three channel equations for the transverse averaged flux in the three directions perpendicular to the hexagon faces are needed (instead of only two in Cartesian geometry) . These hexagonal channel equations are solved semi-analytically. It is assumed that the transverse averaged group source is well approximated by an expansion in terms of orthogonal polynomials. The solution of the inhomogeneous channel equation then has two parts, namely its homogeneous and the particular inhomogeneous solution (a polynomial ansatz) .

The thermal-hydraulic part of HEXTIME makes use of the open-channel methods as implemented in COBRA III-C/P / 3, 4 / . The HEXTIME subchannels are coupled by two types of cross-flow mixing, one being the net diversion cross-flow resulting from flow redistribution, the other being turbulent mixing without net mass exchange. The diversion cross-flow is considered to be small compared with the axial mass flow rate. In the case of boiling, a one-dimensional homogeneous two-phase slip flow is assumed to exist in each affected subchannel.

The coupled neutron kinetics / thermal hydraulics program HEXTIME is designed for the calculation of reactor core behaviour under conditions ranging from close-to-nominal to far-off-nominal. It allows 3-dimensional steady-state and transient full core as well as subchannel analyses to be performed. HEXTIME may be applied to high converter reactors as well as to any light water reactor with hexagonal fuel rod lattice.

The capability to calculate cross-flow effects is essential to achieve the high degree of spatial resolution and accuracy aimed at in the coupled calculation. HEXTIME thus can serve for high accuracy HCR core design applications, achieving economically efficient reload strategies. Its capability to evaluate thermal safety margins based on local hot channel fuel pin values is especially valuable in safety analyses for events resulting in highly non-uniform power density distributions and/or low mass flow through the core. HEXTIME is currently being applied to the investigation of such events (e.g. steam line break) in an HCR core design and safety study.

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III. REACTOR PHYSICS ACTIVITIES AT THE UNIVERSITY OF STUTTGART (IKE) 1988/89

1. Treatment of Cavities in Three-Dimensional Diffusion Calculations Based on the Finite Element Method

W. Bernnat, F.A.R. Schmidt (IKE)
W. Gießer (HRB)

The treatment of cavities in diffusion calculations necessitates procedures to account for the divergence free flight of neutrons in regions with very low material density.

Assuming the diffusion equation describes the neutron streaming outside the cavity (core and reflector) sufficient accurate, a reliable solution of the complete problem can be achieved by a coupled transport/diffusion solution for the cavity and the region outside the cavity respectively. Adequate boundary conditions for the coupling of regions in which the transport- and the diffusion equations are solved simultaneously can be formulated. These boundary conditions require steadiness of the normal component of the current density along the boundary between diffusion- and transport region. The angular dependency of neutrons going into the cavity can be described by the linear anisotropic term in Φ (r, Ω, E) which can be derived from the solution of the diffusion equation. The solution of the transport equation can be performed by any adequate method (eg. S_N or Monte Carlo). For the case of a cylindrical symmetric problem (e.g. for the mentioned HTR-cavity) good experiences with this coupled method were made using pre-calculated response matrices (calculated by Monte Carlo) for the transport solution and a finite difference method for the diffusion equation.

To improve the treatment of neutron streaming in the cavity the coupled method used for 2D(r,z) problems was extended to general 3D-problems.

DIFGEN /1/ is a modular program system in which the source distribution along the cavity boundary can be calculated iteratively. The solution of the transport equation can be performed by means of a pre-calculated response matrix which describes the neutron transport from every surface element of the cavity to each other.

Results achieved with the 2D-option of DIFGEN (r,z-geometry) agreed excellently with the corresponding method based on the finite difference algorithm.

The 3D-case can be solved by means of the same coupling method as for the 2D case. Using isoparametric finite Elements of order two (or one) and marking the element surfaces which are a member of the cavity surface boundary, a cavity region can be separated from the diffusion region and treated by a transport method (e.g. by a response matrix). For the case of empty cavities (no inserted rods or other materials) the corresponding response matrix can be calculated easily by means of a semi-analytical formalism. If control rods are inserted into the cavity different procedures were developed to account for the interaction of neutrons with the control rods.

The modules for the 3D-treatment of cavities with inserted rods are completed (VAX and CRAY-2 version) and will be validated presently. Production calculations are planned for September 1989. In the first instance, main applications will be the determination of reactivity changes due to the stepwise movement of rods in the cavity (single rods or groups) of a HTR which represent complex 3D-problems and require reliable results. Due to the flexible geometry of both the finite element program and the procedure for calculating response matrices for the cavity regions there may be also other interesting applications.

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2. Evaluation of Neutron Cross Sections for Liquid Hydrogen and Deuterium for the Design of Cold Neutron Sources

W. Bernnat, D. Emendörfer, J. Keinert, M. Mattes

Cross-sections for slow neutron scattering from H_2 and D_2 have been calculated taking into account the liquid state. The ability of the model is demonstrated by comparison with experimental results for differential and total cross-sections. For applications the scattering law S (α, β) has been prepared in the ENDF-6-format for different temperatures as a basis for the generation of scattering matrices. From calculations of neutron spectra with different scattering models it turns out that the gain factor for cold neutrons is sensitive to the liquid state. Blow 1 meV the frequently used Young-Koppel model for a molecular H_2-gas overestimates the gain factor by about 50 %. For D_2 the Young-Koppel theory underestimates measured leakage spectra because of the inadequate transport cross-sections.

3. Calculation of PWR Cycles on the Base of JEF-1 and ENDF/B-IV/V Data

D. Lutz, W. Bernnat

The validation of JEF-1 data in respect to light water moderated systems was performed mainly by corresponding benchmark calculations at room temperature. An important part of its applications, however, is in the range of operating conditions in power reactors. Only a few clean experiments are available for these conditions. Therefore, calculations were performed for the cycles 18 and 19 of the Obrigheim power station (KWO), a 350 MW_{el} pressurized water reactor in the Federal Republic of Germany. The JEF-1 results (boron concentration, activation ratio) were compared with results based on ENDF/B data and measurements.

Cycle Calculations

The cycles 18 and 19 of the KWO PWR were calculated with both data based on JEF-1 and ENDF/B. The burn-up distributions at the beginning of these cycles were taken from the plant data base.
The resulting boron curves using ENDF/B data deviate from the measured values up to 30 ppm, the JEF-1 based results, however, are up to 50 ppm higher than the corresponding ENDF/B-values.
For both uranium and MOX-assemblies the infinite multiplication constants derived from JEF-1 lie 0,2 to 0,4 % higher than those derived from ENDF/B. These differences could not yet be explained sufficiently, but it seems that (in opposite to room temperature) at higher temperatures the multiplication constants calculated with JEF-data are systematically higher than corresponding ENDF/B results.

Power Distribution

To get a realistic comparison of the measured and calculated power distribution a comparison of the activation rates measured by the aeroball system and direct calculated activation rates was performed for an octant of the KWO core at a time point in the 18th cycle. The burn-up distribution as well as fuel temperatures, moderator densities, boron concentration and xenon equilibrium densities were taken from the cycle calculation. Then for this octant a detailled Monte Carlo calculation was performed for 45 energy groups representing uranium fuel assemblies homogeneous (but in a 3x3 subdivision) and MOX-assemblies pinwise.
The normalized results of measured and calculated activation rates are in good agreement between measurement and calculation was found. The 1-sigma deviation was estimated to 2 %.

4. Contribution to the HCLWR Benchmark at BOL and EOL Conditions

D. Lutz, W. Bernnat

For the HCLWR lattice benchmark at BOL and EOL conditions comparisons of results achieved with our standard calculation method (CGM/RSYST /1/, /2/) were made with reference solutions calculated by JAERI /3/ with the continuous Monte Carlo code VIM.

The calculation methods for these second phase of the HCLWR benchmark were improved against the first phase reported last year. The main improvements were

- regarding a fission matrix for evaluating of the fission spectrum,
- special weighting of structural materials and oxygen for the void case,
- explicite resonance treatment of the regarded fission products Tc-99, Rh-103, Xe-131, Cs-133 and Sn-149 and of all actinides by solving the slowing down equation for 8500 groups.

The results (k_{inf}, conversion ratio, reaction rates) agreed well with the continuous Monte Carlo solution and with the average of all contributions to the benchmark. From a detailled analysis of the thermal, epithermal and fast cross-sections we concluded that the largest uncertainties come from the treatment of the unresolved resonance range.

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IV. REACTOR PHYSICS ACTIVITIES AT THE TECHNICAL UNIVERSITY OF BRAUNSCHWEIG (IFRR)

J.K. Axmann

Four different activities have been carried out during the last year in the field of neutron physics and radiation shielding at the Institute for Spaceflight and Reactortechnology (IfRR), Technical University of Braunschweig.

1) Neutron physical code development

The 35 energy-group-, multizone-cellcode SPEKTRA has been improved and performed as a source of group constants for the nodal code HEXNOD, which was implemented in the DITUBS program system of Braunschweig.

2) Installation of the nuclear data processing code NJOY

At the beginning of 1988 the IfRR received the nuclear data processing code NJOY from the NEA data bank in the version 6/83. The installation of the program system has been carried out with very little experience evaluating nuclear data, processed with NJOY. That caused the uncertainty whether real program errors or "home-made" mistakes were handled. Another problem arose from the use of an IBM/AMDAHL computer system with less accuracy in the word length representation. But with the help of the computer center at TUBS and in cooperation with the Paul-Scherrer-Institute in Würenlingen (CH) the NJOY-system has been implemented in an adequate IBM version and several errors have been detected /1/.

3) Verifications of the data base and benchmark participations

As in the former years main activities have been evolved in the validation of the cross section libraries, basing on ENDF-B/IV and V and on JEF 1.1 data. Therefore, the IfRR is participating at the PROTEUS experiments in Würenlingen /2/ as an associated partner of Siemens/KWU-Erlangen in coopertation with KfK-Karlsruhe and PSI-Würenlingen. Additional participation in the follow-on--NEA-CRP-Burnup benchmark /3/ has extended the knowledge of the behavior of tight PWR-lattices not only in the case of coolant loss but also during burnup /4/.

4) Development of a Monte Carlo Code for shielding calculations

A two dimensional Monte Carlo code MONTUBS in cylindrical geometry has been developed to calculate the spectrum and the released energy in the surrounding of high active waste in a salt layer. The code validation has been carried out with the help of several benchmarks and messured data /5/.

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REACTOR PHYSICS ACTIVITIES IN ITALY

Compiled by R. Martinelli, ENEA

1. INTRODUCTION

The second year of the moratorium imposed on the construction of new nuclear power stations -and tacitly extended to the operation of the two available LWR plants- is almost over. It has been characterized by record growths in electricity demand and import, but difficulties still exist in finalizing the new National Energy Plan, which should be discussed within the end of 1989.

Meanwhile, the Parliament took a final decision in February on the conversion of Montalto (designed to be a twin 1000 MWe BWR units station) to a conventional multifuel plant where all the options proposed by ENEL -natural gas, oil and coal- will be kept. Besides, in July ENEA and ENEL agreed to terminate the contract with Ansaldo-ABB on Cirene (a 40 MWe HWR/BLW experimental plant) without requiring the completion of the pre-operational tests; which means that the original plans to utilize Cirene "for training and simulation" have been abandoned.

As a consequence of the re-orientation of the nuclear programme and of the significant re-organization taking place at both ENEA and ENEL, future reactor physics activities are expected to follow new directions. In particular, they will be connected with advanced/innovative reactor concepts embodying intrinsic and/or passive safety features. An "ad-hoc" committee appointed by the Minister of Industry is setting guidelines and targets for a medium-term research and industrial development programme in this area, to be carried out in the framework of an international co-operation.

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2. INTERPRETATION OF FAST CRITICAL EXPERIMENTS

ENEA's participation in these activities, carried out under the AGT-3 European agreement, have been progressively reduced. Nevertheless, significant contributions have been given to the analytical effort which is underway in some areas such as:

- gamma heating and neutron energy deposition. The analysis of the BALZAC-DE experiments in MASURCA has been completed /1/. The results have been used to validate the pertinent formulaire developed with ENEA's experimental and analytical collaboration, and to support the design of future experiments in the CONRAD campaign;
- reactivity scale for SPX-1. The study, based on the intercomparison of calculations made by different AGT-3 partners /2/, has been completed with the assessment of the best estimator for SPX-1's effective beta;
- SPX-1 startup data analysis. Emphasis has been placed on the determination of experimental control rod worths. To this purpose, a final set of Modified Source Multiplication (MSM) factors has been produced, resulting from an intercomparison between ENEA/CEA and KFK values /3/. The analysis has also been supported by the development and validation of diffusion calculation methods that take into account heterogeneity and transport effects /4/. (ENEA-VEL, Casaccia)

3. DEVELOPMENTS AND APPLICATIONS OF CALCULATIONAL METHODS

3.1 LWR Dynamics

3.1.1 Implementation of NADYP-LWR. A first LWR version of the two-dimension core dynamics code NADYP /5/ -originally developed to analyze transient and accidental sequences in fast reactors- has been implemented at ENEA Casaccia. The availability of such a code was necessary for the analysis of the Chernobyl event (driven by a typical spatial effect like the abnormal increase of the void coefficient in the inner core region) and will be of further use in severe accident studies for the LWRs of the next generation.

Both neutronics and thermohydraulics modules and models had to be modified in a significant way. From the neutronics point of view, it was shown that the metastatic method, as well as all quasistatic methods, cannot correctly represent the very fast time evolution of neutron fluxes and precursor densities in water reactors. An improved metastatic method was then developed which allows the asymptotically dominant part of the neutron flux to include a larger number of terms, thus correctly accounting for

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the differences among fundamental, intermediate and prompt modes. A new algorithm was also developed for an efficient computation of these modes. From the thermohydraulic point of view, changes in the modules involved in the coolant calculations were needed. Water boiling is described by a single pin slip model algebrized by the characteristic method and then integrated by the method of approximation of exponential operators, as developed in /6/. The coolant flow rate radial redistribution during the transient, which may be important in some accident sequences, will be implemented in NADYP-LWR in late 1989. (ENEA-VEL, Casaccia)

3.1.2 Void Reduction in BLW/HWR. The method developed for the reduction of coolant void reactivity in light water cooled, heavy water moderated reactors /7/ was applied to CIRENE, in an unsuccessful attempt to revive its licensing process which had been declaredly suspended on the grounds that the core had a large positive void reactivity feedback. The feasibility of reducing void reactivity to virtually zero without changing the structural characteristics of the fuel elements was demonstrated. Safety-related implications of the modified design were also reviewed in a preliminary analysis, which showed that the control system response and the plant behavior in several hypothetical severe accidents would be fully satisfactory /8/. (ENEA/ENEL-CIRENE, Casaccia)

3.2 Monte Carlo Work

3.2.1 Theoretical Advances. An effort has been underway for some years in the field of Monte Carlo variance reduction methods, in support to ENEA activities in the area of fast reactor core and shielding design. It has been focused on optimizing splitting and russian roulette parameters and has employed the general purpose shielding code, MCNP, as the vehicle in its implementation. Having dealt successfully with purely spatial splitting, the energy variable has been introduced, and the optimization of joint energy/space splitting is nearly completed. Work in progress involves the extension of the method to coupled neutron-gamma problems /9/ and to optimizing discrete angle biasing at neutron and gamma rays collisions. (ENEA-VEL, Bologna)

3.2.2 Shielding of Fuel Casks. As a contribution to NEACRP's intercomparison of codes for the shielding assessment of transportation packages, an extensive study was carried out in order to evaluate equivalent dose rate distributions and point values at various places outside the flask wall, with high degree of heterogeneity. The effects of the source axial distribution and spectral dependence on the response were evaluated as well /10/. The analysis is still being carried out using the MCNP code with various estimators and variance reduction techniques, and is demonstrating the flexibility and reliability of the code for the design of fuel transport flasks. (ENEA-VEL, Bologna)

3.2.3 Computerized Criticality Guide. Monte Carlo codes have been widely used in the calculations of a large number of benchmarks considered for the validation of the knowledge base of an expert system ("SEPI") which has been developed in the form of a computerized criticality guide supporting fuel cycle plant design and operation /11/. The system is characterized by an interactive computer-aided calculation module which relates a computerized procedure (developed in such a way to reproduce the designer's "reasoning process") to each step of the design process. The safety margins associated with the criticality guide are analyzed in /12/. (ENEA-COMB, Casaccia and DISP, Rome)

3.2.4 Static Analysis of Chernobyl Core. Time-independent Monte Carlo simulations were made of different Chernobyl core conditions, to assess the contributions of various possible mechanisms to the second power burst. Fuel fragmentation followed by bursting of the central channels, due to contact between the hot fuel particles and the cooling water at the end of the first peak, could have started a mechanism capable of producing a second reactivity peak, provided that fuel damage was extended to a sufficiently large portion of the core /13/. (ENEA-COMB, Casaccia and DISP, Rome)

3.2.5 Applications to Reliability Analysis. Efforts have been made to investigate the potential of Monte Carlo methods for the calculation of the time dependent reliability and/or availability of complex systems. An attempt to illustrate possible applications was made in /14/, where probability estimates for a rare event (unavailability of the decay heat removal system for a LMFBR) are obtained by using biasing techniques similar to those used in deep penetration shielding calculations. (ENEA-VEL, Casaccia)

3.3 Neutron Noise Analysis Methods

3.3.1 Core Barrel Motion. Data from Borssele PWR, distributed as a part of the SMORN Physical Benchmark, are still being analyzed -with the main purpose of extracting useful information relating to the short-term variation of the vibration pattern- via filtering techniques and Hilbert transforms. (Polytechnic of Milan)

3.3.2 Pu Monitoring in Waste Drums. Statistical methods typical of neutron noise analysis are applied for the non-destructive determination of Plutonium content in waste drums from fuel reprocessing plants. In the actual experimental setup, the sealed drums are contained in a cylindrical well equipped with He-3 counters which may be located in two rings, thus permitting measurements with drums of different sizes. The drums may also rotate or oscillate so that the methods for the localization of a vibrating control rod in a reactor core, developed in the past years /15/ are also suitable for locating point sources of neutron noise /16/. Concerning the analysis of the experimental data, a new Monte Carlo code simulating the whole experiment has been written. The code supplies the time sequence of the pulse and also the group constants relating to different energies and zones. These constants are then used in another home-made code which computes the mean value, variance, and time/space covariance of the pulses in a time varying multi-energy, multi-zone model. Preliminary experiments with a set of Cf-252 and (α,n) sources are scheduled in November 1989. (Polytechnic of Milan)

4. DATA EVALUATION, PROCESSING AND VALIDATION

4.1 Nuclear Data Evaluations

In the frame of the international co-operations for JEF/EFF files, the evaluation of n + Fe (natural isotopes) cross-sections has been performed. The neutron energy region above resolved resonances from 0.5 MeV up to 20 MeV was considered. Particular emphasis was placed on the inclusion of pre-equilibrium effects (exciton model) and on the model parametrization of nuclear level densities /17/.
The Si-28 file evaluated at ENEA Bologna and included in EFF-1 has been critically compared with the recent BROND file. Now, the home evaluation is in an advanced state of progress with main reference to gamma production data calculations and pre-equilibrium component estimates in emitted neutron spectra. Double-differential distributions are correctly estimated by model calculations. (ENEA-TIB, Bologna)

4.2 Data Management and Processing Codes

The participation in two benchmarks on Thermal Fission Products (NEACRP) and Fusion Blanket/Shield Corrosion Products (JRC Ispra and Garching) offered the opportunity to carry out some revision and updating work on ORIGEN's Master Library and on the data processing procedures used to produce working libraries for burnup, activation and decay calculations /18/.
In the framework of the last activities in support to CIRENE's licensing review, a new corrected version was produced of WIMSCORE/ENEA /19/, a code used to process the interface output files of WIMS/D-4 and to generate two-group data for core

calculations. (ENEA-TIB, Bologna)
In co-operation with ECN-Petten, the group library GEFF was produced from basic EFF-1 data, mainly for tritium breeding, shielding and energy deposition studies in fusion reactor design. Work is in progress on the intercomparison of best available Kerma data (from MACKLIB) with those computed by NJOY 87.1 starting from EFF-1,in view of an improved method for calculating Kerma factors. Besides, a complete working library for MCNP has been produced from EFF-1 via NJOY-ACER processing /20/ and is now ready for testing on a shielding benchmark configuration, according to the specifications recommended by the ITER team. (ENEA-TIB, Bologna and ENEA-FUS, Frascati)
As a contribution to the development of a unified AGT-3 formulaire for LMFBR cores, the MERGE code -which combines the probability table produced by CALENDF with the group data generated by THEMIS- has been designed and implemented. The code is already in use: one of its features is to accept both THEMIS and NJOY GENDF formats. (ENEA-TIB and ENEA-VEL, Bologna)

4.3 JEF Data Validation

Extensive computational work has been carried out in order to assess the validity of the new Cr isotope evaluations, with particular emphasis on the gamma-ray production files and the comparison with existing evaluations and experimental data. For this purpose the version 87.1 of NJOY implemented on the new IBM 3090/30E computer has been used.
The multigroup neutron library VITAMIN-J has been validated together with additional nuclides processed from JEF-1 in the same 175 group structure through the NJOY/THEMIS system. To this purpose, one- and two-dimension discrete ordinate transport calculations have been performed on the (H_2O/Fe) PCA-Replica shielding benchmark, proposed by the NEACRP for nuclear data validation /21/. Work is also in progress aimed at identifying possible systematic error contributions in covariance data estimates. Based upon a technique firstly developed by Y. Venohara, Kyushu University, the method uses an expert system to scan experimental data from EXFOR to extract the needed information; ancillary programs put the data in ENDF format. (ENEA-TIB, Bologna)

5. FUSION BLANKET NEUTRONICS

The blanket design activities carried out by the ENEA team (Frascati) follow two different programme lines:

- design of a reactor-relevant ceramic breeder, helium cooled blanket to be tested as a single sector during the technological phase of NET operation. In this area, neutronics calculations have been made to analyze and compare the performances of the different concepts proposed. ENEA has strengthened its co-operation with CEA (Saclay) in an attempt to converge to a single design, as recommended by the European Community;
- design of a ceramic breeder, water cooled driver blanket which should contribute to the tritium supply for the NET/ITER system operation, but would not be required to have a full reactor-relevance. The effectiveness of the neutronics analysis of this component has been enhanced by the calculation of radial TBR sensitivity profiles with respect to the material configuration. Sensitivity coefficients have been interfaced with an optimization code in order to determine optimum solutions for different "payoff" functions taking into account design constraints /22/. (ENEA-FUS, Frascati)

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REACTOR PHYSICS ACTIVITIES IN JAPAN

October 1988 - September 1989

Compiled by
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1. INTRODUCTION

Analytical and experimental efforts have been continued to support the developments of Liquid Metal Fast Breeder Reactor (LMFBR), Advanced Thermal Reactor (ATR) and High Temperature Gas Cooled Reactor (HTGR), which have been promoted as the national projects. Interest for the innovative design study with aiming at increasing the inherently safe features and the very high conversion ratios has been emphasized for advanced fuel loaded FBRs. Reactor physics studies on HTGR, including critical experimental VHTRC (JAERI) have contributed to validate the core design of the High Temperature Engineering Test Reactor (HTTR).

Attention was drawn to some other topics in reactor physics. One of these topics is concerned with High Conversion Light Water Reactor (HCLWR). Critical experiments are being performed at FCA (JAERI) and KUCA (Kyoto University). Improvement and assessment of HCLWR core design methods were proceeded. Remarkable trend is an attention to the criticality safety problems. This trend has been enhanced by starting the construction of nuclear fuel cycle safety experimental facility (NUCEF) program at JAERI.

Investigations related to the incineration of transuranic nuclide (TRU) have been extended as the Omega Program is initiated. Development of the neutronics computer codes treating the spallation reaction and neutron transport and also design studies of TRU burning reactor and accelerator driven reactor have been performed.

Much efforts have been devoted to the blanket neutronics of fusion reactor at JAERI and universities. The major parts of the 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 collaborative research program between JAERI and US-DOE.

Concerned with radiation shielding, continuous studies have been made for the radiation streaming problems at various facilities. Recently, much efforts have been paid on benchmark problems, of which the major activity is the correspondence to the shielding benchmark connected with NEA and ICRS7.

It is also to be noted that a new program to support creation of reactor concept and facilitate reactor design, named ADES (Advanced Design and Evaluation System for new type reactors) was initiated.

2. NUCLEAR DATA EVALUATION

Thermal and fast reactor benchmark calculations have been performed for a temporary nuclear data file JENDL-3T/R1 which was revised from JENDL-3T(1) for several important nuclides such as ^{235}U, ^{239}U, ^{239}Pu and nickel, to assess the adequacy for use in nuclear reactor designs and applications.

The lattice experiments of TRX and ETA cores, many critical experiments at ORNL, PNL and the other facilities using aqueous solutions for ^{235}U, ^{239}Pu and ^{233}U fuels and the PROTEUS experiments were selected as thermal reactor benchmark cores. As the benchmark results, the k_{eff}'s calculated for ^{235}U fueled cores underestimated the experiments, and those for ^{239}Pu fueled cores gave good agreement with the experiments. Furthermore, the k_∞ for PROTEUS cores agreed very well with the experiments in any moderator voidage states. However, the fission and capture reaction rate ratios related to ^{238}U were not in good agreement with the experimental values.

The benchmark cores selected for fast reactor were 22 critical assemblies which have a wide variety of characteristics and the FCA-IX assemlies with uranium fuel which cover a wide of neutron spectrum shapes. The k_{eff} obtained with the JENDL-3T/R1 data were in good agreement with the experimental data for plutonium fuel cores and were underestimated for uranium fuel cores.(2) The overestimates for sodium-void worth and ^{239}Pu fission reaction rate distribution obtained by JENDL-2 were remarkably improved by JENDL-3T/R1.

Modification of the ^{238}U capture cross sections has been made on the basis of the benchmark test of JENDL-3T/R1. The compilation for final data of JENDL-3 has been completed, and 325 nuclides including 172 fission product nuclides have been stored.(3) The detailed benchmark tests of JENDL-3 has already started. The final data of JENDL-3 will be opened on the end of September, 1989.

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3. CALCULATIONAL METHOD DEVELOPMENT

During the period Oct. 1988 - June 1989, various works were performed on calculational methods of multi-dimensional transport and diffusion equations, and related computer codes have been developed.

A three dimensional first order perturbation method based on the diffusion theory has been developed to calculate reactivity change due to core deformation of LMFBRs at accident events. Macroscopic cross sections and anisotropic diffusion coefficients in deformed cores are defined to be used in the same mesh division as normal core calculations. Computation time was significantly reduced compared to a direct eigenvalue calculation method.(1)

A second order differential equation has been derived for multi-group transport equation using a spherical harmonic moment. The equations were solved by the finite difference method in r-z geometry. This method gives more accurate solutions compared to the discrete ordinate method for shielding problems with strong absorbers because of disappearance of the ray effect.(2)

Various works were continued to improve nodal methods to solve three dimensional diffusion equations in hexagonal-z geometry. Characteristics of each method are (1) to use a polinomial expansion method based on curvilinear coordinate transformaton(3), (2) to solve directly three dimensional equations using high order finite difference technique(4) and (3) to apply the group theory in analytical formulation of neutron flux distribution(5).

A series of transport codes using multi-group double differential cross section has been developed for one, two and three dimensional calculations. The method can treat accurately strong anisotropy of scattering and overcome negative flux problems encountered in a conventional Legendre expansion method(6).

Efforts were devoted to develop vectorized Monte Carlo codes. The GMVP code has been developed for general purpose in production use and achieved speedup of 7 - 10 times compared to conventional scalar codes(7). A new technique was proposed to reduce computation time of a psudo scattering method in the Monte Carlo method by treating strong absober regions as special regions.

A new project to support creation of reactor concept and to facilitate reactor design, named ADES (Advanced Design and Evaluation System for new type reactors) was initiated under the direction of Science and Technology Agency (STA) of Japan. It is characterized by an intellectual integration of computer codes, data and knowledge related to reactor design by the use of the information processing technology in the state-of-arts. The committee organized by STA having discussed the program for one year concluded that the program realizes the following matters;

(1) To facilitate a foreseeing drive of design steps each of which is tends to be sophisisticated and specialized.

(2) To possesse as a common property the knowledge and experiences accumulated in particular persons.

(3) To economize design work by substituting mockup experiment by accurate simulation.

The project is planned to be completed in 10 years by JAERI and PNC with the aid and support of knowledge engineers, plant designers and plant operators in other institutions under the direction of STA.

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4. FAST REACTOR PHYSICS

A series of benchmark critical experiments for axial heterogeneous Liquid Metal Reactors (LMRs) conducted at the ANL's ZPPR facility as a part of the USDOE/PNC JUPITER-III program, were analyzed in Japan by a Japanese core analysis method using the JENDL-2 library. A radial variation of C/E results was observed for reaction rates, control rod and sample reactivity worths. Reaction rates were underpredicted in the internal blanket as well as in radial and axial blankets, and as the result sample reactivity worths were underpredicted in the internal blanket.

The physics parameters of axial heterogeneous LMFBR core, measured using partial mock-up assemblies at FCA[1], were analyzed. The results were compared with those of prior experiments with assemblies representing conventional homogeneous core. The calculation disagreed to some extent with the experiment in the internal blanket.

Neutron streaming in a fast breeder reactor fuel subassembly caused by the double heterogeneity of the pin structure and the wrapper tube structure is estimated by double heterogeneous modeling[2]. The neutron streaming is decomposed into three components: the pin-cell heterogeneity, the wrapper tube heterogeneity, and the homogenized fuel/wrapper tube subassembly effect. The streaming effect is evaluated based on the Benoist's diffusion coefficient. The total streaming effect caused by the double heterogeneity structure of a fuel subassembly is found to be about $-0.2\%\Delta k/kk'$ for k_{eff}, which is almost twice as much as that obtained from the conventional pin-cell model of about $-0.1\%\Delta k/kk'$.

Three-dimensional diffusion and transport theory calculation methods have been developed based on nodal method and Sn method. SIXTUS-2 is a two-dimensional multigroup diffusion nodal method code in hexagonal geometry, developed by J.J. Arkuszewski. SIXTUS-2 has been extended to 3-D hexagonal-Z geometry. Comparing the calculational results by SIXTUS-3 with those by finite difference method, it was found that SIXTUS-3 corresponds to 24 mesh calculation rather than 6 mesh calculation, and the calculation time of SIXTUS-3 is the same as that of 6 mesh calculation.

A nodal solution method[3] of neutron diffusion equations, using a higher-order finite difference scheme, was developed. Nodal solution techniques based on an interface current method have been developed for efficient neutron diffusion calculations in Cartesian geometry, and were applied to analyses of light water reactor cores. The method was extended to the hexagonal geometry for fast reactor cores, and test calculations gave sufficient accuracies with reduced computation times, compared with usual nodal method calculations.

An improved technique for inferring the eigenvalue separation, which is important in spatial stability analysis, was developed using the noise coherence function.(4) It was applied to fast reactor critical assemblies of various sizes and compositions which exhibited a wide range in spatial decoupling. In each experiment four lithium-glass detectors were used to measure noise coherence functions. Various ratios of the coherence functions were used to obtain the first two modes separation with consideration of higher modes and variations in detector efficiencies. The eigenvalue separation obtained by noise analysis gave good agreement with calculation.

Critical assembly experiments on large fast breeder reactor were analyzed by a current neutronics analysis method for fast reactor core, using a group constant set based on the JENDL-2 library. It was made clear that there were some radial dependence of C/E values for integral physics parameters, in addition to discrepancies between calculation and experiment. A cross section adjustment was performed not only so that the radial dependence of C/E values might be solved, but also so that other C/E discrepancies from unity might be minimized at the same time. As the result of adjustment, the remarkable disagreement between calculation and experiment has been significantly improved, and also the Pu-239 fission cross section was increased by 2% for the energy range below 10 keV, and the U-238 capture cross section was decreased by about 6% for the 1 keV - 1 MeV range.

The formulas of prediction uncertainty of neutronic performance parameters have been derived for three methods, the bias factor method, the adjustment method and the combined method.(5) The prediction uncertainties were obtained by including the both experimental error and method error. It was found that the adjustment method, in principle, yields the same uncertainty as the combined method. The derived formulas have been applied to a large homogeneous FBR core of 1000 MWe.

Investigation of a future core program of JOYO, following the present MK-II core, has been started.[6] The reactor core will be expanded radially, in order to increase the irradiation test rigs which can be loaded in the core at one time. In addition, to decrease the time needed for the irradiation objectives, both the modification of the fuel subassembly to obtain higher neutron flux and the improvement of the fuel handling system to shorten the outage time for fuel exchange are investigated. R&Ds for innovative technologies are also planned. For instance, studies needed to eliminate the secondary heat transport system of fast breeder reactors are picked up as a future plan.

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5. THERMAL REACTOR PHYSICS

5.1 High Temperature Gas Cooled Reactor

Since the construction of the High-Temperature Engineering Test Reactor (HTTR) went into the fixed schedule, the operation of the critical facility VHTRC has been concentrated to provide the experimental data to verify the accuracy of calculations related to the neutronics design of the HTTR. A series of measurements for the simulated control rod worth was done in the fundamental core VHTRC-1 loaded uniformly with 4% enriched uranium fuel where a control block is installed at the central axis of the core. As the result of analysis, it is shown that the 2-D S_4 calculation instead of the usual collision probability method is needed to have proper homogenized cross sections of the block in the case of the control rods inserted and that the anisotropic diffusion caused by the three holes in the block should be taken into account in the core calculation in the case of the control rod withdrawn. At present, experiments are conducted for the VHTRC-4 core which is loaded with the 2, 4 and 6% enriched fuel in an axially zoning pattern. Experiment covers critical masses at the room temperature and 200°C, flux distribution by $^{63}Cu(n,\gamma)$, reactivity worth of the burnable poison rods. The prediction performed by the SRAC code shows the satisfactory results within the required design accuracy.

5.2 Research Reactors

Along the international Reduced Enrichment for Research and Test Reactor fuels (RERTER), the core conversion of the JRR-2 of JAERI from the use of highly enriched uranium (HEU) fuel to the medium enriched uranium (MEU) fuel (45%) was achieved November, 1987. The JRR-2 is a D_2O cooled D_2O moderated 10 MW reactor. The core is composed by 24 fuel assemblies and 6 control rod channels arrayed in a hexagonal lattice pattern. The fuel assembly consists of a multi-layer of cylindrical fuel plates. A series of neutronics characteristics measurements were conducted. These were well predicted such as the cold critical mass by the C/E value 0.990, the excess reactivity by 0.982, the total control rod worth by 0.967, the radial peaking by 0.973 by using the SRAC code system. It was verified that the effect due to the conversion is little from the view point of operation and utilization.

The JMTR (a swimming pool type reactor of 50 MW) loaded with the medium enriched uranium continues the normal operation since the core conversion from HEU to MEU realized in 1986. At present, the next core conversion to use low enriched silicide fuel is under preparation. At the same time, the prolongation of the cycle length from 14 days to 24 days is intended. A burnable poison to use Cadmium wires embedded in every side plate is considered. The number and the thickness of Cd wires were chosen so as to keep sufficient reactivity and duration. About 18 wires of 0.4 mm dia. per assembly are expected to effectively burn out at the middle of the cycle. The geometrical modelling on cell calculation for such a highly heterogeneous assembly has been validated by use of a continuous energy Monte Carlo code VIM.

Both experimental and analytical studies[1] have been performed on the temperature coefficient of reactivity in a light water moderated and reflected core loaded with HEU fuel at the Kyoto University Critical Assembly (KUCA). The temperature effect on reactivity was measured for the 20 to 70°C range to investigate separately the effect of the $H/^{235}U$ atomic ratio and the core shape on this quantity. The calculated results by the SRAC code system approximately reproduced the experimental data. It was found that the contribution of the core region to the temperature coefficient was negative due to the degradation of moderation. The positive contribution of the reflector region became larger as the $H/^{235}U$ atomic ratio became smaller and the core shape became more slender.

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6. HCLWR PHYSICS

Since common understanding has been established for the data sets and the computational methods used in HCLWR analysis and design in Japan, major portion of efforts has been devoted to design studies.

A design study of plutonium generation BWR continues to be carried out.[1,2] An HCBWR with effective V_m/V_f = 0.25 and 900 MWe output has been designed so as to produce the same amount of plutonium as the consumed plutonium using natural uranium. This reactor has a non-positive void reactivity coefficient. Moreover, an axially zoned HCBWR core has been proposed.[3] In this concept, the upper part of the core consists of MOX fuelled assemblies of tight pitch lattice, where the void fraction of coolant is higher, and the lower part of the core uses the conventional BWR assemblies.

A new concept of axially heterogeneous HCLWR was proposed to achieve high conversion ratio and high discharge burnup, while maintaining a negative void coefficient.[4] Design study of a double-flat-core HCLWR based on three dimensional core burnup calculation shows that considerably high value is obtainable for both the conversion ratio and discharge burnup. In order to generalize the the double-flat-core HCLWR to an HCLWR with multiply stacked cores, an optimization study has been made for the basic core parameters, core height, blanket thickness, V_m/V_f and so on.

Additional NEACRP HCLWR benchmark calculations with any of continuous energy Monte Carlo codes, proposed at the first specialist meeting, were performed by the VIM code with the library based on JENDL-2. The calculations agree very well with the experimental PROTEUS-Phase I results for k_∞ but not so much for reaction rates. In connection with the benchmark calculations, a series of calculations were carried out with emphasis on resonance methods, effects of geometrical modelling and boundary conditions, in order to assess and verify the accuracy of cell calculation methods used in analyses of tight lattice systems.

Reactor physics characteristics of a high conversion light water reactor (HCLWR) were investigated using plutonium-fuelled zone-type FCA-XV cores which simulate various moderator voidage states of the HCLWR core. Analyses on the measured data were made by using the SRAC code system with the JENDL-2 data file.

(1) The calculated values of k_{eff} show good agreement with the measured ones for all the moderator voidage states.

(2) The discrepancy between the calculation and experiment for k_∞ increases with increasing moderator void fraction. The calculation overestimates the measured k_∞ value by about 2% at 95% moderator void fraction. Alternative experimental techniques are being applied for determination of k_∞ based on the cell reactivity worth to reduce the experimental error at higher voidage states.

(3) The calculation predicts well the reactivity worths of the absorber material (B_4C and H_f) and plutonium with different isotopic compositions. The C/E values of B_4C worth do not depend on ^{10}B contents of the B_4C absorber for all the void fractions.

(4) Large discrepancies between the calculated and experimental values of the reaction rate ratios are still observed in higher moderator voidage states. To explain these discrepancies, further study is being made by using JENDL-3 data file.

A basic study on the tight-pitch lattice cores has been performed since 1986 at the Kyoto University Critical Assembly (KUCA) as a joint study of a university association. A series of critical experiments has been carried out using uranium fueled cores. In the experiments, V_m/V_f and the average enrichment of ^{235}U were systematically varied to investigate the nuclear characteristics of the tight-pitch lattice cores. Through the analyses, the validity of the computational methods including the nuclear data libraries were assessed in comparison with the experimental results and the calculations by a cotinuous energy Monte Carlo code VIM and a multigroup Monte Carlo code KENO-V.(5) Here, a special attention was paid to the transport and the anisotropic scattering effects, since the KUCA core is very small in size and polyethylene is used as moderator and reflector.

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7. ADVANCED CORE DESIGN

The conceptual design of a passively safe and simplified light water reactor progressed. A 700 MWe plant with two modulars of a passively safe and simplified light water reactor, SPWR (System-integrated PWR) was proposed.[1,2] It contains an annulus steam generator in the reactor vessel and is controled by borated water without use of any control rod. A poison water tank envelopes the reactor core and supplies borated water to the core by automatically opening hydraulic pressure valves at drop of delivery pressure of a primary pump which is installed at the top of the reactor. Nuclear and hydraulic analyses have been performed. By simplification of the reactor and shortening of the construction period make construction cost equivalent to or less than that of a corresponding conventional PWR plant.

Aiming high burn-up up to 100 GWD/t, the burning process of a PWR has been assessed by means of spectrum shifting for a wide range by scattering movable absoption rods of natural uranium oxide fuel all over the core of 15~20% enriched plutonium fuel to suppress the initial excess reactivity and to keep void coefficient negative through the core life.[3] Natural uranium oxide rods can be also used as control rods. Thermal peaking after extraction of these rods was evaluated and it was cleared not to so large as to affect on the core characteristics. This concept of long burning is also applicable to TRU burning by thermal spectrum in light water reactors with small residual quantity of TRU.

Two concepts of TRU burning reactor only with TRU fuel have been assessed with metallic fuel/sodium cooling and coated particle nitride fuel/helium cooling.[4] A plant with six 170 MWt reactor modulars of metallic fuel can incinerate TRU of more than 300 kg annualy by fissions with hard neutron spectrum and high power density. Equivalent capability can be achieved by a 1200 MWt reactor of coated particle fuel.

To achieve high plutonium production capability as the most essential feature of the fast reactor, a DU-Pu-Zr alloy fuel fast reactor of 670 MWe was proposed with a new concept of FP gas purge/ tube-in-shell type fuel assemblies.[5,6] Fuel occupies more than 50% of the volume fraction of the core and its very hard spectrum provides high breeding ratio up to 1.8 and reactor doubling time be less than 7 years. The structure of fuel assemblies, their fabrication procedure, mechanism of FP gas purge were engineeringly assessed and the concept was cleared to be feasible. Because of good retaining capability of filling sodium as expected, FP gas purging does not affect on shielding. The scale of the cover gas treatment system is small enough to be easily installed in the reactor container. The concept of FP gas purge is applicable also to the conventional pin-bundle type metallic fuel fast reactor. Even with metallic uranium fuel, high plutonium production capability by this concept.

A preliminary assessment has been done on a concept of a very small fast reactor of 1.5 MWt for the use in the isolated circumstances, with coated particle fuel and a heat-pipe cooling system of lithium. A highly enriched uranium core is surrounded with beryllium reflector in which neutron absorber drums rotate to control reactivity. Generated heat is transported to thermo-electric converter by primary heat pipes.

Analyses have been made on neutronics performances for a partial-refueling ultra-long-life core (ULLC) using metallic fuel for 1000 MW (electric) liquid-metal fast breeder reactors[8]. Once this core is initially loaded, only fertile materials are needed as core reload fuel for the rest of the reactor lifetime, taking advantage of the superior breeding characteristics of the metallic fuel. The fuel management strategy demonstrates the core concept and establishes relevant performance parameters such as a manageable reactivity swing and flat power distributions over the burnup cycles. The following advantages of this ULLC concept over the nonrefueling ULLC were found: smaller control reactivity requirements over the cycle (up to ~1/3), lower power peaking factor (~3/4), and lower power swings during burnup (~1/3).

Recently new approaches such as higher actinide transmutation using actinide burner reactors, LMRs (metal fuel FBR) as described in the above and spallation reactions with proton accelerators have been studied.

A conceptual design work is performed on a concept of a high current proton linear accelerator system with a sub-critical system of TRU metal fuel with k_{eff} of 0.95 with stainless steel reflector.[7] The size of the core is 113 cm in diameter and 150 cm in height in the case of sodium cooling and a little bit larger in the case of Pb-Bi cooling. The proton beam is 10 mA with the energy of 1.5 GeV. The

k_{eff} of 0.95 with stainless steel reflector.(7) The size of the core is 113 cm in diameter and 150 cm in height in the case of sodium cooling and a little bit larger in the case of Pb-Bi cooling. The proton beam is 10 mA with the energy of 1.5 GeV. The system can incinerate the equivalent quantity of TRU from ten LWR's of 1,000 MWe. Transmutation of fission products such as ^{90}Sr and ^{137}Cs with an electron accelerator for photo nuclear reaction is proposed by PNC. The optimization of higher actinide recycling into Pu-LWR and MOX-LMFBR is also studied.

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8. FUSION REACTOR NEUTRONICS

A new unfolding code UFO/Q[1] for the NE213 spectrometer, based on the least squares fitting with Lemke's method, was developed at Tokyo Institute of Technology, and a neutron spectrum calculated with this code was compared with the FERDOR spectrum. It became clear from the comparison that the UFO/Q code gave good results, i.e., reductions of oscillations and error, and no negative values. The tritium production density, kerma heat production density, dose and certain integral values of scalar neutron spectra in bare and graphite-reflected lithium-fluoride piles irradiated with D-T neutrons were evaluated from the pulse height distribution of a miniature NE213 neutron spectrometer with the UFO/Q code. The results were compared with the values calculated with the MORSE-CV code[2].

Two types of experiments were performed at OKTAVIAN. One is the measurement of gamma-ray energy spectra from eleven materials fusion reactor such as W and Pb. From the comparison with the analysis by MCNP, the accuracy of nuclear data for low energy gamma-ray spectrum in JENDL-3T seems to be better than that in ENDF/B-IV[3]. The other is a new approach in the fusion neutronics, i.e., the measurement of time-dependent reaction rate profiles in simulated blanket assemblies. The data have the information of energy and can be used for the integral evaluation of nuclear data below 14 MeV.

Helium production rate in stainless steel was measured by Kyushu University using OKTAVIAN. The measured data agreed well with previous works.

The analysis of Phase-IIB experiment of JAERI/USDOE Collaborative Program using the FNS facility at JAERI was made and reported at the ANS 9th topical meeting[4,5]. It was pointed out that the accuracy of predicted tritium production rate in simulated Li_2O blankets with Be was worse than that without Be. This fact suggests that the accuracy of Be nuclear data in JENDL-3T and ENDF/B-IV is not enough for the estimation of tritium breeding rate. The Phase-IIC experiment has been carried out to examine the heterogeneity effects of beryllium and water coolant. The measured data in the heterogeneous systems should provide the problems related to both experimental arrangement and calculational modeling, especially the parameters sensitive to low energy neutrons such as tritium production rate of ^{6}Li.

An integral experiment on a Be assembly was performed at FNS as a series of clean benchmark experiment. In this experiment, neutron spectra in the assembly were measured by a small sphere NE213 spectrometer and proton recoil counters as well as reaction rate distributions. Angular flux spectra leaking from load, iron and liquid nitrogen slabs were also measured by time-of-flight method. The sample irradiation for the NEACRP International Comparison on Measuring Techniques of Tritium Production Rate was performed using a Li_2O assembly. Five countries and eight organizations participated in this program.

Nuclear heating in structural materials was measured at FNS using newly developed technique. The technique was based on calorimetric method, i.e., thermo-sensor of Platinium resistance or thermocouples of CA was placed in probes of SS304, Cu or graphite, and the temperature rise due to 14 MeV neutrons was measured by a nano-volt meter.

The neutron response of α-track detector was examined analytically at Hokkaido University for fusion neutronics applications.(6) Tritium production rates in a simulated fusion blanket were measured by the α-track method using FNS.(7)

A cross section set of 125-group for neutron and 40-group for gamma-ray is being processed to be used for integral tests of JENDL-3. At this time the integral tests will be done not only for neutron but for gamma-ray data.

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9. CRITICALITY SAFETY

Using a multi-core type critical assembly Kyoto University Critical Assembly (KUCA), the criticality safety study has been performed. At a light-water moderator core C-core among the three cores of KUCA, the critical mass and the neutron flux distribution of a coupled-core were measured as a function of the separation distance between coupled two cores with varying the $H/^{235}U$ atomic ratio systematically.(1) The subcriticality down to approximately 35 $ was measured with use of the Feynman-α technique.(2) At a solid moderator core A-core, the nuclear characteristics of a core with non-uniform fuel distribution are being investigated. Here, the investigation is focusing on the relation between the fuel importance and the criticality.

The Joint United States and Japanese Criticality Data Development Program for the Fast Breeder Reactor Fuel Cycle was finished its all work scope in August 1989. The experimental data obtained through this program have been analyzed for publication.

Experimental studies have been performed for the criticality safety using Tank-type Critical Assembly (TCA) at Japan Atomic Energy Research Institute. Temperature coefficents of reactivity in light water moderated and reflected cores with soluble poisons such as boron and gadolinium were measured to evaluate the safety margin of dissolvers in reprocessing plant. The reactivity effects of concrete were investigated for single cores and two-coupled cores. The basic characteristics of reflector effect and isolation effect of the concrete with different boron contents were obtained. In addition, the experimental technique for measuring the subcriticality are being investigated applying the Avery's method for two-coupled cores.

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10. SHIELDING

The Research Committee on Engineering of Radiation Behavior in Atomic Energy Society of Japan has established two working groups in 1988 for performing research activities on high-energy accelerator shielding through acquiring knowledge of high-energy radiation behavior, and also on neutron shielding design aiming at completing its handbook.

Many works for better shielding design were continued in this period. A new semiempirical formula which describes energy-space distributions of streamed neutrons and gamma-rays in cylindrical ducts was derived. The basis of formulation is the albedo model, and it is confirmed that the present formula well reproduces experimental data in several cases. Application to more ducts and slits was reported at the 7th International Conference on Radiation Shielding(ICRS7)(1). Empirical formulae were also derived from experimental data that described neutron flux distributions in steel-walled cylindrical ducts. The applicability of these formulae was made clear with respect to neutron energy, duct geometries, steel wall thickness, bent angle, and surrounding medium(2). An additional shield design method was reported for the Fe compensational shield which is located around the air gap of a straight slit, crank slit and cylindrical crank plug in concrete shield wall against gamma-radiation to compensate the lowering of shielding efficiency caused by these shield irregularities(3).

A neutron spectra study for a semi-monoenergetic neutron field which was developed using a simple Be target system bombarded by 20 to 40 MeV protons was reported(4). Neutron dose equivalents outside the concrete shielding of a 2.5-GeV electron linear accelerator were measured and analyzed(5). A shielding experiment to find out an optimum arrangement minimizing the total dose rate was carried out for iron-polyethylene slab shields with a ^{252}Cf neutron source. Appearances of the optimum arrangement were also reproduced by the Monte Carlo calculations(6).

The gamma-ray dose rate distributions in a low-level radioactive-waste shipping vessel in which as much as 3000 radwaste drums were loaded, were estimated by using the Monte Carlo coupling code system MORSE-CG/RADWASTE-VESSEL(7). Two studies on spent fuel transportation were performed. One is concerning neutron and dose rates on the surface of a spent fuel transport cask(8), the other is a new proposal for calculation of radiation dose rate distribution in a ship loading spent fuel casks[9].

A calculational method of photon dose equivalent was proposed, based on the revised technical standards of radiation protection law[10]. An interpolation of buildup factors for an arbitrary elemental material was examined using geometric-progression (G-P) parameters for an equivalent atomic number. Various tests over a wide range of atomic numbers confirmed that values by interpolated G-P parameters were in good agreement with the basic data[11]. A vectorized three-dimensional discrete ordinates code ENSEMBLE for neutron streaming analysis in a large and complex system[12], and a code package INTEL-BERMUDA aiming at the easy and automatical use as a standard shielding calculation tool[13], were presented at the ICRS7.

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11. NATIONAL PROGRAMS

11.1 JOYO

Following the 7th annual inspection, the reactor operated the 7th and 18th duty cycles from January 1989 to July 1989. Then, the reactor continues the 19th duty cycle operation from August 1989. The average burn-up of the core and the maximum burn-up of the driver fuel will reach to about 35,000MWd/t and 70,000MWd/t, respectively, at the end of the 19th duty cycle.

Nuclear characteristics of the core such as burn-up reactivity, reactivity changes due to power level and due to coolant temperature were measured during these duty cycle operations. The measured values of the nuclear characteristics except reactivity change due to power level did not change very much, for the duty cycle operations.

11.2 MONJU

The construction work of the MONJU plant, a 280MWe loop-type FBR plant, has been making steady progress since its commencement on October 1985. Overall construction work is now progressed more than 70 percent, and scheduled to be completed by April, 1991. Subsequently, pre-operational tests of components and reactor systems will be conducted for eighteen months, and the initial criticality is expected to be achieved in October, 1992. As a start up nuclear test, neutron flux measurements are planned for the core, blanket and radial shield using special assemblies which are equipped with activation foils. Neutron fluxes and spectra will be determined by these measurements and will be compared with the designed values.

11.3 Demonstration Fast Breeder Reactor (DFBR)

The development of Liquid Metal Fast Breeder Reactor(LMFBR) has been continued in Japan as a national development project. The present objective of the Japanese program is to develop large oxide-fuelled LMFBRs. Japan hopes to bring the development of LMFBR to a level of commercialization by the time around 2030, which can economically compete with Light Water Reactors (LMRs).

The Demonstration Fast Breeder Reactor now planned is expected to demonstrate the prospect of commercialization for LMFBR. The basic specifications of the plant will be defined around 1990, and the start of construction is scheduled for sometime in late 1990's, and the start of operation is scheduled for the beginning of the next century. The Japan Atomic Power Company (JAPC) is doing preparation works to define basic specifications of the plant.

11.4 FUGEN

The 13th refuelling was carried out on January 1989 and 8 UO_2 and 31 MOX fuel assemblies were charged. The Fugen continued stable full power operation until July 1989 when the 8th annual inspection commenced. The 8th annual inspection and the 14th refuelling, is being conducted from July to November 1989. At the 14th refuelling, 44 MOX fuel assemblies were loaded.

Five 36-pins experimental fuel assemblies, while the standard fuel assembly for the Fugen consists of 28 fuel pins, are now irradiated in the Fugen in order to develop 36-pins fuel assembly for the demonstration plant and the high performance fuel assembly.

Up to date, 294 UO_2, 300 MOX and 12 special fuel assemblies have been discharged for refuelling. The maximum burn-up is 19,900MWd/t for UO_2 fuel and 19,300MWd/t for MOX fuel, and no leaking fuel has been found for more than 2,490 effective full power days of operation up to July 1989.

The post irradiation examination of the MOX fuel assembly burned up to 18,200MWd/t has been carried out to investigate the performance of high burnup MOX fuel. The results ensured the adequacy of design methods for MOX fuel assemblies.

11.5 ART Demonstration Plant

The construction program of the ART Demonstration Plant has started with the decision given by the Japan's 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.

ART is a heavy water moderated boiling water cooled pressure tube type reactor originally developed by PNC. EPDC took over the results of PNC's design development work for the ART Demonstration Plant, and started the plant design work. At present, EPDC is finalizing the design based on recently completed design rationalization work, and preparing for an application for construction permit.

The capacity of the plant is 606MWe and the whole core can be fuelled with MOX fuels. The plant is expected to be located at the site in Ohma-machi Shimokita-gun, Aomori-ken. According to the current schedule of the project, the construction is to start in 1993, and the commercial operation in 1999.

11.6 High Temperature Engineering Test Reactor

Much efforts have been done for research and development of a High Temperature Gas Cooled Reactor (HTGR) in Japan, which includes high temperature engineering, coated particle fuel technology and plant design studies. In April 1989, JAERI made application for the construction permission of High Temperature Engineering Test Reactor (HTTR). The HTTR is a block type graphite moderated, helium gas cooled reactor, loaded with low enriched uranium coated particle fuel. Its power is 30 MWt and out put gas temperature is 850°C ~950°C. The expected construction site is Oarai-machi, Ibaraki-ken.

The objective of the HTTR is

(1) establishment of HTGR technology,

(2) fuel irradiation test in large scale,
and

(3) test for passive safety graphite modular reactor.

11.7 Partitioning and Incineration of Radioactive Nuclear Wastes (Omega Program)

Studies of basic technologies for the transmutation of nuclear wastes and nuclide partitioning have been continued for last several years, although the vitrification and geological disposal techniques are widely supported in the community. The expert committee, which was set up in the Japanese Atomic Energy Commission, discussed the problems of high-level nuclear waste managements. The Commission has concluded in June 1987 that R&D efforts for these technologies should be substantially strengthened as the national research project, where the possible use of valuable resource in the wastes and aggeressive improvements of safety assurance in management processes have to be extensively evaluated. This project is named the Omega Program. The Committee for Technical and Economie Studies on Nuclear Energy Development and the Fuel Cycle (FCC) of NEA convened the expert committee for an international collaboration on "Making Extra Gains from Actinides", where it is confirmed that studies on actinides partitioning and transmutation should not be presented in opposition to current policies on the geological disposal of radioactive wastes. JAERI is now making a proposal to construct a high intensity proton linear accelerator in order to develop the incineration technology with use of the spallation reactions.

Report on the Reactor Physics Activities in the Netherlands in the period September 1988 - August 1989

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. On-line monitoring of reactor noise.
2. Activities in the field of neutron metrology.
3. Installation and use of a new code package for thermal reactor physics calculations.
4. Evaluation, development of methods, calculation and testing of neutron cross-sections, especially for fission products, for fusion materials and for use in calculations for the High Flux Reactor (HFR) at Petten (no report available).

1.1. Experience with on-line power reactor noise monitoring system for the Borssele reactor (E. Türkcan).

On-line monitoring of the Borssele Nuclear Power Plant [1] (PWR-450 MWe 2 loop-system built by KWU in 1974) has been continued for the 15th and 16th core cycles.
In the course of the plant operation the reactor signals have been monitored continuously and surveillance methods improved. More attention is given to on-line determination of the physical parameters of the reactor as wel as on-line determination of the damping coefficient of the secondary system steam flow. On-line pattern recognition algorithms have been tested with multivariate signal spectra and decomposed spectra of the core-barrel motions [2]. Discriminants of Piety's algorithm [3] as wel as the others describing geometrical properties of the peaks in spectra are constructed during the learning period [4] and continuous checks on discriminants are included in the on-line surveilance algorithm.

Recently the method of signal detection through measured dc-signals [5] of the process by characterizing the statistical distribution is implemented to the on-line measured 32 reactor signals. It seems that it is possible to identify the characterization of the types of changes in the process. This study should be connected to the distinguishing false alarm and alarm failure.

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1.2. Neutron metrology (J.B.M. de Haas, H.J. Nolthenius)

To investigate the feasibility of a Boron Neutron Capture Therapy (BNCT) facility at the Petten High Flux Reactor (HFR), neutron spectrum measurements were performed at the entrance of the beam-holes under consideration. These measurements were done by means of the well-known foil-activation technique where the activation by 28 nuclear reactions, induced in the different detector foils, has been used to probe the neutron spectrum. Input spectra required for the spectrum adjustment, were calculated by the two-dimensional DOT transport code. Good agreement was obtained between the measured and calculated activities for all reactions. Angular group fluxes could be used to obtain the beam characteristics.

The promising outcomes resulted in a further investigation to design and construct a pilot beam for beam filter and spectrum shifter optimization. For this pilot beam new activation measurements have been executed which will adjust input spectra from MCNP calculations.

A guide has been edited with nuclear data for neutron metrology. This Nuclear Data Guide for Neutron Metrology (1988 edition) gives the data which are needed for the execution of neutron measurements in various neutron fields. For a selected number of nuclides which are used in this type of measurements, physical data comprising the number of atoms per unit mass and the mass density are given. Also physical information for the reaction product after neutron capture such as decay schemes, half-lives, cross-sections, gamma-energies and results for response calculations in a few reference spectra is supplied. The editing of this guide with data selected from open literature was done under favourable auspices of the Euratom Working Group on Reactor Dosimetry (EWGRD) and with contributions of several colleagues from other research institutes. According to the decisions of the working group the guide should be seen as an official recommendation of the EWGRD.
The guide will be published in the near future.

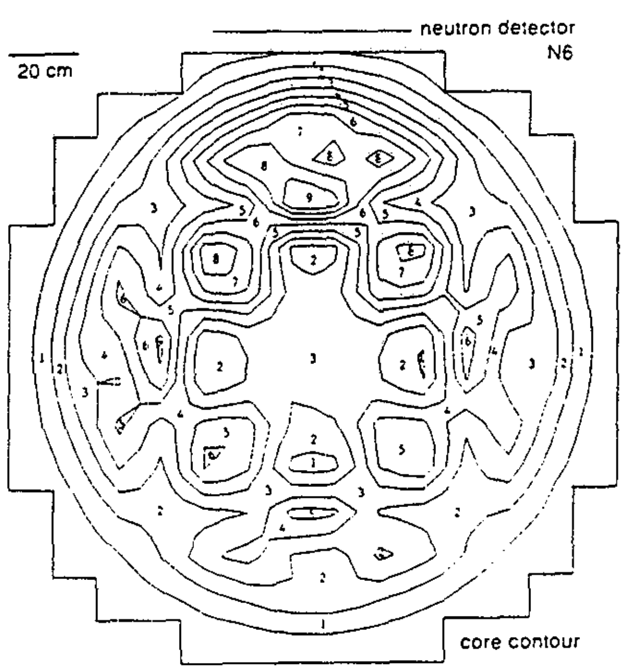
1.3. Application of AMPX/SCALE code system (H. Gruppelaar, J. Oppe, B.J. Pijlgroms, J. Slobben, Yu Peiha - guest from CNDC, Beijing).

The installation of the code systems AMPX-2 and SCALE-3 and several other codes (e.g. CITATION, ANISN, TWOTRAN, DOT 3.5) on the CDC-CYBER (NOS-BE and NOS-VE) is nearly completed; a large part of this system has also been installed on a SUN workstation. The NOS-BE version of this PASC-system (Petten Ampx / Scale Code system) has been handed over to the NEA Data Bank.

The PASC-system has been used for the execution of some ORNL benchmark calculations and for criticality calculations for storage of highly enriched fuel.

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

The research project on noise in BWR's, performed in connection with the Dodewaard nuclear power plant, was continued. Measurements performed after the annual refuelling showed a high degree of stability of this BWR with natural circulation cooling. For a better test of the stability monitoring, an asymmetrical power distribution was created in the core in order to decrease the stability. In this way it was proven that the stability can be adequately monitored with three ex-vessel neutron detectors, symmetrically positioned around the reactor vessel; both total-core stability and local changes in stability can be measured in this way. In order to study in more detail the "field of view" of these detectors, the theory developed at IRI to calculate complex frequency-dependent detector response functions was elaborated for multi-group neutron diffusion theory. It was confirmed that the standard computer code EXTERMINATOR is suitable for this type of calculations where negative cross sections and "fluxes" may arise and should not lead to interruption of the calculation. As shown in the figure, these calculations give a quantitative insight into the detector response to perturbations in the core as dependent on the position of the perturbation.



Spatial distribution of the influence of a local change of the moderator density on an ex-core neutron detector. The presence of four partly inserted control rods can clearly be seen. (1:10-20% of maximum value; 2:20-30%, ... 9:90-100%).

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With the experimental loop for boiling detection in the core of the IRI research reactor extensive measurements have been performed. Although boiling detection with noise methods was shown to be feasible, more detailed measurements are needed; it was proven that due to heat leakage from the loop the calculation of the power balance, which is needed for checking the onset of boiling, is hampered.
The reactor physics computer code package was extended by incorporation of the NJOY87 code for generating fine-group cross sections. A special module for the treatment of cross section resonances was developed, based on the XLACS code.
The feasibility of a new method for in-core flow measurement with a "signal cable free" turbine flow meter was studied; the experimental results are not yet conclusive but may lead to continuation of the project depending on external interests.
In the framework of the gaseous-core fission reactor (GCFR) project, supported by the Netherlands Technology Foundation, analyses were made of the dynamical aspects of a magnetically pumped GCFR. Neutron kinetics studies show that this system can be operated with a subcritical fuel mass as a consequence of non-linear effects. The chemical thermodynamics of the complicated fuel gas mixture was studied and used as input to calculations on hypothetical thermodynamical cycles. A real-time GCFR computer simulator was developed.
In the framework of a shielding project, supported by the Ministry of Economic Affairs, a flexible and user-friendly computer code MARMER was developed, based on the point-kernel method and furnished with a general geometry package. Several methods for treating build-up of radiation in layers of different materials were implemented. Many features were introduced to facilitate the preparation of input, and an updated data library was developed.
The second phase of a Noise Benchmark, supported by the NEA Data Bank, has been started during the report period. Fourteen groups from nine countries participate in this benchmark exercise which aims at accurate noise analysis of a multi-variable system and the assessment of methods for anomaly detection.

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3. Reactor physics activities at KEMA, Arnhem and GKN, Dodewaard
(C.D. Andriesse)

In January 1986 a prototype water level gauge was installed in the Dodewaard reactor vessel. This system uses eight different heated thermocouple sensors, called BICOTH, creating eight bit digital codes. The codes are uniquely related to 5 cm water level intervals in a normal water level range of 115 cm. The system has been developed in co-operation with JAERI and HRP. Based on the experience with this BICOTH system a new sensor type, called TRICOTH-III, is designed. This sensor has stand-alone capability with the same digital resolution as the eight sensors of the BICOTH system installed in the reactor now. At this moment a patent for the TRICOTH-III continuous water level sensor is applied for. The signal evaluation has been improved by using a micro-processing unit.

The new sensor and the new processor for translating sensor codes to water levels promise more technical reliability and ease in the handling. Moreover the axial temperature transition functions in the TRICOTH-III sensor have been optimised for additional use with the digital codes of the sensor. A very smooth and continuous water level indication can therefore be expected when a TRICOTH-III system is actually applied in a reactor.

Further development of our LWRSIMS model for the core of the Dodewaard reactor concerned improvement of the thermohydraulics. The driving force of the recirculation has been made dependent on the slip in the downcomer and on the condensation length, whereas all relevant parameters have been made pressure dependent. Also comparisons were made between the predictions of our LWRSIMS and the CASMO-code for k_∞ of the Dodewaard fuel elements, resulting in differences up to 3%. These differences refer to the presence of gadolinium in Dodewaard's fuel grid. Mass spectroscopic studies of boric acid in ten water samples from the primary circuit of the Borssele reactor have revealed variations, up to 5%, in the isotope ratio of boron-11 and boron-10.

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REACTOR PHYSICS ACTIVITIES IN SPAIN

COMPILED BY

R. CARO

NUCLEAR PANORAMA

The most important event in the nuclear field last year in Spain has been reaching roughly 40% of nuclear electricity in the over all electric production. After the very recent connection to the grid of the last two nuclear power units, of about 1000 Mwe each, one from Westinghouse -in the Vandellós site- and the other from KWU (in Trillo), the total installed capacity is 7700 Mwe with 10 units. The loading factor has been above 80% in the last 2 years. It is to be reminded that five other nuclear units in a well advanced state of construction were included in a "sine die" moratorium after the General Elections of 1982. However, at present,it is insistently commented that after the next General Elections, the issue will be reconsidered. One of the reasons should be an increase of the electricity demand above 5% in 1988 and above 7% in the first 6 months of 1989.

The two units in Valdecaballeros, both BWR are the better qualifies candidates for their construction to be restarted.

An other highlight has been the aproval by the government of a general plan for radioactive wastes disposal. In addition , the site of El Cabril (medium and low level radioactivity repository) has been given green light by the Consejo de Seguridad Nuclear, (that is the Nuclear Regulatory Body in Spain).

Two experimental nuclear reactors of the old Atomic Energy Commission, at present Ciemat, (Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas), a thermal swinming pool and a fast zero power assembly- are in the process ofbeing dismantled. Two other, of the Argonaut type , in the universities of Bilbao and Barcelona, inaugurated in 1962 , whose operation was stopped more than 15 years ago are being dismantled, too.

In the same chapter of decommissioning, the Andujar Uranium Mill is to be included. This proccess means firstly the stabilization analysis, environmental evaluations aimed at the optimation for the waste dykes and the decommissioning of the instalation itself.

The Junta de Energía Nuclear (that is a kind of Atomic Energy Commission) was transformed into a research center for alternative energies (CIEMAT), nuclear energy being just one of the components.

At present, the R+D nuclear activities (most of them connected with operational safety in nuclear plants) are carried out by the following public and private bodies:

Consejo de Seguridad Nuclear (that is the Nuclear Regulatory Body).
ENRESA (Radiactive Wastes Disposal).
ENUSA (Nuclear Fuel Fabrication).
CIEMAT (R+D for Energy Alternatives)
UNESA (Utilities Consortium)
Universities
Other

Most of the R+D activities are carried out as joint ventures by all or some of the above mentioned bodies, very frequently in collaboration with International Organizations (NEA, IAEA and Euratom), and with other countries through bilateral agreements, namely: France (IPSN) and the US (NRC).

Currently, the most important projects under progress are:

1) LOFT (OECD)

2) TMI-2 (OECD)
3) FEBUS (Core Severely Damaged), France
4) ICAP (International Code Assesment Program) NRC (US)
5) LACE + ACE

A major activity, too, is the carrying out of PRA's that was decided by the CSN in 1986, since it is supposed to be the most valuable tool to assess the safety of NFP's CSN in 1986. So far, one has been completed, three more are under progress and two more have been requested. All of them are level 1. At present an analysis of the IPE type (Individual Plant Examination) is being contemplated by the CSN as possible next step to be requested to NPP's.

In relation with IAEA, one OSART (Operational Safety Analysis Review) was performed last year for the Almaraz nuclear power plant (PWR), and another is scheduled for 1990 for Cofrentes (BWR).

Other activities are: Steam Generator Tube Damage analysis, BWR's stability analysis, Human Factors, Transient and Accident Simulators for training purposes, and a number of minor projects.

To finish this brief description the activities of the Politechnic University in Madrid are to be mentioned. They are mainly concerned with the neutron and radiation transport problems involved in the inertial confinement fusion, and with the fuel management of NPP's.

ENRESA, too, has its own R+D program mainly concerned with wastes characterization, sites selection, environmental impact, systems design and dismantling.

CONSEJO DE SEGURIDAD NUCLEAR

CSN ACTIVITIES IN REACTOR AND PLANT PHYSICS

J. M. Izquierdo et al.

Activities in the area of transient simulation

I The Treta Projet

Description

Treta[1] is a time domain general purpose simulation package, developed at the Nuclear Safety Council (CSN). Emphasis has been placed in a modular structure, quite similar to simulation languages, allowing for a fast implementation of changes in the system configuration and/or settings.

There can be distinguished two parts in Treta:

a) A general driver called Capta able to solve block diagram nets of numerical models, By using Capta, the simulated system, including its topology, is completely described via input data, which allows for progressive enlargements or modifications of the simulated system at the user level.

CAPTA deals with:

- input data management
- time point renewal

- sequential computation of each block of the system, by calling the necessary subroutines of the Block Catalogue, for each time point.
- feedback management, by using an iterative algorithm.
- storage and management of partial and final outputs.

b) A Block Catalogue subdivided in three categories:

- Algorithm library
- Module library
- Component library

which allows the user to define, with different level of detail, the set of operations to be performed in each block of the system.

New features are being added to the Treta system. Specific algorithms for automatic steady state adjustments are under development. Interactive capability and graphic output facilities are expected to be incorporated in the near future.

c) Other important features of Treta are its capabilities for combining other models, much more detailed, of particular components, with the rest of the block diagram, producing unique combinatios. The coupling is made at each time step. Examples of these are:

c1) the recently incorporated option of coupling[2] to the popular COBRA code for the thermohydraulics of the cores (TRETA-COBRA). This will allow the detailed analysis of core and system assymetric behavior.

c2) the ongoing work for dynamic coupling of reliability and dynamics[3], called TRETA-DYLAM. This aspect of the projet is part of the System Response Analizer Project in cooperation with the ISPRA establishment of the European Comunity.

TRETA is not machine dependent. It has been implemented on a variety of computers, including a reduced version for PC.

More details about TRETA were described in the 1987-88 version of this report.

Advances during 1988-89

The most salient features of this period are:

1 Developments in the improvement of the system

i) The coupling between the Colapso Code and Treta has been completed and satisfactorily checked against independent calculations. Colapso acts as an auxiliary code that simplifies the treatment of large linear portions of the block diagram. A new algorithm for time domain analysis of combined linear-non linear networks has been deviced. See ref 4 for details.

ii) The existing single loop PWR version has been enlarged to incorporate[5] one failed loop and a "double" intact loop. The new version is being applied to the study of transients in plants with assymetrically plugged steam generators.

iii) The library of models to be used in the BWR version has been completed to 80% and it is expected to be ready by the end of 1990 the modules of this library were implemented by the department of Nuclear and Chemistry Engineering of the Politechnical University of Valencia. Some examples are documented in ref 6.

iv) The library of models por PWR's to be used in the assessment of the vendor's code for accident simulation (chapter 15 of safety analysis reports) is near completion and is expected to be finished by the end of 1989. Examples are documented in ref 7.

2 Developments in the applications of the system

i) A methodology[8] for the application of TRETA-DYLAM to the safety assessment of software aspects of protection systems has been developed. It includes a conceptual framework with appropiate definitions, acceptance criteria and rules for the utilization of dynamic event trees in evaluating the optimization of a given protection system.

ii) A generic study[9] of the rod with drawal design basis transients for Westhinghouse PWR 3 loop plants has been performed, including cases with and without trip.

iii) Also, we have started a generic study of the influence of steam generator plugging on safety aspects.

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PIN POWER AND BURNUP CALCULATION BY THE SEANAP PWR CORE ANALYSIS SYSTEM

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ABSTRACT

A detailed pin-by-pin power and burnup calculation capability has been included in the SEANAP Core Analysis System. The COBAYA code, that solves the two-group finite-difference neutron diffusion equations for two-dimensional core planes by an iterative local-nodal procedure, has been extended to include at the cell level the effects on the cross sections of the pin burnup, effective fuel temperature, equilibrium xenon concentration, local water density and boron concentration. The quarter-assembly nodal correction factors and the maximum pin-to-node average power factors calculated by COBAYA are given to our three-dimensional PWR core simulator to obtain fast and accurate nodal and local ($F\Delta H$, Fxy and FQ) power distributions along cycle operation and limiting conditions. The validation is done by comparison with actual in-core measurements in several cycles of spanish PWR plants and with the PWR depletion benchmark, with quite good agreement in the pin powers and burnups. Applications to the detailed core reload optimization of PWR's are also discussed.

PIN POWER AND BURNUP CALCULATION IN THE SEANAP PWR CORE ANALYSIS SYSTEM

The SEANAP system (Ref.1) is integrated by three main codes or subsystems: the MARIA subsystem for PWR fuel assembly modelling and calculation, based in an extended version (Ref.2) of the WIMS-D4 code; the COBAYA code (Ref.3) for detailed (pin-by-pin) two-dimensional calculation of PWR core planes; and the SIMULA code(2,3) for three-dimensional nodal (quarters of assemblies) PWR core simulation.

In previous versions(1), the COBAYA code used the macroscopic cross-sections per cell type supplied by the WIMS-MARIA calculations for every fuel assembly class at each core condition along cycle burnup. There the cells of the same type and fuel class had the same cross-sections. The configuration heterogeneities were treated but not the operational ones, due to the different pin burnups, fuel temperatures, xenon concentrations and water densities (and, thus boron concentrations) at the cell level.

In the more recent version, COBAYA-87, the operational heterogeneities at the cell level have been included in the following two main steps:

- The library written by WIMS-MARIA of two-group cross-sections per cell type and fuel assembly class has been extended to include, in addition to the homogeneized macroscopic cross-sections, the microscopic ones for water, boron, xenon and samarium at each burnup step.

- The feedbacks on the cross-sections at each cell are included in COBAYA-87 as a function of the local variables: cell burnup, power density (through fuel temperature and equilibrium xenon) and water density (and boron concentration) from the previous step or iteration.

With the above and several other extensions the computation time of the COBAYA code increased in less than 50%. Now each 1/8 core plane calculation with all feedbacks and critical boron search converges to 10-5 in pin powers in about 150 seconds in a CYBER-835 computer, rated at 0.47 MFLOPS. The code and the whole SEANAP-87 system are being implemented in low-cost engineering workstations with floating-point accelerators of similar performance.

VALIDATION OF THE PIN POWER AND BURNUP CALCULATIONS

The validation of the SEANAP-87 capability for pin power and burnup calculations in PWR cores has included the comparison of results with the PWR depletion benchmark and with in-core measured power distributions along several cycles of 5 spanish PWR units.

The PWR depletion benchmark (Ref.4) is a severe test for pin powers and burnup calculations in typical reload patterns of PWR cores, although it only includes the effects of within-assembly continuous gradients in the microscopic depletion of the (initially homegeneous) fuel and burnable absorber, with constant microscopic two-group cross-sections.

In the SEANAP-87 calculations of this 2-cycle PWR problem, consistently with its standard procedures, each fuel assembly class with different fuel enrichment or burnable absorber loading is depleted independently at its average power density along each cycle with the given boron letdown concentration versus time. At each time step the fundamental mode two-group spectrum is used to integrate the isotopic reaction rates and the isotopic depletion chains are solved by the analytical method in discrete time steps. Thus, an auxiliary ad-hoc code provides the cross-section library in the standard format for COBAYA-87, but using the prescribed microscopic cross-sections. Then, COBAYA interpolates as a function of the pin-cell burnup, includes the equilibrium xenon concentration at each pin-cell and searches for the critical boron concentration at each burnup step along cycle.

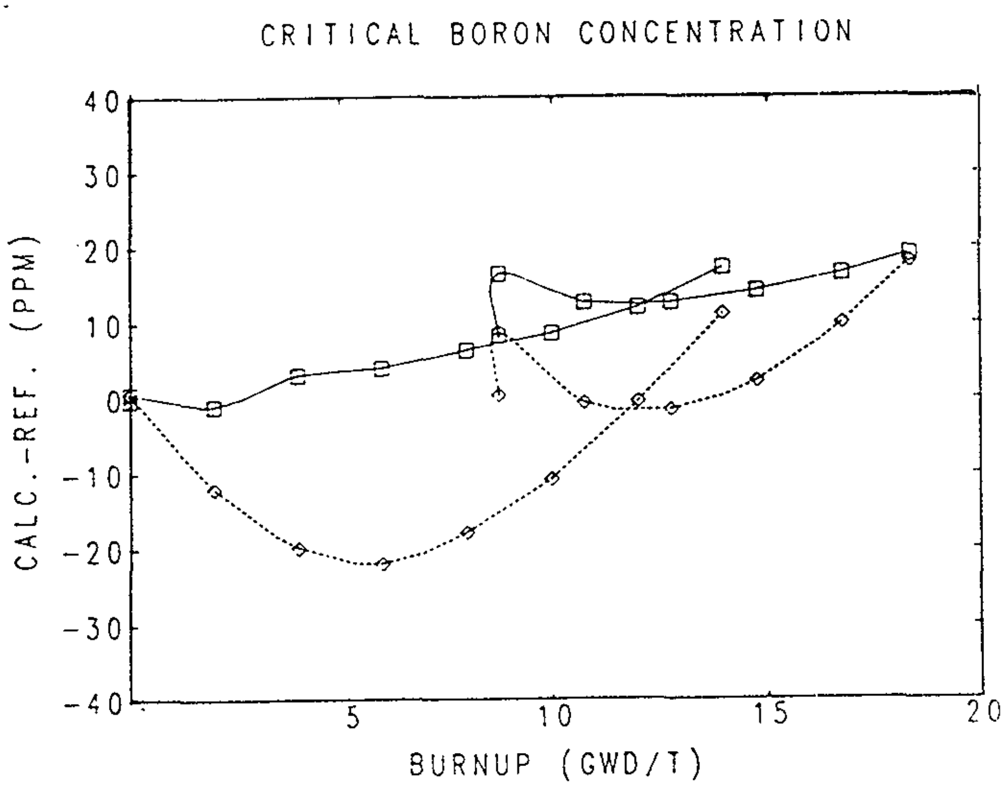

Fig. 2. Differences (ppm) between calculated and reference critical boron concentrations versus average core burnup (GWD/T) along two cycler for two differents cross section libraries: with constant boron (dotted) and with boron history (solid).

In table I are summarized the differences obtained between our results and those given as a reference(4). This table includes also results obtained from a library generates with constant boron concentration, that we will discused later.

The differences between the COBAYA predicted boron letdown and the reference solution for the two cycles are plotted in fig. 2 (solid lines) versus the average core burnup. The maximum difference obtained at EOC-2 is lower than 20 ppm.

It can be seen that the maximum errors in the BOC power distributions are very small (less than 1.0 %), even for the rodded case. Thus, errors introduced in solving the diffusion equations should not be a source of significant error in the burnup calculation. Then, the only source of potential errors is the use of burnup-dependent macroscopic cross sections, as opposed to the explicit depletion of each nuclide in each node of the reactor (reference solution).

CONCLUSIONS

Since the 2D COBAYA-87 solution has the fine-mesh pin-by-pin accuracy, the PWR depletion-benchmark checks the accuracy of the block depletion, with macroscopic cell cross-sections, procedure used in SEANAP-87, where the fuel isotopic depletion is unique for each fuel assembly class, at its average fundamental mode conditions along cycle. The small differences found for this benchmark (table I) show that most probably the microscopic pin-by-pin fuel depletion is not necessary in PWR cores. On other hand this benchmark does not include the intra-assembly heterogeneity effects, due to the discrete configuration of water guides, burnable absorbers and control rods, in the pin burnup distribution. The analysis of measured in-core power distributions by the SEANAP system shows that the COBAYA-87 code includes these effects very accurately and efficiently. This analysis together with the one carried out previously (Ref.1) for other core parameters, qualifies the SEANAP system as the intended support capability for independent in-house analysis of PWR core design, tests, operation and in-core fuel management.

Table I.

Summary of differences obtained in the PWR depletion bechmark

Condition	Boron history				Constant boron			
	BOC-1	EOC-1	BOC-2	EOC-2	BOC-1	EOC-1	BOC-2	EOC-2
	Critical boron concentration (ppm)							
No Xe	-.3	4.0	8.4		-.3	-3.3	.4	
Eq Xe	+.6	+17.4	16.5	19.1	+.6	+11.3	8.8	18.1
	Maximum average assembly power (%)							
Rodded	-.5		-.3		-.5		2.3	
No Xe	-.3		-.9		-.3		.9	
Eq Xe	0.	.7	-1.0	-.3	.0	.0	-.1	.1
	Deviation in average assembly powers (2σ in %)							
Rodded	1.1		1.0		1.1		5.9	
No Xe	.4		1.4		.4		2.0	
Eq Xe	.3	1.2	2.1	1.6	.3	1.0	.9	1.5
	Maximum core pin power (%)							
Eq Xe	-.7		-1.3		-.7		.0	
	Deviation in maximum pin powers per assembly (2σ in %)							
Eq Xe	.4		2.2		.4		1.2	
	Deviation in average assembly burnups (2σ in %)							
	.4		.5		.6		.8	
	Deviation in maximum pin burnups per assembly (2σ in %)							
	.4		.6		.7		.9	

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ISOTOPIC DEPLETION OF SOLUBLE BORON IN A PWR

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The purpose of this work was to determine the isotopic depletion of the soluble boron in the primary of a PWR along cycle operation, under the limiting condition of continuous boron dilution, without feed of fresh boron, that maximizes the boron isotopic depletion effect. Presented here are the results for a PWR cycle, that has been operated close to these continuous dilution conditions at rated power along most of the cycle.

In the continuous boron dilution model the boron concentration letdown curve B(t), i.e. in parts per million of total boron weight along time, is achieved by adjusting the mass flow feed of clean water $G_f(t)$, equal to the mass flow bleed of borated water from the primary, with a constant total water mass inventory M_p. The balance equation for the total boron mass in the primary is given by

$$\frac{d}{dt}\, B(t)\, M_p \;=\; -\, B(t)\, G_f(t) \tag{1}$$

Hence the integrated feed mass of clean water required to reduce the boron concentration from B(0) to B(t) is given by

$$M_f(t) \;=\; \int_0^t G_f(t)\, dt \;=\; M_p\, \ell n\, \frac{B(0)}{B(t)} \tag{2}$$

To determine the isotopic abundance in ^{10}B of the soluble boron in the primary, e(t) in weight, the balance equation for the total ^{10}B mass in the primary should include the neutron absorptions integrated in the whole core and surrounding irradiation volumes, as given by

$$\frac{d}{dt}\, e(t)\, B(t)\, M_p \;=\; -\, e(t)\, B(t)\, G_f(t) \,-\, e(t)\, B(t)\, M_c < \sigma_a^{^{10}B}\, \Phi > \tag{3}$$

obtaining:

$$e(t) = e(0)\, exp\left\{ - \frac{M_c}{M_p} \int_0^t <\sigma_a^{10_B} \phi> dt \right\} \tag{4}$$

where M_c is the total mass of water in the active core volume and $<\sigma_a^{10B} \phi>$ is the average ^{10}B absortion probability in the active core per unit time. The ratio of core to primary water mass, M_c/M_p, about 6% in the C.N.Almaraz PWR, is essential to avoid a very quick ^{10}B depletion in the soluble boron, since the absorption probability is higher than in lumped burnable absorbers with spatial self-shielding by flux depression.

The calculation of the ^{10}B absorption probability per unit time has been introduced in our COBAYA-87 code [Refs. 1,2] of detailed (pin-by-pin) 2-group diffusion calculation of PWR X-Y average core planes, using the following cell-by-cell accumulation of ^{10}B absorptions:

$$<\sigma_a^{10_B} \phi> = \frac{\sum\limits_{i \varepsilon V_r} V_i N_i^{10_B} \sum\limits_{g} \sigma_{ai,g}^{10_B} \Phi_{i,g}}{\sum\limits_{i \varepsilon V_c} V_i N_i^{10_B}} \tag{5}$$

in terms of the cell volumes, ^{10}B number densities and microscopic group cross sections and fluxes per cell i and group g. The sum in the numerator extends over all active core and reflector cells, while that in the denominator is restricted to the active core volume, including all fuel and water cells.

The PWR cycle designed for a length very close to 1 year at rated power, with a low-leakage loading pattern using a moderate number of lumped burnable absorber wet tubes (WABA's), the calculated core-average ^{10}B absorption probability is about 3.5-4.0 per year from beginning to end of cycle. This results, within the continuous boron dilution model and the above mentioned data, in an almost linear reduction of the ^{10}B isotopic abundance in the primary system along the cycle that reaches about 80 percent of the initial abundance after 1 year of effective length at full power.

The inverse of the ^{10}B remaining fraction is the correction factor to be applied to the calculated critical boron concentration letdown curve along cycle burnup, with a constant (natural) ^{10}B isotopic abundance.

The limiting continuous boron dilution model with the ^{10}B depletion calculations based in our COBAYA code is in quite good agreement with the measured boron concentrations for this cycle, explaining very well the diferences found with the expected design letdown curve, that increased up to +60 ppm at middle of cycle when this work was done, motivated by the worry of the reactor operation staff.

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REACTIVITY EFFECTS OF FISSION PRODUCT DECAY IN PWR

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The purpose of this work was to analize the effects of the fission product chains with radioactive decay on the reactivity in PWR cores, calculating their accumulation and absorption rates along fuel burnup at continuous operation and after shutdown periods extending from one day to a few months.

Our PWR version[1] of the WIMS-D4 code[2] is used in first place to obtain the individual number densities, absorption rates and averaged cross sections for every nuclide of the fission product chains with significant decay rates, as a function of fuel burnup at continuous irradiation. Next by an auxiliary ad-hoc code we have processed this data, together with the required one for the fissile nuclides and boron, also taken from WIMS at each burnup step, to calculate the average or effective values relevant for the analysis and the decay and change in overall absorption after several shutdown times.

Table I includes a list of selected data for the four fission product chains with significant radioactive decay included in WIMS, where the averaged values are taken from one of the calculated burnup steps and PWR fuel types considered in the survey applications to be described later.

The ^{135}Xe chain is included first in Table I for reference purposes only. This transient chain is the most important and is always treated individually and in detail by the reactor analysis codes. Henceforth it will not be considered in the following analysis, focused on the other transient chains.

The ^{149}Sm chain is next in importance. The WIMS-D library includes the five main absorbing nuclides given in Table I, while ^{149}Nd and ^{149}Pm are assumed to decay inmediately to ^{149}Sm. In our auxiliary code these nuclides are reconstructed from the data at each burnup step. In many other cell burnup codes the ^{147}Pm via to ^{149}Sm is not included. The ^{147}Pm has an effective period (decay halflife plus absorption probability $<\sigma_a.\phi>$) of about one year and a fission yield nearly twice that of ^{149}Nd. Hence after one year of burnup this via increases the ^{149}Pm and ^{149}Sm concentrations in about 50%, and nearly doubles them at the fuel discharge burnup. This also increases the so-called "Samarium peak" of increased absorption in the ^{149}Sm produced by decay of the balance ^{149}Pm concentration existing at power, after several days of reactor shutdown.

The last column in Table I has the relative absorption values for each of those decaying nuclides, expressed in equivalent boron concentration (ppm). In this case the increase in absorption after complete decay of the ^{149}Pm produced via ^{149}Nd, the only one taken into account by many authors, that is equivalent to about 35 ppm of boron, should be increased by the buildup via ^{147}Pm (20 ppm) and by the decay of ^{103}Ru to ^{103}Rh (14 ppm), but is decreased by the decay of the absorbing nuclides ^{148m}Pm (35 ppm), ^{148}Pm (12 ppm) and ^{105}Rh (16ppm), resulting in an overall absorption increase of only 6 ppm after complete decay.

From the resulting absorption increase, in ppm of equivalent boron, versus the shutdown times, from 1 to 100 days, when the fuel burnup has reached 20 GWd/t, for different PWR fuel assembly types of Westinghouse 17x17 design.

The following conclusions may be extracted:

1. The reactivity effect of fission product decay (excluding ^{135}Xe) changes significantly with the shutdown time. The maximum absorption increase by decay is reached for about 10 days shutdown.

2. The dependence with fuel type, enrichment and burnup (after 10 GWd/t) is slight. But the change with previous power density is nearly linear, which might be significant after coastdown in previous cycles.

3. For long shutdown periods, the overall reactivity effect of decay in the three fission product chains considered is much less than if only the Samarium peak due to ^{149}Nd is considered.

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TABLE I

DATA OF FISSION PRODUCT CHAINS WITH DECAY IN WIMS-D4

Nuclide	Fission yield (%) ^{235}U; ^{239}Pu	Decay Period	σ_a (b)(1)	Effective Period(1)	Absorption in equivalent boron (ppm)(1)
^{135}I	6.02; 5.88	6.7 h	–	6.7 h	–
^{135}Xe	0.56; 1.25	9.2 h	158000.	2.6 h	325
^{147}Pm	2.15; 2.0	2.65 a	79.	235 d	91
^{147}Sm	β decay ^{147}Pm	–	24.5	2.7 a	6
^{148m}Pm	47% σ_a ^{147}Pm	42 d	2367.	8.3 d	35
^{148}Pm	53% σ_a ^{147}Pm	5 3 d	1474.	4.0 d	12
^{149}Nd	1.02; 1.3	2 h	–	2 h	
^{149}Pm	σ_a $^{148m+148}Pm$ β decay ^{149}Nd	2.3 d	–	2.3 d	(20)(2) (35)(2)
^{149}Sm	β decay ^{149}Pm	–	4630.	5.3 d	130
^{105}Rh	0.85; 5.5	35 h	1593.	32 h	16
^{105}Pd	β decay ^{105}Rh	–	2.5	27 a	5
^{103}Ru	2.9; 5.6	40 d	2.0	40 d	1,(14)(2)
^{103}Rh	β decay ^{103}Ru	–	32.	2.1 a	90

(1) Averaged at 20 GWd/t burnup in a standard PWR fuel assembly with 4.0 w/0 initial enrichment.

(2) After complete decay to ^{149}Sm or ^{103}Rh, respectively.

NEUTRON DAMAGE IN FUSION REACTORS: DISPLACEMENTS AND TRANSMUTATIONS

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1. INTRODUCTION

The objective of the present paper is to calculate irradiation damage behaviour, in terms of displacement and transmutant production in inertial and magnetic fusion reactors. For purposes of planning irradiation experiments in fission reactors, calculations have been also carried out for the peripheral target position (PTP) in the High Flux Isotope Reactor (HFIR). The material included in the calculations is the reduced activation martensitic steel, B-TAHF, based on the specification of 1.4914 steel. Results of this work would be useful to gain insight into the behaviour in different reactor confinement fusion concepts and to guide and assess efforts in fission-fusion correlations. The nominal composition of the alloy B-TAHF (Kfk-CIEMAT) and the considered neutron spectra are given in Ref. 1.

2. DISPLACEMENT DAMAGE

The calculational model2 considers two sources in the production of displacements. Displacements produced by neutron reactions in elements initially present in the alloy (primary reactions), and displacements caused by reactions in nuclides generated by transmutation. Calculations have been done using a displacement function2 which represents the number of stable defects (Frenkel-pairs) of a displacement cascade created at low temperatures by a primary knock-on atom.

For the alloy B-TAHF, displacement production in the three environments caused by the no-primary reactions has been found to be negligible. So, displacement rate can be considered constant, and it can be calculated from the displacement cross sections corresponding to primary reactions.

Spectral-average displacement cross sections of B-TAHF for CCTR-II, HIBALL-II and HFIR reactors are 682, 119 and 196 barns, respectively. The displacement rate in MCF, ICF and HFIR reactors are 40, 3.75, and 31.5 dpa's/year. In HFIR, a 14-year irradiation time is neccesary to obtain a dose approaching to 400 dpa, which at least must be withstanded by the first wall of a magnetic fusion power reactor for economic feasibility. In the FCI reactor, for a lifetime of 30 full-power years the production of displacements is around 112 dpa. These results are very similar to those of 1.4914 steel1.

3. TRANSMUTATION PRODUCTS

The calculational procedure for the transmutation problem is based on the concept of "generating function"2-5. The element i generating function, pi1, gives the cumulative number of atoms of element i produced per initial atom of element 1 as a function of time.

Figures 1, 2 and 3 show transmutation products generated in B-TAHF when exposed to neutron environments of the MCF, HFIR and ICF reactors, respectively.

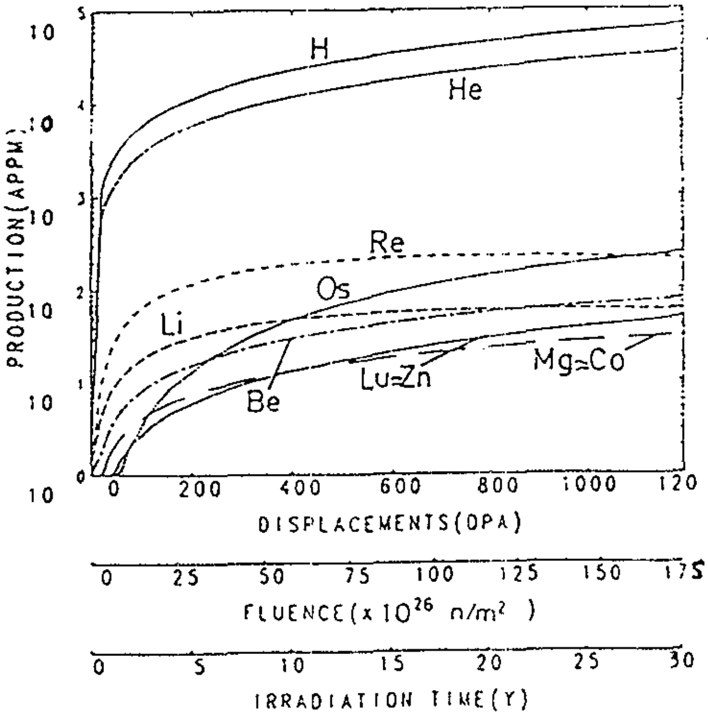

FIGURE 1
Principal elements produced in B-TAHF for the magnetic fusion reactor concept.

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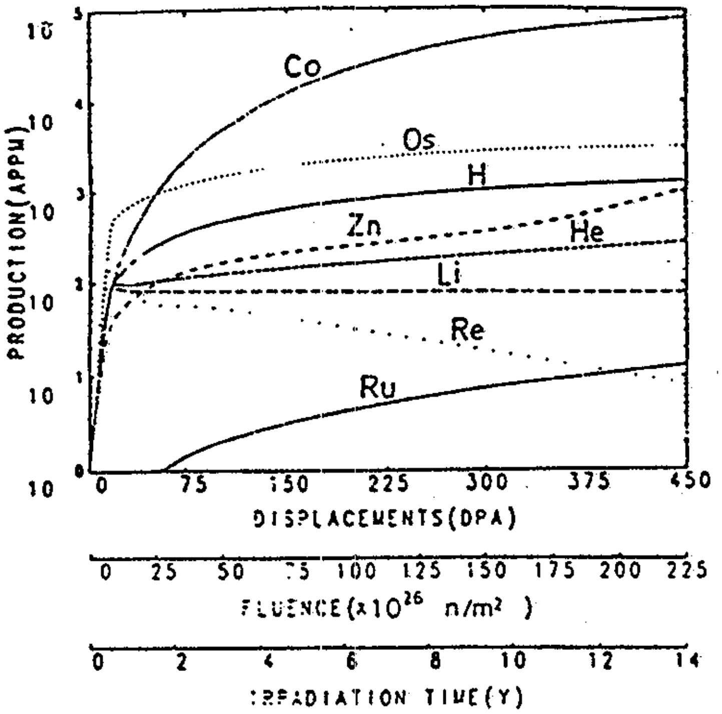

FIGURE 2
Principal elements produced in B-TAHF for the HFIR reactor.

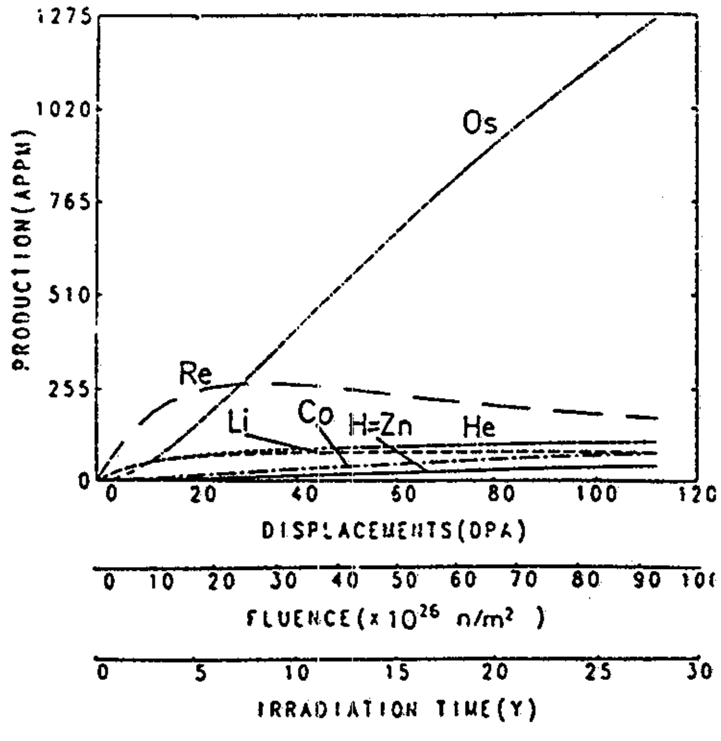

FIGURE 3
Principal elements produced in B-TAHF for the inertial confinement fusion reactor.

Atomic relative changes in the population of initial constituents of B-TAHF steel for MCF and HFIR are given in Ref. 1. Figure 4 represents the results for ICF reactor.

It is significant to note that in the ICF reactor, changes occurring in C, N, Si, P, S, Ti, Cr and Fe elements are less than 1% (in atoms) and less than 2% for vanadium.

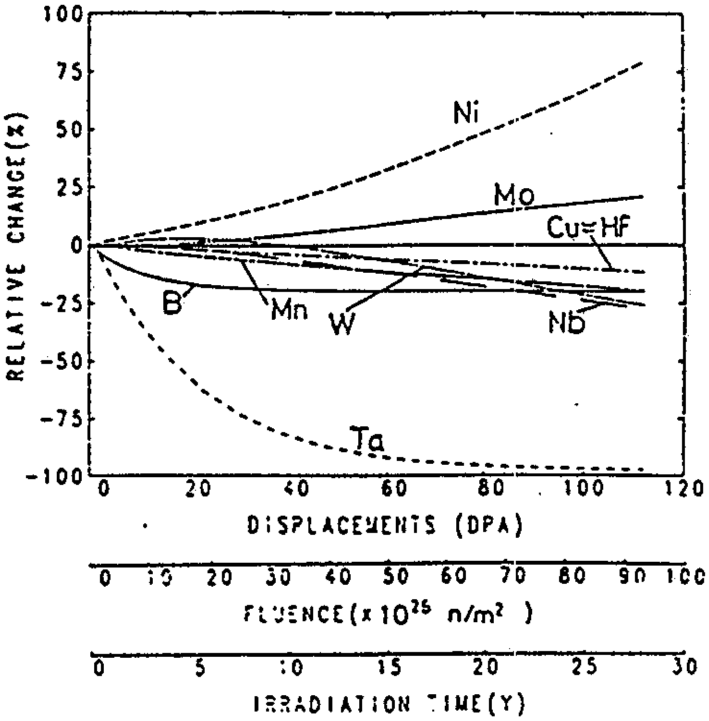

Figure 4
Atomic relative changes (%) in the population of constituents of B-TAHF for the ICF reactor.

4. DISCUSSION OF RESULTS AND CONCLUSIONS

The displacement production rate (dpa/y) in the FSW of HIBALL-II reactor is approximately an order of magnitude lower than that of both HFIR, and the MCF reactors. The MCF reactor has a dpa rate slightly higher than that of HFIR. No additional contribution to the displacement rate from no primary reaction has been found.

The evolution of the elements initially present in B-TAHF as well as the generation of new products is strongly dependent on the neutron spectrum. There are significant changes in the composition of the steel irradiated in both the MCF, and the HFIR reactors. At the first structural wall of the ICF reactor, HIBALL-II, changes are much less pronounced than in the other two cases. In this case relevant changes occur only in Ta, Ni, Os and Re elements. The production of He is small, but should be taken into account because of the great damaging effects of this element

In martensitic steels, the helium production is an increasing function of average spectrum energy, mainly because of the iron effect. In the CCTRII spectrum the generation of helium is linear with fluence, while in reactors containing appreciable thermal neutron fluxes, nonlinear increases in helium production take place due to multi-step reactions and burnup of nickel, boron and a few other elements.

Because of transmutation differences, and similar displacement production rate in fission and fusion reactors, it is very unlikely that reliable engineering data for MCF reactors can be obtained in a reasonable time in HFIR and also in fast reactors. On the other hand, in fission reactors it appears to be possible to obtain displacement damage and transmutation levels very similar to those of inertial fusion reactors, within a reasonable irradiation time. Therefore, we may conclude that fission facilities provides a means of testing several properties of developmental alloys, in integrated damage regimes of engineering interest for fusion inertial reactor applications. The major question that remains unknown in using fission irradiation data for FCI reactor design is that related with pulsed damage effects at high neutron fluence arising from the large instantaneous damage rates of FCI facilities.

The alloy B-TAHF, as well as all the materials that are at present available, does not withstand the neutron damage of FCM reactors for irradiation times that will fullfil the economic requirements. However, it appears that B-TAHF alloy as FSW of the FCI reactor will be able to last for the power plant's entire 30-year lifetime without being replaced.

ACKNOWLEDGEMENTS

The calculational models used in this work have been developed under a research project supported by PIE-UNESA. The applications here presented are included in the cooperation agreement between DENIM and the Spanish Nuclear Research Center (CIEMAT), as a part of the general KfK/CIEMAT/EEC research project, entitle "Development of Low Activation Ferritic-Martensitic Steels.

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REACTOR PHYSICS CALCULATIONS IN HIDROELECTRICA ESPAÑOLA

Carmen Cabezudo
Andrés Galicia

Hidroeléctrica Española (HE) is a Spanish electrical utility which presently owns and operates a BWR (Cofrentes Nuclear Power Plant) and shares the ownership of several PWRs. Since 1985, HE is involved in the development of an in-house nuclear analysis capability, being presently able to perform both reactor physics and thermalhydraulics calculations. Codes selected for such a methodology are CASMO/SIMULATE package from Studsvik and RETRAN, respectively.

This paper is to report on the principal analyses performed recently in the reactor physics area for licensing, operational support or fuel cycle optimization purposes.

1. Criticality Safety Evaluation for Spent Fuel Storage Racks

According to the Energy Utilization Plan for C.N. Cofrentes, in the near future high enriched fuel bundles will be introduced into the core in order to achieve 18-month cycles. That policy requires an update of the criticality Safety Evaluation for spent fuel storage racks in order to know the maximum enrichment allowable without modifying the current design.

The design goal of a fuel storage rack, from a criticality safety evaluation point of view, is to provide underwater storage for spent fuel while maintaining Keff < 0.95, including an allowance for biases and uncertainties for both normal and abnormal conditions :

Current licensing analysis shows that the C.N. Cofrentes racks are able to safely accommodate fuel assemblies up to 3 wt % U-235 enrichment, in a 8 x 8 pins array with two water rods. However, this calculation provides a very conservative result because Gadolinia rods were not considered and then the maximum reactivity condition was overestimated for fresh fuel assemblies. In the update analysis, the real Gadolinium content has been taken into account, providing a more realistic description of bundle reactivity, but requiring additional calculations in order to obtain the maximum reactivity value as a function of burnup, and an evaluation of uncertainties associated with the operation history of the bundle while accumulating such an exposure.

A special feature was used in the analysis, all rods were homogenized at average enrichment of the original lattice, in order to know the sensitivity of this type of calculation in comparison with lattices reported in the Gestar. The knowledge of the differences between both types of calculations, allows, with the introduction of a possible uncertainty due to homogenization, to continue with the parametric study increasing the enrichment above the designed lattices. Other uncertainties taken into account were due to manufacture tolerance, statistical and calculational methods, the uncertainties due to burnup, layout and number of Gd rods and the uncertainties associated with the operational history mentioned previously.

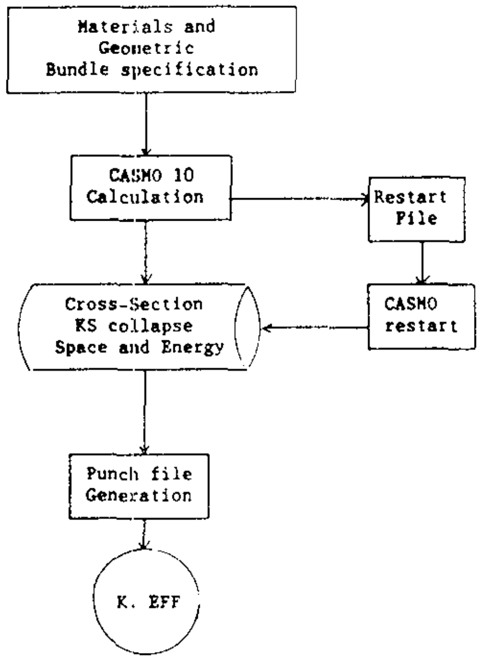
The calculation was made in three parts :

1. Several cases of CASMO, including biases and uncertainties, were performed for three fuel assemblies existing in the GESTAR report.

2. The same cases were performed but with all pins of the lattice at average enrichment of the original lattice. From the comparison of K-inf, including biases and uncertainties, between both calculations, we fixed the final uncertainty due to the homogenization procedure.

3. Parametric study was made increasing the average enrichment of the lattices up to 6.3 wt % U-235.

A reduced flow chart of the methodology used with CASMO-2E is shown in the figure.



The abnormal conditions had also been considered in the analysis.

In abnormal cooling conditions, the coolant temperature could increase.

This effect was studied with CASMO using the rack conditions and varying the voids fraction from 0 up to 100%. The results obtained showing a maximum of reactivity at low density (around 0.3-0.4 g/cc water density)

The results obtained in the criticality evaluation demonstrate the possibility of introducing high enrichment fuel assemblies, up to 43.8 wt % U-235, without any requirement to modify the current design. Although the realistic approach to the Gadolinium content, enough conservative assumptions are still present through the uncertainties evaluation, assuring enough margin to safety criteria.

2. <u>Fuel bundle redesign</u>

C.N. Cofrentes is presently running in 15-month transition cycles, from the past 12 months to the future 18 months. In order to achieve such upgrade in energy, the average reload enrichment has been increased by using the maximum enriched bundle allowable in the GESTAR catalogue. Unfortunately, this type of bundle selected for transition cycles, has a strong content of Gadolinia (8 rods at 3% wt), providing a good performance at the medium and end of cycle, but showing poor behaviour at the beginning of cycle from the reactivity state point. So, enough hot excess reactivity could not be obtained, preventing full power being reached until Gadolinium starts to disappear by burning effect.

By means of the Reactor Physics methodology, a redesign of the fuel bundle took place. Gadolinium rods were reduced from 8 to 6, while two Uranium rods were introduced into the lattice. A short pins shuffling was also required in order to keep the local peaking factor as low as possible. In fact, the final designed bundle shows an improved reactivity behaviour at the beginning of cycle (solving the hot excess reactivity problem), but also provides a reduced local peaking factor which allows more margin to linear heat generation rate limit, and consequently, a more flexible core operation.

When Gadolinium drops to residual values (medium of cycle), the reactivity of new and old designed bundles is exactly the same because the average enrichment has been maintained, and then no impact has been introduced in the areas of previous good performance.

The reactivity improvement obtained at the beginning of the cycle has been around 1% Δ K while the reduction in local peaking factor is around 7%, with only 10 rods shuffled from its original position. These bundles are now under manufacturing process and will be loaded next summer.

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REACTOR PHYSICS ACTIVITIES IN SWEDEN
October 1988 - September 1989

Compiled by Klas Jirlow, Studsvik Nuclear

1 General about energy supply and nuclear power in Sweden

The electric power consumption in Sweden was in 1988 about 130 TWh of which industry used 55 TWh, electrical heating of private houses 30 TWh and the rest of 45 TWh was used for domestic purposes, services and in transportation. The total amount has not increased significantly from 1987 to 1988 being an exception from the long term growth of about 4 % per year. This is due to a warm winter 1988 causing a decrease of the electrical heating - a decrease which has compensated a 6 % increase of the consumption in industry.

Almost half of this electricity was produced by the 12 nuclear reactors operating with an average availability of 87.6 %. The number of reactor scrams was 33 for all the reactors together.

All reactors have been equipped with filtered containment venting systems during 1988, except the reactors in Barsebäck, which already had such systems.

The steam generators in Ringhals 2 were replaced during an extended summer outage (3.5 months) this year and this reactor is now running at full power since then. The steam generator leaks in the Ringhals PWRs (Westinghouse) have been the only serious and costly problems up to now.

A less serious problem appeared this winter at the Oskarshamn nuclear power plant. An increased activity in the off gas system occurred during January-February in the Oskarshamn 2 reactor. During the refuelling outage, four damaged fuel assemblies were found. The corner rods had primary damage due to dry-out and secondary damages further out on the rods caused by hydration. The cause of the dry-out is considered to be the water gap between different fuel types, SVEA-64 and AA-8x8, in the same super cell and channel bow. An increased dry-out margin is applied until the problem has been solved.

The average total production cost in 1988 for the 12 nuclear units was 17 öre (2.6 cents) per kWh. This cost includes the "deposit" set aside for waste and decommissioning of 2 öre per kWh.

Depending on the year of commissioning the cost from unit to unit varies from 14 öre (2.2 cents) to 30 öre (4.6 cents) per kWh. The production costs excluding capital costs but including fuel and back-end costs are almost the same at all units and were in 1988 about 9 öre (1.4

cents)per kWh. This sets nuclear power in a very good position by almost any comparison.

In contrast to the firm intention of the socialist government to phase out all nuclear power in Sweden starting with one unit in 1995 and a second in 1996 for safety reasons, the government has authorized after approval from the nuclear safety authorities the utilities to increase the electric output of Oskarshamn 3 and Forsmark 3 from 1050 MWe to 1200 MWe and the Ringhals 1 from 750 to 820 MWe. This implies an increase of nuclear power generation by 2.5 %.

During recent months the public discussion has been very intense and has shown that the ruling Social Democratic Party is deeply divides on the nuclear phase-out issue. Up to now the government seem to be firmly decided to carry through the decisions taken by the Parliament during recent years:

- the phasing out of nuclear power during the next two decades.
- the total release of CO_2 should not increase
- no more large-scale construction of hydroelectric plants is permitted.

Severe attacks questioning the practicability of this official energy policy is being delivered by leading economists and industrialists as well as the leaders of the main labour unions. They all insist unanimously that these three bounding decisions are incompatible and that large decreases in electricity consumption cannot be achieved without disastrous effects on the Swedish economy.

ABB Atom continues to develop new reactor concepts for the international market. In cooperation with the Finish utilities, ABB Atom has developed the BWR90, the BWR for the next century.

As an evolutionary alternative, the inherently safe PIUS-reactor is being developed. Development work leading up to the present PIUS concept has now been going on for more than a decade. As time has passed and the work has progressed the need for a reactor with its "built-in", immutable safety characteristics has become increasingly evident. International marketing of the PIUS concept is now under way. In Italy, where utilization of conventional LWRs have been ruled out as a result of the Chernobyl accident, the state utility ENEL has shown an increasing interest in PIUS. In the US, an agreement on a joint venture has been made with the architect engineering company United Engineers & Constructors of Philadelphia, PA for working out a detailed plant design, getting a US NRC design certification, and for marketing.

2 Status and development of reactor physics at Studsvik Nuclear.

2.1 LWR In-Core Fuel Management (K Ekberg)

The development work on computer codes and methods for LWR ICFM is continuing at STUDSVIK NUCLEAR and its subsidiary STUDSVIK of America. During the year since the previous report was submitted several new companies have joined the group of STUDSVIK ICFM code users, among them some prominent nuclear fuel vendors.

As before work has been devoted both to the area of benchmark and validation and to improvement of the Code Package itself. Due to the large number of users, now exceeding 35, a major benchmark and validation effort is now carried out by the users. The appended list of references is a selection of papers that have appeared since the previous report.

During the past year the development work has been directed partly on consolidation, partly on studies for long-term new operations. A major new option in SIMULATE was released, the Pin Power Reconstruction for PWR. Using this option it is possible to obtain a complete three-dimensional pin power distribution with the advanced nodal code SIMULATE with an accuracy at least as good as that of the traditional fine-mesh diffusion methods, but with the consumption of just a small fraction of work time and computer resources.

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2.2 Transport theory studies on LWR Lattices

2.2.1 Subassembly or cell calculations. (H. Häggblom)

The Monte Carlo code VIM is being used for checking results obtained with the LWR fuel assembly code CASMO and with the LWR fuel assembly code CASMO and with approximate methods for handling heterogeneous pin cells. Calculations made on internal reaction rates in a fuel pin have shown the peaking of the thermal flux in the neighbourhood of the rod surface. The flux and power distributions in PWR and BWR assemblies and the effect of burnable absorbers have been investigated. The agreement between VIM and CASMO results was good, especially if 2-dimensional CASMO calculations were made in 12 energy groups and the mesh size was half of the pitch of the fuel lattice. The normal mesh size is equal to the pitch.

2.2.2 Fuel pin calculations (E. B. Jonsson)

The integral transport code MICBURN is used for calculation of the radial power distribution over the pin at different burnup. The MICBURN power distribution in fresh fuel pins has been compared with distributions calculated with VIM using point cross section data. VIM gives higher flux depression in the rod.

A difficulty with the MICBURN calculations for increasing burnups is to distribute the resonance integral of U238 over the pin cross section. One way of doing this is to use the old measurements of Hellstrand (1958) for uranium metal rods and extrapolate to the relatively thin oxide pins. Using then this extrapolated U238 capture radial shape the power distribution as function of burnup is calculated by MICBURN. The radial temperature distributions are calculated with the thermo-mechanical fuel pin code INTERPIN and it has been found that the fuel centre temperature tends to decrease with burnup relative to the case with a fixed radial power distribution. Typically a decrease of about 50°C in centre temperature is calculated for a BWR rod at 50 MWd/kg.

2.3 CASMO Applications on normal and tight LWR lattices (E Johansson)

The CASMO-3 code, combined with its current 70-group data library (the J-lib) has been tested in calculations on critical experiments for normal LWR lattices. Most of the experiments used in these tests had been carried out in the Studsvik KRITZ reactor. This reactor, now dismantled, could be operated up to 245°C. In addition to uranium some of the KRITZ cores also contained plutonium. A few room temperature uranium cores from a series of measurements at Babcock & Wilcox were also analysed. The outcome of the tests is as generally quite good, fully satisfactory for application of the method to LWRs. In a modified library version (the K-lib) the plutonium isotopes were represented with JEF-1 data as well as with an extended resonance self-shielding treatment. Tests against plutonium experiments gave, however, almost the same outcome as did the J-lib, both for k_{eff} and for the fission rate distribution.

The CASMO-3 code, combined with the K-lib, has been applied in a continuation of the NEACRP benchmark study for High Conversion Light Water Reactors. The calculations concerned refer both to the PROTEUS phase-1 experiments and to power reactor cases including also higher burnup states.

The outcome of the CASMO application in this benchmark study is satisfactory in many respects but there are also a few discrepancies. For instance, the fission rate in U238, important in HCLWRs, is overestimated by 4 to 5 %.

The main initial fuel for tight lattice reactors studied so far is plutonium obtained from discharge fuel from today's LWRs. This is, of course, a natural choice but, unfortunately, it will lead to void reactivity problems especially upon repeated recycling. As an alternative the initial fuel could instead consist of enriched uranium. This alternative gives a less unfavourable void reactivity situation than does the main alternative, as we have shown in previous studies. Apart from the void reactivity improvement in the enriched uranium case this alternative is also of interest for the plutonium production per se. A few calculations in that context have been carried out recently.

3 Status in development of reactor physics methods for utility applications in Sweden

(T Lefvert, Swedish State Power Board)

3.1 Lattice Calculations

Swedish utilities use CASMO for calculations in fuel bundle geometry. Some recent developments are:

- Gamma production and transport to simulate the response of e.g. gamma TIP detectors and gamma thermometers
- Improvement in modelling the heterogeneity effects of lumped burnable absorbers
- Adaption to provide "discontinuity factors" from surface fluxes and currents

In the near future we probably need to model lattices with varying pitch and development should start during 1990 to make this possible.

3.2 Core Calculations

Systematic comparisons of precalculated and measured reactor power distributions have shown that the present BWR core simulation, Core Master POLCA, could probably be improved with regard to modelling the control rod effect. Also, the thermohydraulic part of the program is presently under study to try to improve the models for void formation and void feedback on the neutronics. This part of the BWR simulator has been put in focus after it was shown that the error in calculated local power was not significantly changed when improving the neutron model from a typical 1.5 group Börresen approach to the modern full two group, discontinuity factor method. The latter approach has proven to be very accurate in calculation of power distributions in PWR's where, of course, the thermohydraulic feedback is weak. We have chosen the SIMULATE-3P code, having these characteristic, for our PWR applications. The difficulty of modelling the correct BWR core geometry (channel bow) is presently discussed by BWR operators and suppliers as well as safety authorities. A pin-wise BWR simulation would be useful in doing parametric sensitivity calculations.

3.3 In-Core Fuel Management

For BWR's we presently use the Core Master system from ABB-Atom to handle the various tasks in ICFM. This system does not, however, make efficient use of the computer hardware and software now available. Therefore, we start development during 1989 of an improved system, keeping the basic tools CASMO, POLCA SIMULATE etc but doing ICFM through an interactive system based on modern graphical and data base software on work station computers under UNIX. The development work is done jointly by utilities and ABB-Atom.

4 BWR stability measurements at Barsebäck unit 1 on May 5 and 6, 1988 (H Johansson, Sydkraft)

The measurements regarding reactor stability were performed at Barsebäck unit 1 at full power as will as minimum core flow.

Previous measurements in other a BWR units have led to the conclusion that a mixed core (8x8 and 9x9 fuel mixed) should be less stable than a full core of either type.

The measurements were performed in a full 8x8 core. After next outage (1988) the core will also contain 9x9 fuel. The analysis of the full core stability measurement is therefore to be regarded as a reference for later mixed core measurements.

The tests were performed at full power and after core flow reduction. The measurements were made towards the end of cycle 11 right before coast down started.

Assigning a mathematic model to the power signals by means of process identification a decay ratio (Dr) has been calculated. The Dr expresses the stability margin and shows how fast (Dr = 0) or slow (Dr = 1.0) a disturbance is damped.

At 57 % of nominal power and core flow 2530 kg/s the analysis showed a Dr of 0.2 which indicates a very stable core.

Reference: Studsvik Report; Studsvik/NI-88/40.

REACTOR PHYSICS ACTIVITIES IN SWITZERLAND

Compiled by P. Wydler

1 General Background

In the past twelve months the Swiss nuclear energy scene has been dominated by the aftermath of the cancellation of the Kaiseraugst nuclear power plant project. After difficult negotiations, the Government and the electricity companies involved agreed on a compensation of 350 million Swiss Francs, corresponding to about a quarter of the accumulated costs of the project. According to the initial plans, construction of this power plant should have started already in 1968, but the project has been delayed due to technical modifications concerning the cooling system and the station output, increasing public opposition, new legislation, and two anti-nuclear initiatives which were rejected in 1979 and 1984 with very small margins.

In Switzerland the nuclear age started with considerable enthusiasm: Twenty years ago, after a construction time of only 4 years, Beznau I, a 350 MWe Westinghouse PWR went into commercial operation. The other nuclear power plants followed in 1971 (Beznau II), 1972 (Mühleberg), 1979 (Gösgen), and 1984 (Leibstadt). The five units have a good performance record, illustrated by the average load factor of 84% in 1988, and contribute 36% to the total electricity production of the country.

The turn of the tide has now come with Kaiseraugst: With two new anti-nuclear initiatives pending and recent opinion polls not indicating any changes in the (on the whole negative) public attitude towards nuclear energy, it is unlikely that a new nuclear power plant project can be launched before the turn of the century (In spite of the already existing SO_2 problem and the much debated CO_2 problem, 67% of the population think that the dependence on nuclear energy should be reduced in favour of an increased oil dependence). Since the electricity consumption is further increasing – in 1988 the growth was 1.7% –, more and more electricity has to be imported from France.

After the Kaiseraugst débâcle the public attention has shifted to the incident with the sodium-cooled fuel store of Superphénix. An initially rather small but active group of opponents at Geneva and in its neighbourhood has repeatedly intervened at different administrative levels in Switzerland and France to prevent the French authorities from allowing the reactor to resume its operation. The group has managed to get the support of several well known environmentalist organisations and the Government of Geneva. The events have led to direct discussions between the concerned ministres and the creation of a mixed French-Swiss reactor safety commission which exchanges information on nuclear power stations located less than 100 km from the common boarder (This applies to all Swiss nuclear power stations and the French nuclear power stations at Bugey, Creys-Malville and Fessenheim).

In Switzerland, nuclear engineering is concentrated mainly at the Paul Scherrer Institute (PSI) at Würenlingen. According to the current research concept, there are two high priority programmes, a light water reactor safety and a waste disposal programme. The principal motivations for these programmes are to contribute to the safe operation of the existing nuclear power plants and support the national cooperative for the storage of radioactive waste (NAGRA) in the development of a basic concept for a high-level waste repository in Switzerland. Reactor development is concentrated on gas-cooled reactors and on small district heating reactors. The necessity to provide services to other organisations, the commitment of resources to the longer-term research and development projects mentioned above, and the political pressure to balance the expenditures for

2 LWHCR Physics Experiments and Analyses

The main aim of the PROTEUS-LWHCR physics experiments is the determination of the k_∞ void coefficient, i.e. the change in k_∞ per % moderator voidage in a fundamental mode spectrum. The contributions of various reaction rates to the void coefficient are determined from measured reaction rate ratios. In the LWHCR Phase II programme the test lattices are single rod configurations with the mixed oxide fuel having a fissile plutonium content of 7.5% and an isotopic composition typical for LWR discharged plutonium. The configurations comprise lattices with two different moderator-to-fuel ratios (0.48 and 0.95) and three different voidage states (0, 42.5 and 100% void), the intermediate voidage state being simulated using the organic liquid Dowtherm. Results for the tight lattice have already been published [1], and the analysis of the measurements using the wider lattice is about to be completed.

The principal codes and data libraries used in the analysis of the measurements are

- the lattice code WIMS/D4 together with the so-called 1981 library,
- the collision probability code KAPER4 together with the KEDAK-4 based data set G69CT005,
- the code system AARE together with a 70 group cross section set derived from the JEF-1 data file.

A comparison of the calculation-to-experiment (C/E) values for the more important reaction rate ratios involving all moderated LWHCR Phase II lattices shows the following trends:

- In the water moderated wider lattice C8/F9 is overpredicted by all calculations by about 5% and the C/E values decrease by between 3 and 9% with decreasing hydrogen content.
- F8/F9 in the wider lattice, on the other hand, is slightly underpredicted and the C/E values vary in the opposite sense, i.e. increase with decreasing hydrogen content.
- The fission ratio F5/F9 is reasonably well calculated by all the codes.

	Experiment	Calculations		
		WIMS/D4'81	AARE	KAPER4
α_{vi} (C8/F9)	-34.1 $\pm$ 1.2	-32.0	-31.8	-34.2
α_{vi} (F8/F9)	7.7 $\pm$ 0.2	8.0	7.9	8.8
α_{vi} (F5/F9)	1.0 $\pm$ 0.1	1.0	1.0	1.0
α_{vi} (F1/F9)	-0.3 $\pm$ 0.2	0.3	-0.5	-0.6
α_{vi} (C2/F9)	1.1 $\pm$ 0.1	2.1	1.2	0.9
α_{vi} (others)	25.6 $\pm$ 1.4	24.7	22.6	24.1
net α_v	1.0 $\pm$ 0.9	4.2	0.3	0.0
Note: All α_v are given in units of 10^{-4} / % voidage				

Table 1: Comparison of the k_∞ Void Coefficient and its Components betwen 0 and 100% Void for the Wider Test Lattice

Table 1 shows a preliminary comparison of the k_∞ void coefficient α_v and its reaction rate components α_{vi} (for definition see NEACRP-A-papers 584 and 636) for the lattice with a moderator-to-fuel ratio of 0.95. The net change in k_∞ is measured to be slightly positive and is well predicted by the AARE and KAPER4 calculations. However, a closer inspection of the measured and calculated α_{vi}'s reveals that in both calculations there are some compensating errors: The less negative C8/F9 component in the AARE calculation and the higher F8/F9 component in the KAPER4 calculation are compensated mainly by smaller contributions from the non-measured reaction rates. Furthermore, the excellent KAPER4 prediction of the C8/F9 component itself is fortuitous, since this code is known to overestimate consistently all C8/F9 reaction rate ratios. Due to an inadequate self-shielding treatment for Pu-241 and Pu-242, WIMS/D4 overestimates the α_{vi} contributions from these isotopes by an amount which accounts for half of the observed net α_v discrepancy. Comparing with the results of the analysis of the α_{vi} components in the tight lattice, it is interesting to note that, whereas in the tight lattice the α_{vi} contributions from C8/F9 and the non-measured reaction rates had been slightly overestimated by the codes, the calculations for the wider lattice show rather opposite trends.

In summary, the analysis of the LWHCR Phase II experiments has revealed an unexpected dependence of the C/E values for C8/F9 and F8/F9 on the amount of moderation. However, due to compensating errors in the reaction rate contributions, k_∞ and the k_∞ void coefficient appear to be predicted satisfactorily. The limited suitability of WIMS/D4 for LWHCR voidage calculations is mostly understood. Some remaining discrepancies in the reaction rate ratios and the above mentioned dependence of C/E's on moderation require further investigation.

To obtain qualitative information on the transferability of integral information from the PROTEUS experiments to the design of a power reactor of the LWHCR type, "sensitivity profiles" for a one-dimensional model of PROTEUS have been compared with the respective profiles for a typical power reactor. Based on the similarity of the sensitivities for the two cases, it was concluded that the PROTEUS data is quite appropriate for utilisation in the design of a LWHCR power reactor. For some cases, particularly with voiding, the sensitivities show certain differences unless a fundamental mode model is used for PROTEUS. This latter approach, however, is clearly justifiable, since the correction factors used to convert PROTEUS results to fundamental mode results are small [2]. An analysis representing the PROTEUS driver zones as a fixed source is given in NEACRP-A paper 1003.

Regarding alternative experimental techniques, two new tools are being developed for the investigation of detailed power distributions in the wider lattices, namely

- a special fuel rod manipulator which can be programmed to insert miniature fission chambers at various lattice positions without shutting down the reactor, and

- a semi-automatic γ-scanning rod-drum apparatus which can monitor the relative fission-product activity of upto 15 different irradiated fuel rods.

While the second method provides results of an integral nature with total fission-product γ-counting being carried out, the use of miniature fission chambers in conjunction with an automatic changer allows fission rate distributions for individual nuclides to be determined separately (as with foil activation). The fission chamber measurements, however, require the application of significant corrections in certain cases – to account for the fact that the reaction rates are determined in an empty tube and not in the fuel itself.

3 LEU-HTR Physics Experiments

In a new series of PROTEUS experiments the physics of pebble-bed HTR systems with low-enriched uranium (LEU) fuel will be investigated. The experiments are intended to broaden the experimental data base and reduce the design and licensing uncertainties for small and medium sized reactors using this type of fuel. A specific objective of the experiments is to provide high quality experimental data on the effects of water ingress on reactivity and control rod effectiveness, and boron and hafnium absorbers on reactivity and power distribution. Approval for the experiments has already been obtained and work on the design and licensing of the modified PROTEUS facility is now underway.

Fuel from the AVR facility of KfA Jülich with 6 g of 16.7% enriched uranium per fuel pebble will be used for the first experiments. The fuel will be arranged in core zones with variable diameters inside the existing graphite structure of the PROTEUS facility. Regular (hexagonal) pebble bed configurations with a removable pebble column will allow central pebbles to be accessed for reaction rate measurements. Altogether, only modest modifications to the PROTEUS facility will be necessary.

Two-dimensional transport theory calculations indicate that 5400 fuel pebbles should be sufficient for the initial phase of the experiments involving criticality and reaction rate measurements and water ingress studies in single core zone systems with moderator-to-fuel pebble ratios ranging from about 1-to-1 up to 3-to-1 (C/U ratios of 1260 to 2500), if core zones with diameters from 1250 to 1500 mm are used. Additional fuel pebbles or the use of some of the existing 5% enriched UO_2 PROTEUS driver fuel rods in multizone systems will be necessary for moderator-to-fuel pebble ratios significantly less than 1-to-1 or for cores with significant numbers of burnable poison pebbles.

The PROTEUS-HTR programme has been initiated together with partners in the Federal Republic of Germany. The Swiss contribution to the programme consists of the facility construction, licensing and operating costs as well as a portion of the scientific support staff. The FRG is making an essential contribution by providing the pebble bed fuel for the first experiments as well as considerable scientific support.

The experiments will be performed in the framework of an IAEA Coordinated Research Programme on Validation of Safety Related Reactor Physics Calculations for Low Enriched HTGR's. Through this coordinated research programme, three other countries, the Soviet Union, the Peoples Republic of China and Japan, have decided to participate and supply some of the scientific manpower necessary to plan, execute and analyse the experiments. Some other countries have indicated their interest and may also join in.

The cooperating partners will meet in October 1989 to define the priorities for the initial experiments and ensure that the experiments are relevant and cost effective with respect to the various national gas-cooled reactor programmes. The actual critical experiments are currently scheduled to begin early in 1991.

More detailed information on the PROTEUS-HTR programme, including calculational results related to criticality, safety and shutdown rod worths, and reactivity effects due to certain hypothetical geometry changes, is given in [3] and in NEACRP-A paper 1002.

4 Testing of Data and Methods for Fast Reactor Applications

Preliminary results of a JEF-1 based analysis of the NEACRP fast reactor benchmark model and a comparison with the results compiled in NEACRP-L-243 were reported in the previous NEACRP activities report and in [4]. The calculations started from a MATXS formatted 308-group library generated with the NJOY cross section preparation system and involved a development version of the TRAMIX resonance shielding and group condensation code. The comparison showed the JEF/TRAMIX predictions for the important parameters to lie in the expected ranges, except for the breeding ratio in the blanket and the isothermal core fuel doppler reactivity, which differed from the respective mean benchmark values (mean values of the 16 solutions compiled in NEACRP-L-243) by 3 to 4 (1σ) standard deviations.

In the meantime, an improved version of TRAMIX has become available in the framework of the linked code system AARE. Using this AARE version of TRAMIX, the reference configurations of the fast reactor benchmark model with normal and elevated fuel temperature have been reanalysed with the result that the differences relative to the mean benchmark values of 3 to 4 standard deviations for the breeding ratio in the blanket and the doppler reactivity were reduced to more credible differences of less than 2 standard deviations (see Table 2). Although the new JEF/TRAMIX calculations evidently show some improvements, there is still no explanation for the following phenomena:

- The "centroid" of the central flux spectrum has not increased significantly and is still far lower than that of any other solution.
- Compared with the other solutions, the JEF/TRAMIX route gives the highest Pu-239 one-group capture cross section and the highest Pu-239 α value.

The remaining problems are now thought to be due to an approximation in the Bondarenko type resonance shielding calculation: In the AARE version of TRAMIX the background cross sections are evaluated from the potential scattering cross sections. Whereas this approximation proved to be appropriate for the resonance shielding in heterogeneous LWHCR and HTR cells, this may not be the case for the homogeneous cell of the fast reactor benchmark model. Preliminary cell calculations, in which the background cross sections were evaluated from the total scattering cross sections, indicate that some of the fast reactor benchmark parameters are rather sensitive to this change and will move in the right direction.

5 Assessment of Data and Methods for Heat Deposition Calculations

Under contract for the Next European Torus project a study has been made to assess the impact of the JEF-1/EFF-1 data bases and the NJOY cross section processing scheme on the neutron heating factors (KERMA factors) of a VITAMIN-J structured MATXS library generated at PSI [5]. Basically, there are two methods to compute heating factors, the "direct" method, in which the sum of the kinetic energy of all charged particles and recoil nuclei produced by the different types of neutron reactions is directly evaluated, and the "energy balance" method, in which the heating factors are derived from the Q values of the reactions and the total energy carried away by the secondary neutrons and photons. In conjunction with ENDF/B-5 formatted files which do not contain charged particle data, only the latter method can be used.

It is known that the energy balance method gives inaccurate and sometimes negative neutron heating factors if, in the evaluation of the secondary distributions, the total energy of the neutrons and photons is not conserved. When processing the JEF-1/EFF-1 isotopes with the NJOY HEATR module, negative heating factors in specific energy

	Configuration 1		Conversion ratio		Breeding ratio			Breeding gain	
	k_∞ Inner Core	k_{eff}	Inner Core	Outer Core	Core	Blanket	Reactor total	Blanket	Reactor total
BENCHMARK									
Mean (of 16 solutions)	1.12428	1.00509	1.09262	0.82144	0.98984	0.40227	1.39211	0.41035	0.36404
Std. Deviation (1σ) in %	1.8	1.3	4.2	4.4	4.2	2.4	3.5	2.4	9.7
PSI (JEF-1)	1.11812	1.00584	1.06104	0.79841	0.96172	0.38344	1.34516	0.39869	0.32351
Deviation from mean in %	-0.5	0.1	-2.9	-2.8	-2.8	-4.7	-3.4	-2.8	-11.1

	Fuel Doppler Na In	Central Reaction Rate Ratios			Central One-Group Cross Sections				Flux Centroid keV
		$^{28}c/^{49}f$	$^{28}f/^{49}f$	$^{49}c/^{49}f$	^{28}c	^{28}f	^{49}c	^{49}f	
BENCHMARK									
Mean (of 16 solutions)	-0.00729	0.16589	0.02216	0.29970	0.30617	0.04121	0.55505	1.85467	110.6
Std. Deviation (1σ) in %	10.8	3.7	4.0	5.8	3.5	5.0	7.0	2.1	3.8
PSI (JEF-1)	-0.00872	0.16265	0.02171	0.32469	0.31297	0.04177	0.62478	1.92420	100.6
Deviation from mean in %	19.7	-2.0	-2.0	8.3	2.2	1.4	12.6	3.7	-9.0

Table 2: Comparison of JEF-1/TRAMIX results for the fast reactor benchmark problem

ranges were obtained for Ti-nat, Cr-nat, Co-59, Ni-nat, Cu-nat, Mn-55, Nb-93, Ta-181, W-182, W-183, W-184, W-186, and Pb-nat (EFF). It has now been shown that the anomalies are due to an energy mismatch between neutron data and photon production data mainly in the energy range above some keV. To provide the user of the respective MATXS library with meaningful neutron heating data for the isotopes concerned, the library was updated with the total upper limit kinematic heating factors (MF3, MT443 in NJOY87 terminology). Nevertheless, it has to be pointed out, that reliable predictions of heat deposition require accurate data for both neutron heating and photon production. The photon production data of the afore-mentioned isotopes should therefore be revised.

For the JEF-2/EFF-2 files the use of nuclear model codes to check the energy balance in the evaluations is recommended. Moreover, these codes provide the energy distributions of the charged particles and recoil nuclei created in the neutron reactions. These data are compatible with and should therefore be included in the ENDF/B-6 formatted JEF-2/EFF-2 files. It can be expected that the accuracy of the heating factors in the high energy range can thereby be improved by an order of magnitude.

6 MCNP Implementation and Application

A new version 3B of the Los Alamos Monte Carlo code MCNP has been implemented under UNICOS 5.0 on the CRAY-XMP computer at the Federal Institute of Technology, Zürich, and under VMS on the VAX cluster at PSI for input tests and plotting. The version 3B has the following new features (with (*) designating items already tested at PSI):

- Multigroup capability with adjoint solutions
- Fokker-Planck solution within the multigroup capability
  (This is a first step in an overall effort to add charged particle transport to MCNP.)
- (*) Repeated structures capability, which makes it necessary to describe only once the cells and surfaces of any structure that appears more than once in a geometry
- (*) Special treatment of tallies
  (This feature allows for more general tally definitions, e.g. Gaussian energy broadening, binning according to the number of collisions and many more.)

- New photon production bias card for the coupled neutron-photon calculations

- More general, improved scheme for writing "bootstrap" information for a subsequent MCNP calculation

- (*) Shorthand definitions for similar cells

- (*) Improved geometry plotting implemented in GKS standard

- (*) Integration of the MCPLOT package for tally plotting in GKS standard

- Removal of obsolete features (e.g. once more collided flux estimator)

With MCNP-3A, Monte Carlo results for a 9 cell fragment of a light water reactor square lattice with a central gadolinium loaded pin have been obtained [6]. The calculations were performed using the ENDF/B-5 library and compared with the results obtained from the BOXER (ELCOS) code system and the JEF-1 library. The objective of this exercise was to study the performance of BOXER for the analysis of Gd loaded LWR lattices in the broader framework of the GAP international benchmark analysis.

A comparison of results indicates that, apart from unavoidable discrepancies originating from different data evaluations, the BOXER physical model needs further refinements to provide reliable and accurate analyses of Gd controlled systems. It is hoped that further similar studies using the JEF-1 library for both BOXER and MCNP will help to isolate the BOXER model problems in a cleaner way.

7 LOTUS Fusion Blanket Programme

After some delay due to the late delivery of a replacement for a faulty neutron generator tube, the LOTUS facility at the Institut de Génie Atomique (IGA) of the Federal Institute of Technology, Lausanne, went into operation again at the end of February 1989. Since then, an extensive experimental programme has been carried out, comprising U-233 production measurements in a thorium oxide blanket assembly, tritium breeding measurements in the lithium-lead module, developed and assembled at IGA last year, and the irradiation of lithium samples for the NEACRP "tritium production rate benchmark exercise".

For many years, IGA and the Bhabha Atomic Research Centre (BARC) in India have been engaged in a cooperative programme of thorium blanket studies. In the framework of this joint programme, new U-233 production rate measurements were made in an assembly of thorium oxide rods placed at IGA's disposal by BARC. The aim of the new experiments was to estimate the U-233 production globally in the entire module and not only on the central axis of the blanket as in previous experiments. In addition, a miniature NE-213 neutron spectrometer, developed at IGA, was used to measure neutron spectra at particular positions on the axis of the blanket module.

The thorium blanket module is in the form of a parallelepiped of thickness 27.7 cm and side dimensions of roughly 100 cm by 100 cm. The assembly has 19 rows of 59 thorium oxide rods in aluminium cladding, the rods being suspended from the top support of the LOTUS cavity and positioned 15 cm in front of the neutron generator tube. The U-233 production rate was measured by means of thorium foils of 6.7 mm diameter and 0.2 mm thickness, having a mass of approximately 80 mg. Two different intrinsic Ge detectors were used to count the 311.9 keV γ-rays emitted by the precursor isotope Pa-233. These detectors were calibrated using Np-237/Pa-233 reference sources in the form of foils of the same dimensions as the thorium foils.

The main difficulty of the new experiments was to estimate the global U-233 production from only a small number of experimental data, on account of the limited number of thorium foils available and the total time required to count all the irradiated foils. Based on preliminary calculations made with 2-D transport codes, a procedure was devised to estimate this global production from the results of a few "diagonal" measurements performed on a limited number of selected blanket planes. Using this procedure, it was shown that a fairly accurate estimate of the global U-233 production had been obtained with only 49 measuring positions (7 diagonal positions on 7 different planes). It has also been shown that, due to the importance of the low energy room returned neutrons, a good theoretical estimate of the U-233 production in the blanket will require that the concrete walls of the LOTUS cavity be included in the calculations, a requirement which is difficult to achieve with a 2-D model. Nevertheless, even neglecting this effect, very good agreement has been found between experimental and calculated data in the high energy range above 1 MeV.

The concept and design of a new IGA blanket module intended mainly for experimental tritium production studies in lithium-lead fusion blankets have been described in the previous NEACRP activities report. In this module two different types of experiments were carried out: in-situ spectrum measurements using the miniature NE-213 neutron spectrometer, and tritium production measurements using a lithium-glass scintillation method. In the first of these experiments, the neutron generator was operated at low intensity (around 10^{10} neutrons/s) in order to avoid saturating the detector, while nominal intensity ($3.5 \cdot 10^{12}$ neutrons/s) was selected for the tritium production measurements.

The response of the NE-213 detector is practically independent of the direction of neutron incidence and the detector's small outer dimensions allow in-situ measurements in the blanket assembly to be made without excessive perturbation of the neutron flux. Pulse shape and pulse height discrimination methods were used to obtain γ-free neutron spectra for processing by the FORIST unfolding code. For the tritium measurements, two types of lithium glass scintillators were chosen: NE-912 with 95% enrichment in Li-6, and NE-913 with 99.99% enrichment in Li-7. NE-912 is sensitive to gammas and slow neutrons due to the 1/v energy dependence of the Li-6 (n,α) He-3 cross section. On the other hand, NE-913 has a much reduced response at low neutron energy, but has the same sensitivity to the gammas; it could therefore provide an estimate of the gamma background.

Comparisons between the experimental data (tritium production rate and neutron spectra) and results of calculations made with the 2-D transport code DOT-3.5 have shown a generally good agreement. The more important discrepancies observed in some cases, especially in the neutron spectra, can be attributed to an inappropriate group structure at high neutron energy of the DLC-47 cross-section library used in the calculations.

In the framework of the NEACRP "tritium production rate benchmark exercise", lithium samples were irradiated both at FNS (JAERI, Japan) and at LOTUS. Eight different organisations, representing six countries, joined this international programme. At IGA the irradiation took place on 18 May 1989 and lasted six hours. The intensity of the neutron generator was fixed at $2.6 \cdot 10^{12}$ neutrons/s; two fission chambers were used to monitor the neutron output and verify that its intensity had not fluctuated more than 0.5% during the irradiation period. To ensure uniform irradiation conditions for all the lithium samples (five samples for each of the participating organisations), the samples were placed on a rotating aluminium disk of 25 cm diameter installed 10 cm inside the lithium module. This lithium module was the same as that described above, excepting that provision was made to leave the necessary gap to install the rotating disk. After irradiation, the lithium samples were sent back to their owners for analysis of their tritium content. The tritium productions per Li atom and per source neutron obtained by each participating organisation will be collected and analysed in a report to be presented at the 32nd meeting of the NEACRP.

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REACTOR PHYSICS ACTIVITIES IN THE UNITED KINGDOM 1988-1989

Compiled by M J Halsall

1. NEWS ITEMS

1.1 The Electricity Supply Industry

Following on last year's announcement of the Government's intention to privatise the UK Electricity Supply Industry, there have been many discussions about the viability (or otherwise) of National Power Company (ie the company to be formed from 70% of the total generating capacity including all the nuclear stations). A major problem is the future decommissioning of the Magnox stations at a possible cost of £15 billion and it is probable that only the AGRs and future stations will be privatised.

1.2 Operation of the Magnox Stations

In the UK there were 11 Magnox stations that first produced power between 1956 (Calder Hall) and 1971 (Wylfa). These have been the subjects of long term safety reviews. Berkeley (1962) is now being decommissioned but for many of the stations a satisfactory case has been established and agreed with the Nuclear Installations Inspectorate for continued operation to a possible limit of 40 years.

1.3 AGR Stations

Both units at the two new AGR stations (Heysham II in Northern England and Torness in Scotland) are now supplying electricity to the grid with few problems. For example, after only 7 weeks from first synchronisation, Torness Unit 1 ran for 122 days continuous operation.

1.4 Progress at Sizewell

Construction of the UK's first commercial PWR station, known as Sizewell B, and being constructed on the Sussex coast, is well advanced and due for completion in 1993.

1.5 Future Nuclear Stations

The public inquiry at Hinkley C in Somerset has been under way for several months. By and large the generic safety issues of PWR plant in the UK were settled at the Sizewell B inquiry. The main issues at Hinkley are of an economic or a site specific nature and include emergency evacuation procedures, environmental impact and local economy. The inquiry is, however, not plain sailing for the nuclear lobby; Mr Benn (ex-minister responsible for the nuclear industry), for example has submitted

- that nuclear power is neither cheap, safe nor peaceful,
- that estimates about the cost of decommissioning and long term storage of nuclear waste are purely speculative,
- and, that fixing a percentage of nuclear generation indicates a political decision that has no bearing on market forces or expected economic returns.

In addition to the Hinkley Point C PWR in Somerset the Board intends to apply later this year for consent for a PWR station at Wylfa in Anglesey (North Wales). A second PWR station (Sizewell C) would complete the set of four originally announced. If approved, the three stations would be expected to be commissioned in 1998, 1999 and 2000 respectively. These stations will not provide the anticipated required capacity at the end of the century and the CEGB has therefore already applied for two coal-fired stations (in Hampshire and Nottinghamshire) and intends to apply for a third in Kent.

1.6 <u>Progress at Thorp</u>

Construction at the Thermal Oxide Reprocessing Plant at Sellafield is on schedule for completion in 1993.

1.7 <u>Leukemia Clusters</u>

The claims that clusters of leukemia cases occur in the Sellafield and Dounreay areas, and that these are due to nuclear plant radiation are assuming serious proportions for the nuclear industry as a whole. Some 35 families in the Sellafield area are to take their claims for compensation from BNFL, the operators of the Sellafield site, to court despite the obvious difficulties in proving their case.

2. <u>AEA TECHNOLOGY</u>

On 16 May, the UKAEA launched a new identity, AEA Technology. This had its beginning in 1986 when the UKAEA became a trading fund to put the Authority on a more commercial basis, with the emphasis on customer/contractor relationship rather than reliance on taxpayers' money from government work. The new identity is part of an overall plan to increase its contract R&D business worldwide, and to bring all its establishments and services together under one banner. Eight important market sectors have been identified: aerospace, defence, electricity supply, oil and gas, manufacturing industry, the process industries, safety and reliability, and the environment. However, the AEA's commitment to providing advice and information to the UK government and the UK nuclear industry remains.

As part of the new image, Power Plant Services (PPS), AEA Technology's marketing centre for its power plant support services, was officially launched on 20 June at the "Power Plant 89" exhibition in Birmingham. It will offer a single point of contact for customers who want to enhance plant performance, improve reliability and ensure plant safety for both nuclear and fossil fuelled generation and for transmission and distribution.

Since the launch of AEA Technology a great deal of attention has been focused on an exercise known as the 'Strategic Initiative Project', which is aimed at identifying strategic options for the AEA up to the year 2000. The impulse for the study comes from the impending privatisation of the Electricity Supply Industry, the reduction for funding for fast reactor and fusion research, and the Department of Energy's wish to move the responsibility for nuclear safety work elsewhere. Three possibilities were envisaged:

(i) 'Cut to the Core' - shrink to become a public sector advisory body

(ii) 'Pushing the Limits' - maximise commercial potential (within current legal constraints) to become a leaner, more competitively responsive technology supplier for a broad range of UK and international customers

(iii) 'Potential Unlocked' - form partnerships with commercial companies for non-nuclear business.

The first possibility was considered unacceptable and the third requires primary legislation. In adopting the second option, the present intention is that the AEA's current activities will be restructured in terms of nine major businesses:

- Thermal Reactors
- Fast Reactors
- Fusion
- Nuclear Fuel Cycle
- Decommissioning and Radioactive Waste
- Risk Management Technology
- Oil and Gas Technology
- Environmental Protection Services
- Contract R&D

In addition there will be a constantly developing portfolio of emerging businesses. The current list of candidates includes:

- SIR (see paragraph 3.2)
- Computational Fluid Dynamics
- Power Fluidics
- Healthcare

3. OTHER ITEMS OF INTEREST

3.1 Cold Fusion

Over 100 different cold fusion experiments were performed at Harwell during a three month period. In none of these experiments was there statistically significant evidence of a fusion reaction taking place, and in the absence of new information Harwell does not intend to reopen its programme.

3.2 SIR

A consortium of four UK/US organisations has developed a design for a small passive light water reactor known as SIR (Safe Integral Reactor). The partners in the venture are Combustion Engineering, Stone and Webster (a leading US/UK architect-engineering firm with major nuclear/civil experience), Rolls Royce and Associates (the foremost UK PWR supplier with over 20 units completed and operational) and the UKAEA. SIR is an integral PWR in which the core, the steam generators, pumps and pressuriser are all contained within a single pressure vessel. The design relies entirely on components, materials, control systems and fabrication techniques which are well established and proven. The safety features of the plant reflect the lessons learned in recent years. The reactor therefore has the prospect of being both economic and safe.

3.3 Waste Disposal

UK Nirex has been charged with the task of designing, locating and developing a single deep underground disposal centre for radioactive wastes. From an initial total of some 500 possible sites, Sellafield and Dounreay are at present the most likely. Nirex will carry out a detailed investigation of these two sites during the next year, with a final choice due in 1990. Further investigations will be followed by a public inquiry in 1992-3 and the first waste should be emplaced in about 2005.

3.4 SGHWR

The 100 MW prototype reactor at Winfrith has now operated for 21 years and in so doing has output 11 billion units of electricity. Its continued operation beyond 1992 will not be limited by safety considerations, but does depend on the reactor output and the selling price for the electricity. The 1989 summer shutdown was one of the shortest ever: 85 days compared to the recent average of 113.

3.5 PFR

Although the sales of electricity were somewhat lower than had been hoped, the reactor achieved burnups in excess of 20%, probably the highest levels in any commercial type fuel pins. It is still probable that Government funding for the operation of PFR will be terminated in 1994.

3.6 The AGR Programme

There are at present seven AGR stations in the UK, five operated by the CEGB and two by the SSEB (South Scotland Electricity Board), with a total capacity of 9GWe. These stations are of three different basic designs. Dungeness B had a poor construction record and has not operated consistently. Hartlepool and Heysham I suffered persistent engineering problems, mainly with boilers and steam plant, but in recent months there have been indications that these may have been resolved. The third design built at Hinkley and Hunterston has proved successful and these stations regularly achieve load factors of 70% or better. The two new stations (Heysham 2 and Torness) have been built to an update of the successful Hinkley design.

Although no more AGRs are planned, the current stations will provide the main contribution to the nation's nuclear generation until the next century.

4. THERMAL REACTOR PHYSICS

4.1 Computing Environment

It is with increasing difficulty that the sites with mainframe computers are able to keep their machines usefully occupied. At Winfrith the bulk of code development and production running is now done on the local area network of microcomputers (mainly SUN4). These are not much slower than the mainframes and are generally much quicker in terms of turnround time at a tiny fraction of the capital cost; indeed they are so cheap that many users now have individual priority on one machine.

Some work is also being done to see how the power of a 'surface' of transputers can be exploited to provide parallelism in the solution of Monte Carlo calculations. Each transputer is individually capable of executing the program so n transputers potentially reduce the elapsed time for the job by up to a factor of n - rather less because of the overhead of gathering information together.

The CRAY-2 at Harwell has been used to investigate the practicality of vectorising a Monte Carlo solution. At each stage in the computations all those neutrons which are about to undergo a similar type of physical event are collected together, their data being processed in a true vectorised process. This becomes feasible because of the provision of special 'gather/scatter' hardware in the computer which enables the collecting together of neutron data to be efficient. Speed-up factors in the range of 5 to 10 times have been achieved. A preliminary study has also been made of using the Cray's four processors in a multi-tasking mode so that a combination of parallelism and vectorisation can be tested, to provide an even greater speed improvement.

4.2 PWR Studies

Further progress has been made with the LWRWIMS/PANTHER package of codes for PWR reactor assessment. This is the joint Winfrith/CEGB project reported last year (NEACRP-L-308). Much of the effort has been in establishing all the relevant user options in PANTHER that have been requested by the Sizewell B Project Management Team. The latest version, although still lacking a full QA statement, has the required capabilities of statepoint, fuel management, and 3D kinetics, all with the nodal type of diffusion approach. A very tight set of target accuracies was set for the code package. With the one exception of moderator temperature coefficient, these have been met by the application of pure physics and without empirical adjustment. The moderator temperature coefficient error of -2.5mN/deg C remains unsolved. The use of standard Fortran throughout the package has ensured portability; LWRWIMS and PANTHER run equally well on IBM mainframes and SUN workstations.

A rapid kinetics code known as MYTMUS is being developed as a complementary option to the detailed 3D nodal solution in PANTHER. This is a coarse zone approach with regions of any shape that are chosen automatically to meet a specified target accuracy; the solution technique involves direct matrix inversion of the equations.

4.3 AGR Studies

A first demonstration of the application of PANTHER to AGR analysis has been set up by the CEGB (Berkeley Labs). At present, the AGR thermal hydraulics options are not fully implemented, and neither are the required algorithms for pin power reconstruction. It is intended that the modular variant of the WIMS code scheme, WIMSE, will be fully packaged with PANTHER for AGR work in the future. At present the older ARGOSY code isstill used for production work, but this is increasingly in need of upgrading as the AGRs move to burnable poisons and longer fuel cycles.

WIMSE is being extensively developed at Winfrith. WIMSE is a general modular lattice capability but emphasis is placed at present on meeting the special requirements of AGR analysis in terms of speed and accuracy, and also on taking over any residual options for which WIMSD is still preferred. The CACTUS characteristics transport solution method is being used extensively to establish definitive solutions for single fuel assemblies and also for supercells with cross-cluster power tilts. The biggest problem solved to date was a 9 by 9 supercell of AGR fuel clusters containing 81*36=2916 discrete fuel pins together with all the associated cluster structure.

The MAX perturbation Monte Carlo approach is a method based on the use of a deterministic method for solving a simple form of a problem (eg a multigroup homogenised form) and Monte Carlo for solving the difference between the deterministic problem and the real 3D situation. This has been most successful in treating the difficult AGR problem of burnable poisons. These poisons are at present in the form of gadolinium toroids placed in the axial gaps between fuel elements; a possible alternative is to poison individual pellets at the ends of the elements within the fuel channel. It is therefore essential to use a 3D model to calculate their effect on power distribution and their rates of depletion. MAX results have compared very favourably with experimental measurements of reactivity worth, radial and axial flux distributions and gadolinium depletion rates. The code itself has now been integrated as a module in the latest standard release of WIMSE available through the Winfrith ANSWERS service.

The fuel management code MOPSY is being examined with a view to providing power shaping (by automatic adjustment of rod positions) facilities in PANTHER. The MYTMUS kinetics code has possible applications here too.

All of the above codes are now run routinely on SUN workstations; with a minimum of effort the whole of WIMSE was recently ported to a Whitechapel HITEC machine.

4.4 Experimental Studies

The UK's experimental reactor physics programme is mainly carried out in the versatile, low power, water moderated DIMPLE reactor. The primary aim of the studies is to validate the methods and data employed throughout the fuel cycle for performance, safety and safeguards assessments. Whilst there is significant coherence in the overall DIMPLE programme, the experimental studies can be discussed in terms of the following four broad areas.

(i) Core Physics

DIMPLE has a major on-going core physics programme. Studies in water reflected cylindrical systems have been extended to power reactor geometries with a cruciform array of 3% enriched uranium dioxide fuel pins. This simulated the rectangular corner configuration of a PWR and effectively represented twelve fuel elements. The first version was water reflected as with the cylindrical systems. The cruciform assembly was then surrounded radially by a 25mm thick stainless steel reflector to evaluate the impact of a typical PWR baffle region. Discrete burnable poison pins, comprising stainless-steel clad annuli of borosilicate glass, were then introduced. The poison pin assemblies ranged from a simple isolated pin arrangement, through a central complex cluster which included simulated empty guide thimbles, to distributed poison arrays adjacent to the surrounding baffle.

Extensive reaction-rate distribution measurements throughout each assembly have provided a wealth of data for the assessment of pin power predictions. In addition, absolute threshold reactions were measured at various locations spanning the equivalent PWR barrel position outside the cruciform core. The objective of this study was to assess the uncertainties in the radial fast neutron fluxes important to ex-core surveillance procedures.

(ii) Criticality Safety

Similar lattice-type studies are used to investigate the impact of design changes on fuel manufacturing, transport, storage and reprocessing situations. These studies include both sub-critical and critical assemblies, allowing conditions ranging from normal to extreme accidents to be simulated. The standard method used to determine the degrees of sub-criticality is the Modified Source Multiplication (MSM) technique, although a second technique based on cross- and auto-correlation noise analyses is also being investigated.

A series of measurements performed in DIMPLE to simulate a transport flask loading error have been re-analysed using a MONK6A Monte Carlo code in an attempt to resolve discrepancies between the experimental results and French predictions using the MORET Monte Carlo code. The experiments were designed in collaboration with the CEA at Fontenay-aux-Roses, with the aim of testing current calculational methods and data in extreme accident configurations.

The programme was based on the 5 x 4 compartment boron-steel skip insert used to previously establish a comprehensive range of sub-critical benchmarks. The reference loading for the sub-critical studies and the accident simulation had identical arrays of 3% enriched uranium oxide pins loaded into each of the 20 compartments. To simulate a gross loading error, a cluster of 7% enriched pins replaced the lower enrichment pins in one of the compartments, firstly at the centre and then at the edge of the skip insert. Details of these experiments and their analysis have been published as a CEC report.

In the next phase of the DIMPLE criticality benchmark programme it is planned to investigate a range of configurations relevant to reprocessing and severe accident issues.

(iii) Plant Safety Assurance and Safeguards

An additional benefit of the monitoring techniques developed for the sub-critical investigations is their potential safety assurance and safeguards applications. To extend the technique to the measurement of k-values in a plant environment requires validated codes and a knowledge of the neutron sources present. The DIMPLE programme of sub-critical monitoring development and neutron source strength validation (discussed in (iv) below) indicates that this stage has already been reached and k-values in plants and stores could be routinely monitored using MSM. The final stage is a demonstration of the absolute MSM technique in DIMPLE followed by its application in a plant environment.

(iv) Fuel Burn-up

Techniques for studying irradiated fuels in DIMPLE have been developed to provide experimental data on actinide and fission product reactivities, together with neutron and gamma-ray source strengths. The main objective of this work is the validation of the code packages used to predict burn-up. Such predictions are essential to efficient and safe reactor fuel management, criticality assessment and providing the primary source data for shielding and accident analyses.

Last year, measurements were completed on four CAGR samples, each with a nominal burn-up of 20GWd/t. Currently, the measurements are being repeated with 20GWd/t and 50GWd/t PWR fuels. The reactivity of the irradiated fuel samples, and a range of mixed-oxide samples, is determined by measuring the reactor periods produced by the insertion and removal of the samples from the centre of the DIMPLE reactor. By modification of the assembly loading, the neutron spectrum and the relative importance of fission and neutron absorption is varied to provide valuable diagnostic data. Additional reactivity measurements are performed with a wide range of reference fissile and absorbing samples.

Two sets of other measurements with the irradiated fuel are also performed. The first of these determines the neutron source strength in the fuel by comparing the change in power level produced by insertion of the irradiated sample, and then a calibrated ^{252}Cf source, in the sub-critical reactor. The second set of measurements, using gamma-ray spectroscopy, identifies the sources of residual gamma activity in the samples.

5. FAST REACTOR STUDIES

Work at Dounreay over the last year has had two main orientations:

(i) SPX1 Neutronics Commissioning Studies

- the prediction of core absolute reactivity levels
- control rod worth prediction
- prediction of reaction rates measured by foil irradiations for U235, Pu239 and depleted uranium fission

The methods studied included the use of homogeneous equivalent control rod cross-sections and adjustments to rod follower diffusion coefficients to allow for axial neutron streaming.

(ii) PFR Reactor Physics Studies

- subcritical monitoring of PFR during refuelling
- measurement of the low power isothermal temperature coefficient and Doppler component of the coefficient
- routine control rod/shut-off rod worth measurements, reactivity effect of raising primary flow, etc

At Winfrith a low level of work has continued in support of the operation of PFR at Dounreay. In particular the COSMOS database management system that is used for operation support has been converted to run on local workstations.

Work has continued on the new cell code ECCO for the European Fast Reactor Collaboration. An outline of ECCO was given in last year's report (L-308). The latest version, ECCO3, can treat a limited range of heterogeneous cell geometries. These consist of one dimensional plane and cylindrical geometries and two dimensional geometries providing more exact modelling of pin and plate cells in critical assemblies. Calculations can be performed by a 'reference route' which involves a fine group calculation for the heterogeneous cell model, or by a faster 'design route' which uses more approximate methods.

A number of validation studies have been carried out (many of these at Cadarache where ECCO3 has been successfully implemented). These include comparisons between fine and broad group calculations and between homogeneous and heterogeneous cell calculations for a range of cells from power reactors and critical assemblies.

Work has also continued on the provision of a JEF based data library for ECCO. There have been proposals to use a common route for deriving ECCO and WIMS libraries but at present these do not seem viable as the data requirements - resonance integral tabulations, thermal scattering laws and group structures - are so different.

Reactor Physics Methods Group
Reactor Physics Division
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Reactor Physics Activities in U.S.

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Introduction

The major priority in the U.S. reactor development program is revitalization of the Light Water Reactor industry. Efforts are focused on reduction of technical, licensing, and institutional impediments to new construction of LWRs. Advanced reactors are being developed to provide simpler designs, increased modularization, improved economics, and a greater use of "passive" safety features. Recent contracts were awarded to General Electric for design of an advanced Boiling Water Reactor and to Westinghouse for design of an advanced Pressurized Water Reactor.

Efforts are continuing to develop a modular High Temperature Gas Cooled Reactor and an Advanced Liquid Metal Cooled Reactor. These advanced reactors are expected to compete for orders in the mid to late 1990s.

Reactor physics activities support the development of advanced reactors through the conduct and analyses of critical measurements, improvements in the ENDF/B nuclear data base and improvements in physics design methods and codes. Studies of advanced fuel cycles and a broad range of "passive" safety features have also been carried out.

Core Design Studies

Within the U.S., the emphasis for the deployment of LMRs has shifted from maximizing breeding and minimizing doubling time to enhancing safety, improving public acceptance, and minimizing capital costs. This changed emphasis in goals has motivated new approaches to LMR core design and fuel management; recent U.S. core design efforts have shifted from 1000 MWe and greater reactor sizes to much smaller outputs of 100 to 500 MWe. Recent core design activities have placed emphasis on the enhancement of the inherent reactivity feedbacks, larger thermal inertial attendant pool design, and the use of passive decay heat removal systems. In addition, the interest in metallic fuel has been renewed as a result of advances in metal fuel design and the safety tests conducted at EBR-II.

Two representative 3500 MWth size cores were designed to assess core size effects and the neutronic and fuel cycle performance differences between metal and oxide-fueled LMRs. Detailed neutronic performance and safety characteristics were calculated and analyzed. Control system requirements were evaluated and compared against the available control rod worths. Several differences in the computed performance parameters of metal and oxide cores which arise from basic differences in their neutronic characteristics were identified.

Sodium Void Worth Studies

Systematic analyses of methods for reducing the sodium void worth for plutonium-fueled liquid metal reactors (LMRs) were carried out at ANL. Performance penalties caused by design changes that significantly reduce the void worth have been quantified and the relative merits of each design option was assessed. Design alternatives encompass changes in composition and geometry. Results indicate:

1. The penalties in burnup reactivity loss and fissile requirements can be minimized by use of a "tightly coupled" radial heterogeneous configuration of minimum volume consistent with fuel rating limits and by adjusting the core height to diameter ratio,

2. The reactor breeding ratio penalty can be minimized by the use of loosely coupled heterogeneous cores or annular cores with a large central blanket region.

3. Penalties in core radius and core volume can be minimized by composition changes, specifically replacement of a fraction of the fuel (or steel) by sodium or a moderating material.

No design option appears to be superior for all performance characteristics. Choice of a "best" method for reducing void worth depends on the importance attached to the various characteristics in a particular design effort. Such a choice must be based on broader considerations of technical feasibility, economic viability, and overall safety.

Fuel Cycle Calculations

Techniques have been developed for reconstruction of pin burnup characteristics from fuel cycle calculations performed using the nodal hexagonal option of the REBUS-3/DIF3D code system. The intra-nodal distributions of the group fluxes, nuclide densities, power densities, and burnup are efficiently obtained using polynomial shapes constrained to satisfy node-average and node-surface-average fluxes and number densities provided by the full-core nodal solution. Special capabilities have been included to permit treatment of assemblies containing different types of pins (e.g., experimental assemblies), and of diverse fuel management operations such as subassembly shuffling, rotation, reconstitution, and out-of-core storage.

The reconstruction techniques have been tested by application to three-dimensional core models of EBR-II, PRISM, and SAFR. The maximum errors in the reconstructed power densities (relative to the spatially converged diffusion result) are approximately 3 percent and 5 percent for driver and blanket assemblies, respectively. Initial test results also indicate that the pin-wise nuclide masses can be reconstructed for irradiated pins with an accuracy of better than 2 percent relative to fine-mesh, point-depletion computations. Additional tests of the nodal depletion and reconstruction techniques for multi-cycle problems are currently being performed. These techniques are also being applied to the analysis of experimental assemblies in EBR-II, and comparisons of predicted and measured local burnups will be made to verify the accuracy of the entire analysis approach.

Improvements in VARI3D

VARI3D performs ordinary and generalized perturbation theory calculations in diffusion theory and is used as a standard design tool at ANL. Recent improvements include the development of a new integrator package which relies on an efficient and streamlined structure and is fully vectorized. For typical LMR calculations, the running time on ANL's CRAY has been reduced by a factor of at least 15.

VARI3D performs ordinary perturbation theory calculations within the nodal diffusion theory formalism implemented in DIF3D. For practical applications, these computations are more efficient and more accurate than their finite difference counterparts.

Improvements in COMMIX Performance

Substantial changes have been made in the COMMIX-1C thermal hydraulics code to improve the code's efficiency and reliability. Improvements include changes in computational methods and algorithm changes to achieve vectorization on the CRAYXMP-14. These improvements have enabled decreases in running time for various applications by factors ranging from 100 to 150,000. Users can now avoid questionable shortcuts and force iterative processes to convergence, as well as run problems which would have been impossible earlier.

Application of Equivalence Theory

The implementation of equivalence theory within the DIF3D nodal hexagonal calculational program has been studied at ANL. Application to several LMR benchmarks indicate a significant improvement in accuracy can be achieved for both nodal quantities and "reconstructed" intra-nodal quantities.

Isotopic Mass Tracking

A new computational code has been developed at ANL for isotopic mass tracking in pyroprocessing systems. A code, PYRO, follows the chemical partitioning in process such as distillation, dissolution, oxidation, electrorefining and reduction as well as the decay of radioactivity in time. Heating rates, radiation levels, photon spectra, isotopic distributions, fissile enrichments, and many other important parameters can now be calculated consistently to support the design and development needs for the Integral Fast Reactor fuel recycle demonstration program.

Critical Experiments

An extensive set of critical experiments for the SP-100 space nuclear power project was completed on the ZPPR-20 assembly. The experiments emphasized criticality, internal and external control worths, and the effects of the external test structure on the core parameters. At the end of the experimental program, detailed tests for the ground system instrumentation were run in cooperation with ORNL.

Analysis of the ZPPR-20 experimental results is continuing as a mutual effort by Westinghouse, GE and ANL. The unique feature of the analysis program is the use of three-dimensional, continuous-energy Monte Carlo for both the reactor design and the analysis of the experiments. More traditional analysis methods are being used for data reduction. So far, good agreement with experiment has been obtained for criticality and power distributions, but substantial discrepancies exist for control worth. The analysis activities are expected to continue for almost another year.

The JUPITER-III cooperative program between USDOE and PNC of Japan provided neutron physics data for a sodium-cooled fast reactor of conventional design and about 1100 MWe size. Several critical configurations were built with a core volume of 8500 liters and fissile masses in excess of 3450 kg. The principal measurements focused on the spatial distributions of fission rates and control rod worths. Additional measurements were made for gamma dose distributions, kinetics and a few sodium worths. For the first time in a large, realistic reactor mockup, experimental results were obtained for the fundamental to first-harmonic eigenvalue separation. The analysis used ENDF/B-V.2 nuclear data and three-dimensional diffusion and transport calculations.

The reference core, ZPPR-18A, contained 24 sodium-filled control positions (CRPs) arranged in three banks of 6, 6 and 12. In ZPPR-18B, control rods were one-half inserted in the 6 inner ring and the 12 outer ring CRPs and the core was made critical by enrichment of the inner and outer core regions. Measurements in this phase were of radial and axial reaction rate distributions. In ZPPR-18C, one of the outer ring control rods was withdrawn and reaction rates were measured in a critical configuration in the same locations as in ZPPR-18B.

ZPPR-19A had the same sectored fuel arrangement as ZPPR-18 but was critical with six inner ring control rods fully inserted. This phase was designed to reduce the eigenvalue separation from 4 per cent in 18A to about 2 percent. ZPPR-19B returned to the ZPPR-18A arrangement with 24 CRPs but with a uniform distribution of plutonium and uranium fuel in the outer core. The principal measurements of fission rates and control rod worths were repeated in ZPPR-19B to enable a systematic comparison of the 18A and 19B analyses.

Calculations for the cores used three-dimensional nodal diffusion and nodal transport codes with detailed models of the assembly obtained directly from the experimental database. The k-effective results by nodal transport were essentially the same for ZPPR-18A (0.9925) and ZPPR-19B (0.9926). These results are 0.1 to 0.2 percent Δ k lower than expected for an all-plutonium-fueled core consistent with the lower criticality predictions for uranium cores with ENDF/B-V.2 data. The k-effectives for ZPPR-18B, 18C and 19A with control rods inserted were about 0.1 percent Δk higher.

Predictions of axial distributions were similar to those in previous assemblies - very good in the core region in general and within 1-2 percent adjacent to CRPs and control rods by transport calculations. The missing control rod in ZPPR-18C produced notable tilts in fission distributions across the core of up to 70%. These were underpredicted by a few percent by diffusion calculations.

Various kinetics measurements were made to characterize core decoupling. Results from noise, oscillator and rod drop experiments were consistent with eigenvalue separations of 4 percent (18A) and 2 percent (19A). The calculated eigenvalue separation for 18A agreed well with the measured value.

Two significant features appear from these studies. One is the invariant C/E for fission rates with radius along the x-axis of ZPPR-18A. These fission rates were insensitive to the predicted reactivity of the uranium sector. Analysis of the earlier plutonium-fueled assemblies in ZPPR-10 with critical masses from 2500 kg to 2600 kg gave an increasing C/E with radius. The difference in prediction for ZPPR-18 appears to be related to the inner/outer core sizes and enrichment ratios.

Second is the question of interpretation of measurements in zoned cores which have been necessary in many critical experiments. In ZPPR-18/19, predictions of control rod worths varied significantly with the prediction of uranium sector reactivity. Consequently, the derivation of bias factors for control rod bank worths for an all-plutonium-fuel core is not straightforward. Nuclear data sensitivity analyses may be necessary for producing bias factors for reactor-wide parameters.

Pu-238 Mission at FFTF

Intensive studies have been conducted over the past year to determine the ability of the FFTF to produce Pu-238 isotope sources suitable for space missions. A novel approach has been adapted which utilizes yttrium hydride to moderate the neutron spectrum surrounding neptunium target material included in the radial reflector region of the reactor. This arrangement maximizes Pu-238 production while also minimizing the (n, 2n) production of the undesirable Pu-236 contaminant. By utilizing structural materials near the targets having low (n, γ) cross sections, the (γ, n) production of Pu-236 is further minimized so that acceptable flight quality material can be produced.

Multiple Isotope Production (MIP) Test

A special Multiple Isotope Production (MIP) assembly was successfully fabricated and irradiated in FFTF for the purpose of reducing uncertainties in the key cross sections associated with the production and decay of Pu-236. Based on the experience gained from an earlier hydrogen-moderated cobalt production test, this highly moderated assembly was equipped with extensive dosimetry and also included several isotopes important in medical applications. Extensive Monte Carlo calculations gave excellent and comprehensive agreement with the numerous measurements to support the interpretation of test results.

Passive Safety Test in FFTF

Results of the FFTF Passive Safety Tests conducted in 1986 were documented in several papers published at the ANS Topical Meeting on Safety of Next Generation Power Reactors, held in Seattle in May 1988. Analysis of the experiments allowed a successful separation of reactivity feedbacks due to fuel (Doppler and axial expansion) and structural effects (core expansion, different control rod expansion, bowing, et al). Substantial work was done to validate portions of the SASSYS neutronic, thermohydraulics code with the data obtained.

Design of Advanced Neutron Source Reactor

Several physics studies were carried out at ORNL to select the Advanced Neutron Source core configuration and to evaluate the accuracy of diffusion and transport calculations for predictions of the multiplication factor, fission source distribution and scalar fluxes. Coupled two and three-dimensional radiation transport calculations were performed to evaluate various single and split core designs. Ongoing studies include design of N-15 and deuterium cold neutron sources and the optimization of beam tube fluxes. These are part of the conceptual design studies for the Advanced Neutron Source Reactor that are necessary to support a construction decision in the early 1990s.

Nuclear Data

Evaluation efforts for ENDF/B-6 are nearing completion and testing of the completed data files is in progress. The status of covariance data files will be reviewed by the Cross Section Evaluation Working Group in November 1989. It is expected that the release of some ENDF/B-6 files will begin in December as scheduled. The contents and performance of ENDF/B-6 will be presented in several papers at the November 1989 ANS meeting.

Reactor Physics Activities in the Joint Research Centre of the European Communities Oct.1988 - Sept.1989

Compiled by H.RIEF

RADIATION PHYSICS AND REACTOR CORE ANALYSIS

3-D Reactor Physics Analysis of PHEBUS

In the frame of the Reactor Safety studies and in particular the Source Term problem the JRC participates with a substantial manpower in the PHEBUS fission product release experiments carried out at Cadarache.

In the preparation of these experiments 3-D neutron transport calculations have to be performed. They aim at the determination of small changes of the neutronic properties, the multiplication factor and the power distribution as a consequence of melt-down processes in the test-loop and other variations of in the reactor configuration.

To this end a detailed 3-D Monte Carlo model of PHEBUS had been elaborated as shown in Fig.1. So far a series of reactor parameters have been calculated by KENEUR and will - at least partially - be validated by measurements. To meet part of the requirements it will be necessary to equip KENEUR with new algorithms which provide explicitly changes of flux and fission rate estimates caused by small perturbations.

Development of a New Neutronic Module for the European Accident Code-2 (EAC-2)

The European Accident code-2 is intended to become an advanced code system for the analysis of low-probability unprotected whole-core accidents in LMFBRs. Besides the detailed modules for fuel behaviour, sodium one and two-phase hydraulics, and molten fuel dynamics, an advanced neutronic module is being developed.

The 3D hex-z nodal diffusion and transport code HEXNOD has been further benchmarked by its developer Siemens/KWU. Comparisons with 3D Monte Carlo calculations from Interatom, FRG for the KNK-II fast reactor at KfK showed very good agreement and the speed of the HEXNOD transport calculation is more than an order of magnitude faster than that of comparable 3-D finite difference diffusion calculations. The transport option of HEXNOD actually uses only about 50% more time than the HEXNOD diffusion option. The calculational speed of HEXNOD makes it attractive for LMFBR safety calculations in which many recalculations of the flux are necessary. Comparison of HEXNOD calculations with one KfK SNEAK experiment simulating a disrupted LMFBR core showed that the HEXNOD transport calculation is not fully satisfactory for nearly voided regions. A correction in which the uncollided neutrons are treated separately is presently attempted.

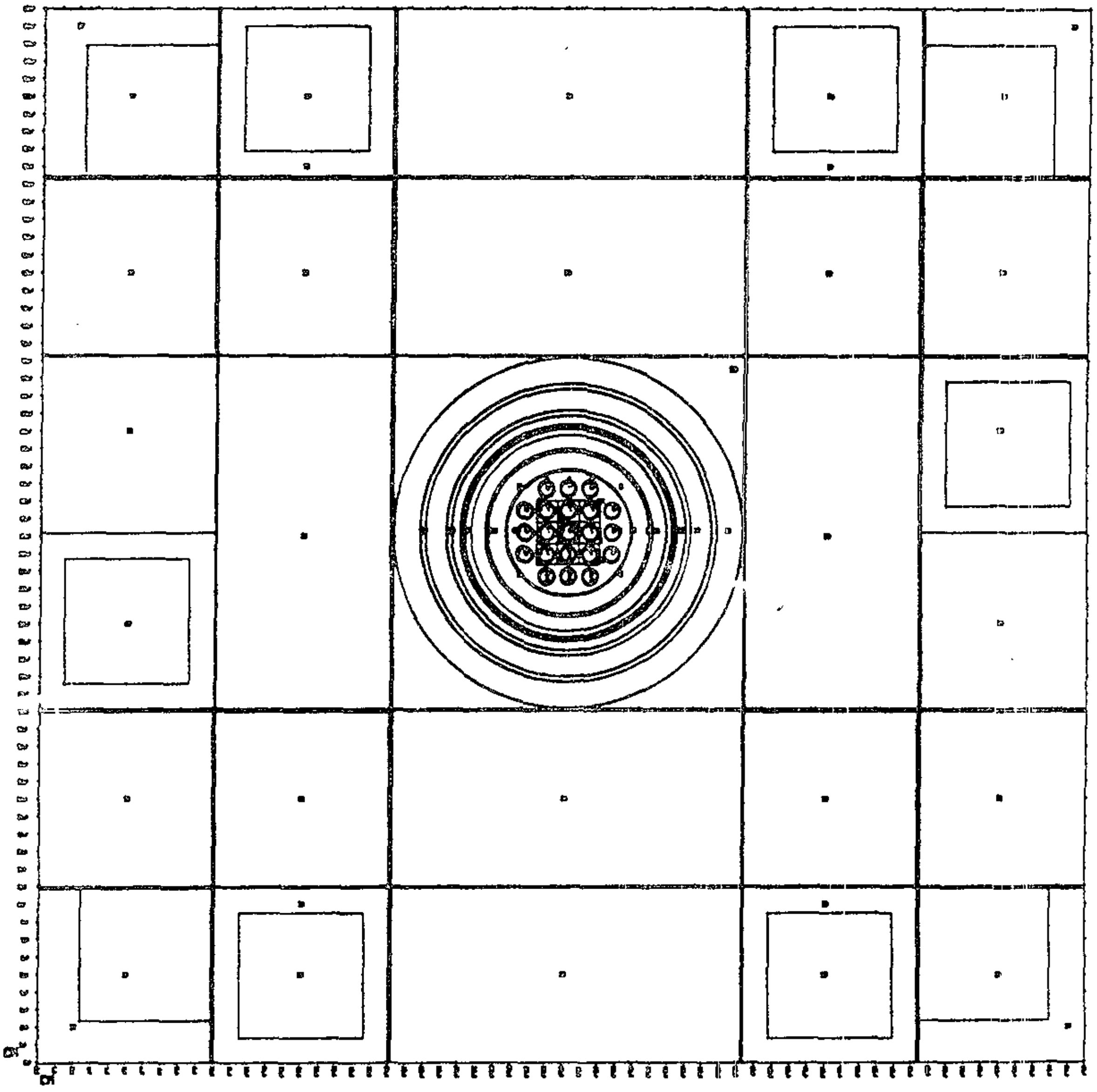

Fig. 1a : Horizontal cross section of PHEBUS as specified in KENEUR calculations

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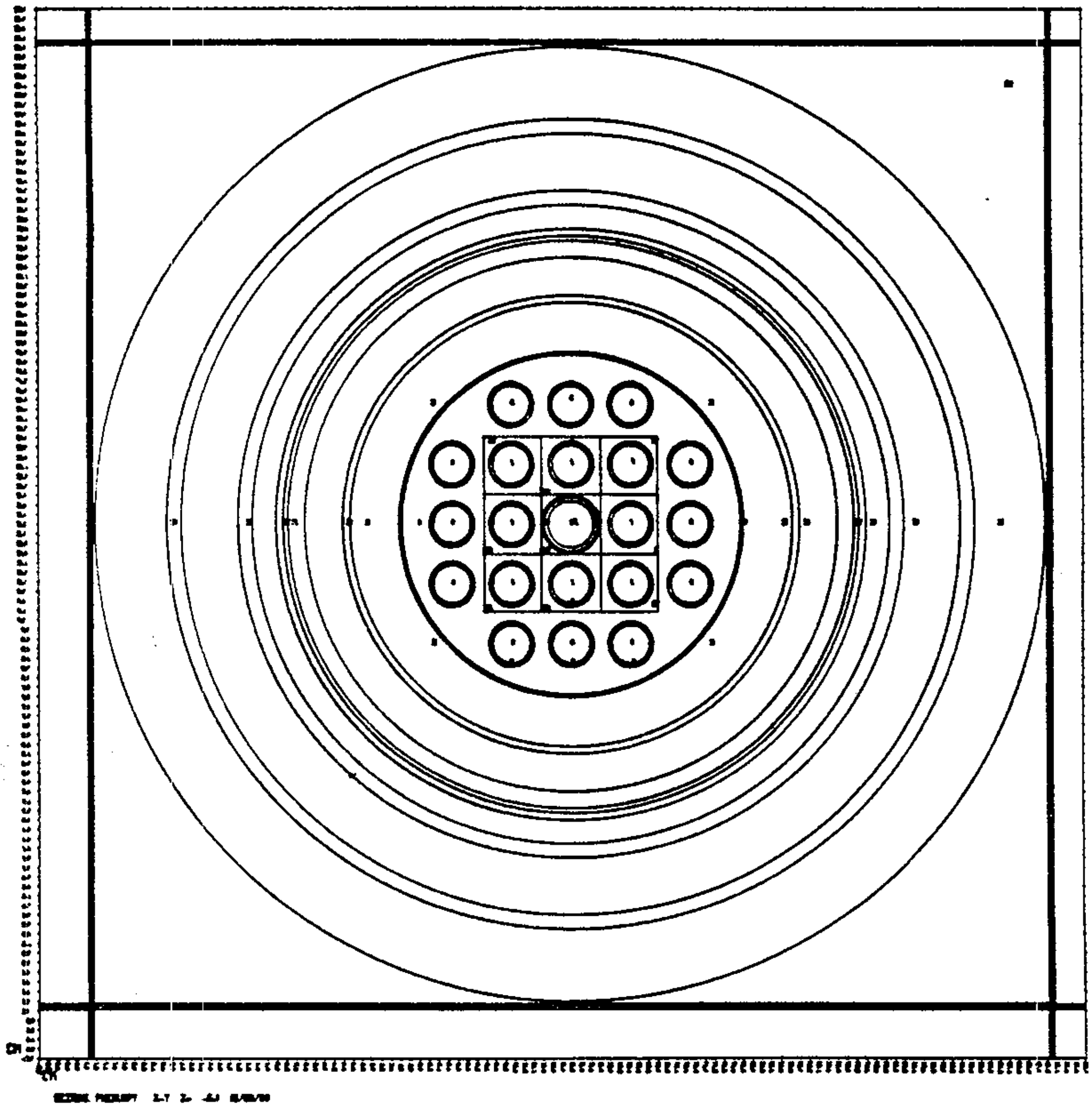

Fig. 1b : PHEBUS Test-Loop as used in KENEUR

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The coupling of the quasi-static time-dependent calculation of the CASSANDRE code has been coupled to HEXNOD by the Université Libre of Brussels. The new code called HEXNODYN has been successfully compared with a time-dependent r-z benchmark problem for a control rod withdrawl accident. More benchmarking is necessary and the coupling to the EAC-2 code is still a major effort.

An efficient cross section generation for the time-dependent calculation appeared necessary. Therefore the MXS cross section preprocessor of SIMMER V.12 from LASL has been made operational at JRC Ispra. MXS produces material macroscopic cross sections and self-shielding interpolation formulas. From these and the changing material densities in EAC-2 it seems possible to generate total macroscopic cross sections rapidly.

Another effort is the inclusion of perturbation theory into HEXNOD; It was found that accurate perturbation results can only be obtained if one bases the products of the flux and adjoind flux gradients on the flux moments. If one uses a reconstruction of the fluxes with second order polynomials, the perturbation results are not very accurate.

Planning of a Boron Neutron Capture Therapy Facility at the High Flux Reactor Petten

At the JRC-Petten since some time project studies of an epithermal BNCT facility to be mounted in a beam tube are underway. In a preliminary design proposal elaborated by Harwell it is foreseen to combine an aluminium spectral shifter with a sulphur filter and a liquid argon photon shield.
This configuration which targets at an epithermal neutron beam is the result of a large number of MCNP calculations systematically examining a wide range of materials and dimensions. Even if it was not possible to review all conceivable combinations of filters and spectrum shifters the configurations which have been considered so far should be close to an optimum choice and significant improvements are not anticipated for other combinations.

The "best" layout parameters are of the order of 1-2E+2 n/cm^{2}sec with an average (weighted) energy of 5-8 keV and a photon dose of less than 40 rem/h.

To validate the calculations a prototypical configuration is being mounted and tested in the beam-tube HB7 (diameter 18 cm).
This facility which does not comply therapy requirements is considered to be a test-bed for a more powerful one to be installed at a beam tube with a larger cross section. It will be employed for dosimetry studies and possibly for animal experiments.

THE NUCLEAR MATERIAL MEASUREMENT ACTIVITIES

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 :

- Development of instruments and interpretation models
- Testing the performance of soft and hardware of the instruments in the PERLA laboratory with a large variety of UO_2, PuO_2 and MOX fuel samples
- Training of inspectors of the IAEA and the EC on individual NDA instruments and in a simulated physical inventory verification exercise (PIV) of nuclear fuel cycle facilities.

Development of NDA Instruments

a) Pu isotopic composition determination by high resolution gamma spectroscopy (Pu meter).

The accurate knowledge of the isotopic composition of Pu fuel is very important especially for the interpretation of neutron correlation measurements and calorimetry. For this reason several gamma spectrum unfolding codes are being tested (PUIC-JRC, LAPIS-LANL, GRPAUT-MOUND, MGA-LLNL) in order to identify the best tool for the various applications in mind. The tests are performed with different samples occurring in field campaigns using JRC's PERLA fuel standards. The experimental programme has been terminated and a technical report comparing the performance of the various codes is in elaboration.

b) Uranium isotopic composition determination by large resolution gamma spectroscopy (U enrichment meter)

The uranium enrichment measurement is executed with high resolution gamma spectroscopy. Data acquisition and processing is performed with a dedicated microprocessor. No further development of this instrument is foreseen at present.

c) Pu-mass determination by a determination of the spontaneous fission rate via neutron correlation counting techniques (Shift register, Correlation Analyser, Fast time of flight multiplet analysis).

The shift register instruments were equipped with personal computers operating with MS DOS. The software consists of a data acquisition, data elaboration, and data interpretation system. The latter uses the JRC models for neutron multiplication and dead-time corrections of the measured singlets and doublets. First measurements were started in the PERLA laboratory in order to improve the present existing dead time correction procedure. Further experiments are foreseen to study the influence of the stainless steel containers housing the PuO_2 samples in order to get a single calibration curve for the Pu mass irrespective of the container type used.

For the radioactive waste programme a waste barrel instrument has been set up mechanically. The main frame, the detector modules and the automatic control of the waste barrel movements is completed.

The instrument will use the Time Correlation Analyser TCA built at HTE-Udine during 1988. The TCA measures the frequency distribution of X signals in up to 16 observation intervals. Each observation interval is triggered by each signal. Analysed are the frequency distribution and the factorial moments of this distribution. The software for the data acquisition, elaboration and interpretation exists for MS DOS compatible PC's. Theoretical and experimental studies with various seized waste barrels and waste matrices are being continued to explore the full potential of the TCA technique.

d) Pu-mass determination by calorimetry

The calorimeter for the mass determination of Pu is being set up in the PERLA laboratory and tests are being carried out with Pu samples having a heat output of up to 40 watts. A typical measurement has a duration of about 8 hours and accuracies in the order of a few ‰ were achieved. The measurement time can be reduced considerably with a preheating of the sample before its measurement in the calorimeter inside a special temperature controlled partially heat insulated box.

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

After the redesign of PHONID (it became more easily transportable) 3 such measurement devices were ordered by the Euratom Safeguard Directorate. These new instruments have a remote facility for the loading and unloading of two 5 Ci SB-gamma sources. They are equipped with a PC controlled interlock system, a performance control of the He-4 counters and the associated electronics and a data acquisition, elaboration and interpretation software with a modified error estimation model.

f) Passive, gamma-scanning (fuel rod scanner, MTR-fuel element scanner, segmented waste barrel scanner)

The PERLA laboratory is equipped with two fuel element scanners, one for MTR fuel elements and one for single fuel rods. The MTR rod scanner measures with a high resolution Ge-detector the Se gamma-transmission in order to verify the completeness of the fuel element, and the axial average of the 186 keV gamma-ray activity of the U^{235} alpha-decay product Th^{231}. This scanner is now completely redesigned and equipped with PC controlled drives for longitudinal and rotational movements. No additional work has been done on the fuel pin scanner.

g) Training

Training courses are being performed at the JRC using the techniques listed under the headings a) to f). These courses are partly courses on individual instruments or courses performing a simulated "Physical Inventory" of a fuel cycle facility. The latter courses are performed with IAEA and EC inspectors having teachers from IAEA, LANL, NUKEM and other EC laboratories or industries. During these courses a part of the PERLA uranium and Pu samples are used in several instruments of different kind.

h) The Performance Laboratory PERLA

This laboratory is now functioning for the use of Pu or U fuel test samples of known isotopic composition and weight. The laboratory and its infrastructure can be used by any customer either via collaboration or customer contracts. The samples available are :

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

- a reduced family of MOX powder of various size

- a family with existing fuel pins of reactors

- a family of samples with high enriched U (powder, U spheres, metal plates, fuel elements)

- a family with intermediate enrichment

- a family with low enrichment.

The laboratory is used mainly for measurements with larger fissile masses in order to :

- check interpretation models and associated nuclear data

- performance tests of instruments with respect to measurement accuracy and reliability of components

- develop measurement techniques.

FEASIBILITY STUDIES OF PASSIVE SAFEGUARDS INTERROGATION OF SPENT-FUEL CONTAINERS

Neutronic Monte Carlo calculations of spent-fuel containers indicate that it might be possible to safeguard such containers by passive measurements, based on the internal neutron sources. The spectra and azimuthal distributions of the neutron flux at the outer surface of a seven-fuel-element container were calculated for fully and partially loaded containers, and for several burnups.
Empty positions in the outer ring of six fuel-element cells could be easily detected by the telling changes in the azimuthal flux pattern. As for an empty central cell although the differences between the flux patterns for this configuration and for the fully loaded container decrease with increasing burnup, some corresponding azimuthal flux ratios still differ by as much as 10%. All in all results suggest that certain measurements could detect any missing element in a seven-element container.

SAFETY OF FUSION TECHNOLOGY

Low Activation Materials

The most significant factor determining the safety and environmental characteristics of a fusion reactor is the intense radioactivity of the materials surrounding the plasma. Unlike the radioactivity associated with the fuel cycle of fission reactors, this is not an intrinsic feature of the thermonuclear reactions themselves but arises from the nuclear reactions induced by plasma neutrons on surrounding materials. LAM (low activation materials) development is an important means of enhancing the advantages of fusion as a safe and environmentally beningn energy source.
The importance of LAM has been widely recognized /1/ but it is difficult to specify which nuclear and non-nuclear properties the material should exhibit, and establish priorities.

As long as the radioactivity remains confined within stable and immobile materials, it does not raise serious concern. There is concern, however, when the material, or part of it, is moved across the different containment barriers or through circuits and finally is released into the reactor hall and/or the environment. LAM are those materials that either decay in short times; or have a negligible probability in any normal or abnormal event, during or after the operation of the plant, to be released into the reactor hall and/or into the environment; or the effects of their possible releases remain within "acceptable limits".

Unfortunately, it is not always clear what the "acceptable limits" are, or what are the appropriate radiological criteria to apply. This is true in particular for waste disposal: there is no regulation applicable to all EC countries /2/ but only acceptance criteria different for each site and different from those applicable in the US.

More attention has been devoted to the waste problem and to the long-life radioactivity; recommendation for material composition have been given by several authors /3/. In line with these recommendations studies have been carried out to produce and characterize low activation austenitic and martensitic steels. The main criteria were to achieve shallow land burial conditions according to US norms and to extrapolate them to fusion. Actually the analysis of the European situation /2/ shows that these norms are not applicable to many states on this side of the Atlantic.

In FRG all nuclear wastes must be disposed of in deep geological repositories.

In May 1987 the UK government announced that although it is believed that a safe near surface facility for low level waste (LLW) could be developed, for economic reasons this option would no longer be considered.

In France the Centre de Stockage sur la Manche accepts only LLW and MLW containing nuclides with less than 30 years half-life.

Two geological repositories are currently operated or planned for the management of nuclear waste in Sweden.

Only relatively small repositories within nuclear sites are available in other European Countries. Shallow land burial (SLB), when applicable, is limited to solid waste that decays to very low levels while institutional controls are operating.

The final disposal of wastes from the FW and blanket of European fusion reactors will be geological disposal /4/. According to this line, the criteria for LAM may be completely different: decay heat could be more restrictive than activity or dose, the allowable concentration of radioactive nuclides could be quite different from that of US shallow land burial criteria, more relaxed, and site dependent.

The attempt to reduce activation will emphasize other items, related to short-medium term radioactivity, and in particular to accidental release.

Data and experimental information on accident situations, mobilization and transport of radioactive material are being developed at present /5/ but the state of the art is far from being satisfactory and there is still much to learn.

The development of LAMs will go through more steps. The first will be the production and characterisation of reduced activity steels by the replacement of troublesome elements with better ones. The second step will be the development of non-ferritic alloys mainly based on vanadium and chromium. Finally possible new types of LAM could be studied (e.g. SiC).

Validation of Activation Inventory Codes

The data base for the activation calculations may be usefully assessed through a set of benchmark calculations. The decision to launch such a benchmark was taken by the laboratories involved in the programme of fusion material development, in a meeting held in Garching on Nov. 1987; the coordination of this activity has been assigned to the JRC-Ispra.

The benchmark has been divided into two parts. The first part has to validate the codes and the decay data, the second part will address the activation cross section libraries.

Only two laboratories have till now provided results for the first part of the benchmark, that is using the GREAC-ECN1 library. These are Harwell and Ispra /6/.
The agreement between the results of the two codes is quite satisfactory; both ANITA and FISPACT may be considered as reliable and valuable tools for activation calculations in the complex neutron irradiation conditions of a fusion reactor.
This study was very useful to validate the codes; a few minor improvements of the two codes had been carried out during the study. Major changes in the decay data base of the ANITA code had to be made.

The next step, to compare differences due to the use of a different activation cross section library is now in progress.

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INSTRUMENTATION IN SUPPORT OF THE FUSION PROGRAMME

Particle Diagnostics

Some experimental work and feasibility studies are performed in support of the IGNITOR project. It is intended to measure the total neutron yield by classical techniques using fission chambers, BF_3 He-3 and He-4 counters. For thermal neutron detectors the counters will be surrounded by suitable moderator assemblies. Eventually solid state detectors will be used to discriminate between neutrons originating from (p,d) and (d,t) reactions. The neutron emission rate as a function of time could be studied with the same detectors. For the study of the neutron emission profile two types of collimators are proposed. For the measurement of the neutron spectrum a telescope (n,p)-spectrometer and a telescope Li^6 (n,t)- spectrometer are considered.

Cold fusion experiments

Immediately after the communication of the discovery of "cold fusion" the JRC Ispra has set up two experiments with electrochemical cells. The cathode was a palladium tube with a diameter of 10 mm, a length of 50 mm and a thickness of 0.8 mm. For the anode a platinum ring electrode was used. The heavy water had an isotopic title of 99.7. Li OD, 0.1 mm was used as electrolyte.

In all experiments a constant voltage of 5 V DC was applied. In the first experiment neutrons were counted using a scintillator detector having an estimated detection efficiency of about 5%. In the second experiment a cylindrical array of 18 He detectors imbedded in a polyethylene moderator was used, with a detection probability of about 25% and a background rate of about 2 counts/s. In both cases no statistically significant neutron emission could be detected.

Other experiments are continued to repeat the measurements of SCARAMUZZI. During the execution of the different experiments it became clear that the normal neutron counting equipment used for fissile material management purposes is not sufficient for low level neutron counting if counting rates close to cosmic radiation have to be measured. For this purpose a special neutron detector head is in construction, which will also serves for the measurement of low level Pu contaminated waste.

IAEA
(USSR)

REACTOR PHYSICS RESEARCH IN THE USSR
(Review)

I. S. SLESAREV

Main efforts on fundamental neutron-physical investigations are now concentrated on increasing in an accuracy of predictions of reactors behavior in emergency situations, the development of "safety physics" of reactors of both the present-day and future generations of APWR, LMP, MHTGR, HCLWR types; the search for fundamentally new concepts of enhanced safety reactors and optimization of fuel cycles; the further development of reactors theory and methods for their mathematical simulation.

1. THE DEVELOPMENT OF THEORY AND METHODS FOR MATHEMATICAL SIMULATION

To increase safety and economical efficiency of nuclear reactors the attempts are being made to attain higher accuracy of calculations of neutron fields in reactors, without noticeable increase in computational efforts.

The surface harmonics method (SHM) being developed by the group of researchers headed by Doctor N.I.Laletin, makes it possible (with an insignificant increase of computional efforts) to reduce the methodical error of calculation, being in the available approaches, to the level of an error related to the uncertainty in the knowledge of microconstants [1].

From the mathematical point of view SHM is the unique method for constructing the finite-difference equations of higher accuracy, describing the neutron fields in heterogeneous reactors.

The finite-difference equations in SHM are obtained as follows:

- a solution is sought as a sum of trial functions each of which is assumed to satisfy the equation of neutron transport in a certain cell (or "node"). Each trial function in the set, relating to the given cell,differs from the others by the specified condition on the cell boundary;

- by using a certain variation principle the connection between the amplitudes of surface harmonics relating to the adjacent cells is found;

- the obtained system of algebraic equations is transformed to such a form that the SHM system in the initial approximations becomes similar to the equations of the conventional method of homogenization.

The resulting refinements can be classified as follows:

- the main correction for "a large pitch";

- the allowance for the surroundings of the given cell in calculation of its characteristics;

- the refinement of the Fick group law with the result that a full (nondiagonal) matrix of diffusion coefficients arises;

- the allowance for higher spatial harmonics;

- transport corrections on the cell boundaries.

From the use of variational principle it follows that even in the low SHM approximations not the current and the neutron flux are joined but rather the current and the local neutron level. The obtained equations prove to be more accurate. These corrections are particularly important for description of neutron field inside the fuel assemblies as well as for small cores (for instance, for the research reactors).

When calculating energy release for VVER assemblies with use of SHM differed from experimental data by 1-4%, which is 3-5 times less than in the similar calculations by the homogenization method in heterogeneous areas near the reactor core boundary with the reflector or near the "strong" absorbers.

Among new precision methods is the method of surface pseudosources for solution of the neutron transport equation. The method is based on use of the Green function in the subareas (zones) which comprise the whole area. Inside the zones macro-cross-sections of neutron interaction with the matter are deemed independent of the coordinates. The solution in each zone depends on unknown sign-invertable (pseudo) sources located at the zone boundaries. The system of integral equations for determining of pseudosources strength is obtained from the requirement of continuity (along a beam) of a solution on the adjacent zone boundaries [2]. The success of the method and the corresponding codes [3] is stipulated by obtaining convenient expressions for Green functions moments in the main coordinate systems: Cartesian, spherical and cylindrical.

To carry out precision reactor calculations based on the Monte-Carlo method [4] the module calculation complex MCU is developed. It has been functioning since 1983 and has been verified by several hundreds of real and benchmark-tests (uranium-water, graphite, D_2O, assemblies and fuel lattices). The used neutron constants libraries are such that the accuracy of description of critical benchmark-experiments commonly-used for their verification is as high as that of the known programs MONK, MCNP (based on ENDB/V libraries).

The program is used for analyzing practically all types of the power reactors (HTGR, ER, RBMK, advanced reactors) and fusion blankets.

The specific features of the code are:

- the opportunity of group and point presentations of cross-sections (including the infinite number of points within the region resolved resonance , the allowance for thermalization effects and the subgroup description of unresolved resonances;

- the wide set of geometrical modules, specialized and multi-purpose, which makes it possible to describe effectively systems of practically any geometry (including HTGR micro fuel elements);

- the wide set of calculated functional, including parameters to check engineering codes using both the traditional methods (for example diffusion coefficients for homogenization method parameters for Galanin-Fainberg method) and the present-day ones - surface harmonics, surface fluxes etc;

- the module structure and computer-independence.

In particular, there exists the possibility of utilization of various physical modules.

New algorithms were developed making it possible to use the MCU calculations to increase the accuracy of engineer codes (for example the fast calculation of the first collision probabilities for complex geometry system without using random numbers).

The new version of MCU-2 is produced on FORTRAN-77.

The ultimate objective of this work is the development of benchmark mathematical models and codes based on the Monte-Carlo method for calculations of reactors main parameters without simplifications in describing the geometry and processes of particle interactions with substance, with an accuracy limited only by the accuracy of nuclear data and spent computer time. The perturbation theory algorithm is being developed.

The MCU-20 specific features are:

- the possibility to carry out calculations with the aid of most reliable nuclear data libraries available in the USSR;

- lifting of all methodical restrictions for an accuracy of mathematical simulation of particles interaction with substance within the scope of the used libraries of neutron constants;

- the opportunity of simulation of systems with a very complex geometry, taking into account the dependence of parameters on coordinates;

- the possibility of solving a number of problems of the perturbation theory;

- the possibility of solving problems by combined (determnistic and statistical) methods, including the non-analog simulation.

One more promising trend is associated with the development of variational combined schemes to solve a transfer equation in media with a complex geometry structure, having practically any neutron spectra: from hard to resonance and thermal, with a detailed description of the energy dependence of cross-sections. Such methods are for a simulation of neutron fields in new hard-spectrum breeders, breeders-actinides-burners, tight LWR lattices, advanced steam-cooled reactors, etc. [5].

Creation of a safe nuclear reactor requires the development of methods for analysis and optimization of reactivity effects, an effective yield of delayed neutrons, an average lifetime of prompt neutrons and their various combinations being a part of the dynamic equation, as well as combinations of reactivity effects providing the balance of reactivity. To estimate these characteristics in addition to direct calculations the perturbation theory methods are applied. The classical and generalized perturbation theories (PT) have been developed, together with accurate PT use is made of linear perturbations approach within the framework of which are determined coefficients of sensitivity of the examined characteristics to the change of reactor parameters.

2. STUDIES OF SENSITIVITIES OF PHYSICAL CHARACTERISTICS TO NUCLEAR CONSTANTS [6]

To increase the safety of designed fast-neutrons reactors the calculated error of their reactivity characteristics should be known. One of significant components of an error is constant one specified by nuclear data uncertainty.

The constant software APAMKO-CI (БНАБ -78) was used as nuclear data.

Errors of K_{eff}^{nom} and K_{eff}^{unf} amountes to 2.2%. The greatest contribution to the K_{eff}^{nom} error are made by constants σ_f ^{239}Pu - 31% (+0.44), σ_f ^{235}U-23% (+0.66%), χ - 16% (+1.0), σ_c ^{238}U-9% (-0.19), σ_{in} ^{238}U - 5% (-0.04), σ_f ^{241}Pu - 5% (+0.10).
In the unflooded state the error is determined by the same constants and their contribution to the error is practically not changed except for σ_{in} ^{238}U, which grows up to 7% due to hardening of neutrons spectrum.

The error of VRE was 0.45% (in % from K_{eff}). The main contribution to the error is made by constants of "fast" σ_{in}Na groups - 29%, as well as by constants σ_{tr} Na - 12%, σ_f ^{239}Pu - 11%, σ_f ^{235}U - 10%, σ_c ^{238}U - 10%, σ_e Na - 8%, σ_{in} ^{238}U - 5%.

The error of reactivity losses (RL) (in % from K_{eff}) amounts to: RL (110 days) = 0.4%; RL (440 days) = 1.3% at values: RL (110 days) = 2.9%; RL (440 days) = 10.7%, i.e. the RL relative error with respect to a core cycle length reduces slightly from 14 to 12%.

To refine reactivity effects the following values should be corrected: first of all, the constants σ_f: ^{239}Pu, ^{235}U, σ_c ^{238}U, the fission neutron spectrum, σ_c of fission products; and also

σ_{in} Na to make more precise the reactivity void effect with respect to sodium.

The light-water power reactor with a fast-resonance neutron spectrum is one of the promissing trends of perfection of light-water reactors and has high fuel utilization factors (BR $\geqslant$ 1.0) and comp etitiveness in the system of developing NP. The most important feature of the reactor is its enhanced safety level based mainly on the specifically "designed" reactivity properties. The negative values of the void reactivity effect (VRE) and the reactivity effect of flooding with water without boron, the correspondence of reactivity maximum to the reactor working point ensure the reactor self-protectiveness agains possible accidents and its neutron-physical stability. Because of stringent requirements upon a value of reactivity effects, imposed by the condition of reactor self-protectiveness, requirements for an accuracy of their calculations grow. One of important components of a calculation error in reactor's characteristics is the component related to uncertainty of nuclear data.

For the considered 1000 MW electric capacity reactor with non-positive close to zero VRE the calculation errors of the discussed functionals were (in % from K_{eff}): δK_{eff}^{nom} = 2.3%, δK_{eff}^{unf} = 2.3%, δ VRE = 1.2%, δ RL = 0.27%. The values obtained are close enough to those characteristic of fast reactors with a sodium coolant, excluding VRE and RL errors, which, correspondingly, are approximately 3-4 and 1.5 times as high. The higher error of VRE is due to a more significant change in spectral properties of the considered reactor when a hydrogen-containing coolant is removed.

The main subject of the paper is the analysis of components of the calculation error of reactor reactivity effects.

The greatest contribution to the error K_{eff}^{nom} (i.e. the error in estimated determination of a specific critical load) is made by $\sigma_f\,^{239}Pu$-16%(+0.33); $\sigma_c\,^{238}U$-15%(-0.24); $\sigma_{in}\,^{238}U$-15%(-0.07); $\sigma_f\,^{235}$-14%; (+0.57); χ-13%(1.0). In the unflooded state the error K_{eff} is determined by the same constants but there takes place a certain redistribution of their contribution due to the spectrum deformation: $\sigma_f\,^{239}Pu$-26%(+0.43); $\sigma_f\,^{235}U$-22%(+0.69); $\sigma_{in}\,^{238}U$-20%(-0.08); χ-12%(+1.0); $\sigma_c\,^{238}U$-10%(-0.18).

In the reactor unflooding is characterized by the decrease in a fraction of processes in resonance region of neutrons energy. Correspondingly, the main contribution to the VRE error is made by constants of radiation capture and fission of "resonance" nuclides:

$\sigma_c\,^{238}$-32%(-37); $\sigma_c\,^{239}Pu$-14%(-41); $\sigma_c\,^{240}Pu$-13%(-25); $\sigma_f\,^{235}U$-9%(-85); $\sigma_f\,^{239}Pu$-8%(-67); σ_c of fission products - 15% (-18).

The RL error is, mainly, determined by: σ_c of fission products -51%(+84); $\sigma_c\,^{238}U$ - 22%(-2.57); $\sigma_c\,^{240}Pu$ - 17%(-0.88).

Radiation capture cross-sections of ^{238}U, ^{240}Pu, ^{239}Pu are needed to be corrected, above all, in the resonance energy region. To make more precise the burnup effect calculation the fission product radiation capture cross-sections should also be corrected.

3. THE SAFETY PHYSICS OF VARIOUS TYPES OF REACTORS

Light-water reactors

It is expected that cores of advanced reactors of VVER type will use more and more burnable absorbers to compensate a reactivity margin, to flatten a power distribution field over the core volume, to sustain a temperature reactivity coefficient at a present level and to satisfy nuclear safety requirements.

One of key questions of the problem is the reliable prediction of nuclear physical parameters of (UO_2-Gd_2O_3) - absorbers, which is complicated by the following:

- anomalously high resonance neutron absorption by ^{155}Gd, ^{157}Gd;

- an overlap of neutron absorption resonances of ^{155}Gd, ^{157}Gd and ^{238}U.

The complex spectral character of the problem on determination of physical parameters of cells with (UO_2-Gd_2O_3) - obsorbers and its surrounding cells with UO_2 - fuel elements is related to the necessity of a detailed allowance for a spatial-energy distribution of neutrons practically within the entire energy band.

The reliable prediction of nuclear-physical parameters of (UO_2-Gd_2O_3) - absorbers is necessary for studying changes in time of a spatial-energy neutrons distribution in the absorption zone and for determining a residual negative reactivity caused by formed strongly absorbing gadolinium isotopes.

The investigations are carried out on the VVER subcritical facility at the IRT test reactor of the Moscow Engineering and Physical Institute. The VVER uranium-water lattice was used with a pitch of 12.70 mm, enrichment of 6.5 mass % of ^{235}U (UO_2 - Gd_2O_3) pellets are prepared by pressing and sintering powders of mixed oxides having homogeneity at a molecular level. Geometric sizes of (Zr-Nb) - clads and (UO_2 - Gd_2O_3) - pellets correspond to sizes of clads and UO_2 - pellets of VVER fuel elements.

The experimental procedure is based on the use of the activation method applying of γ-spectrometry and β-count.

The results of measurements of spectral indexes and reaction rate distributions are compared with analogous results in a regular lattice and can be used as a test for calculation procedures and applied nuclear data.

The use of (UO_2-Gd_2O_3) - absorbers is considered as one of trends of iptimization of VVER reactors fuel cycle. The application of (UO_2-Gd_2O_3) - absorbers in operating conditions ensuring an increased burnup and a decrease of neutron leaks from the reactor core, reduces radiation of the reactor vessel, improves uranium utilization and decreases the cost of nuclear fuel cycle.

Fast reactors

Research of the physics of LMR reactors safety are carried out for configurations of traditional homogeneous and heterogeneous types; search [7] conceptual investigations are performed with non-burning non-traditional lead-base coolant.

Adequacy of models of core heterogeneous configuration dispurtions (in-assembly heterogeneity [8]), formations of secondary critical masses are checked.

The safety of designed fast reactors is determined greatly by the value of positive reactivity arising at a partial or full removal of sodium from the reactor core. Such a situation can be feasible in using a mixed plutonium fuel, metallic fuels in the core and while increasing the core sizes. In this connection the experimental and calculation studies of sodium void effect of reactivity (SVRE) have been carried out recently on critical BFS facility for a number of critical assemblies with simple geometry but different in composition reactor cores; BFS-49 (plutonium mixed fuel), BFS-55 (plutonium metal fuel). The purpose of the work - the experimental assessment of reactivity effects occuring when sodium is removed; as well as the verification of calculation codes, procedures and constants used in design calculations to substantiate the reactors safety.

In the experiment the SVRE is determined as a difference in states of reactivity of a critical assembly with sodium and without sodium in the given volume.

In practice the sodium removal is performed by replacing sodium blocks by empty boxes in fuel rods (tubes) BFS. Since in the last few years the fuel assemblies (FA) of BN-600 type have been produced for BFS both with uranium oxide fuel and with metallic uranium-238, new opportunities made their appearance in experimental assessments of SVRE by comparing reactivities of two FA (filled with sodium and emptyone).

The SVRE value was determined in the center of critical assemblies studied. For the BFS-55-1 critical assembly the experiment was performed on a removal of all sodium blocks with a replacement by empty boxes for 37 central fuel rods (10% of the core volume). In this experiment up to 21 kg of sodium was removed from the critical assembly. For this case the calculation/experiment ratio is 1.7.

In the other experiment on the BFS-55-1 critical assembly the SVRE value for the central sodium column in the reactor core was estimated by measuring the critical assembly reactivity when two FA of the BN-600 type with metallic uranium-238 were put in it by turns. The SVRE value obtained in this way amounts to $10.6 \cdot 10^{-5}$ $\Delta k/k$ at the removal of 1 kg of sodium.

Of certain interest is the study of SVRE in the BFS-53-1 critical assembly - the reactor model with hybride core in the lower part of which a metallic uranium fuel is used and in the upper part - oxide uranium fuel.

The results point to the increase of SVRE value by 1.2-1.4 times in calculations for small central volume in models with small (up to 1 m^3) cores with a mixed fuel. In cores with metallic fuel the calculation/experiment derivations is higher than in cores with oxide fuel. The SVRE value measured in the core central part being a tenth part of the core, is 1.7 time less than the calculated one.

Such discrepancies require a further perfection of calculation procedures and means for analysis of experimental data [6].

To substantiate the safety of large FBR with in-assembly heterogeneous (IAH) oxide-metallic configuration of the core and various composition of blankets the calculation analysis of reactivity effects (RE) was performed. The task was undertaken to compare the sodium void and Doppler RE for a high capacity BN with uranium and thorium blankets and to assess the impact of their changes on certain characteristics of reactor safety. As a raw material for blunkets use was made of: a depleted uranium, thorium oxide and metallic thorium.

Excursions of temperature were determined with the aid of asymptotic reactivity balance method for some severe accidents (failure of heat removal to an external circuit, insertion of excessive reactivity) without scram. In the case of failure of heat removal from the primary circuit at a rated power without scram the asymptotic increase in the coolant temperature can cause a sodium boiling for the considered variants of reactors. An insertion of thorium into blankets leads to some decrease in an emergency heating-up of the coolant. Heating-up of coolant in an accident of transient over power without scram is decreased when thorium is used in blunkets but the heating-up level remains markedly lower than in the case of accident with heat removal failure.

The thorium insertion into fast sodium reactor blankets is one of possible ways of impact on RE and other safety characteristics.

<u>High temperature thermal reactors</u>

A high-priority work is urgent on generalization of calculated-methodical and experimental studies of specific features of HTGR with spherical fuel elements, for example, two reactor types developed in the USSR: the reactor of module VGM type and the reactor of integral VG-400 type.

The studies are aimed at a creation of enhanced safety reactors with high self-protectiveness against severe accidents. The VGM module reactor can comply to the full extent with these requirements [9].

One of specific features of the VGM reactor is the location of controls rods outside the reactor core - in a radial blanket, which resulted in the development of appropriate non-diffusion calculated methods and codes to determine their efficiency implemented in the code system (CS) "Kristall" (Crystal).

The code system "Kristal" is intended to carry out neutron-physical calculations of reactors with a complex configuration, taking into account a heterogeneous structure, with the possibility for every of separated reactor's region to use various approximations of the neutron transport equations (combined calculated schemes). At the present time within the framework of CS "Kristall" the code modules have been implemented and tested which ensure the use of the diffusion approximation, the integral method of first collision probabilities and the method taking into account the angular dependence of neutron flux density within the scope of the collision probability method [10]. Evaluation tests of calculated methods and codes used for HTGR were performed on critical assemblies with spherical fuel elements [3]. Further program of investigations of the HTGR physics has been developed, which is aimed at the refinement of reactivity effects, efficiency of controls and reactivity balance.

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REVIEW OF FAST REACTOR PHYSICS ACTIVITIES IN 1988-1989.

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Paper for Presentation at the
32 nd NEACRP Meeting

BR-10 REACTOR.

In January 1989 30 years had passed since the day when the BR-5(BR-10) reactor was put into operation.

The BR-10 reactor operation with the fourth core load of the uranium mononitride fuel continued. Since 1983 the reactor had been operating with this load at various power levels for 28 thousand hours, the uranium burn-up reached 8,3% h.a.,thus exceeding the design value (8,0%). In 1988 the first stage of investigation has been completed for the fission yield from fuel elements with various fuels and with simulated defects of their cladding.

In 1988 a "hot" sodium loop of the primary circuit was put into operation, where a tritium detector, a manganesse trap made of nickel and an operating section with samples were installed. The first measurements of tritium activity were performed in the circulating primary sodium.

Tritium content was also measured in the sodium samples from the primary circuit.

Started in 1980, the B-3 fast neutron beam irradiation of oncological patients continued. For three and a half years about 130 patients have been irradiated.

The fission yield of the defective fuel elements during the steam-water washing of fuel subassemblies and its long storage in water have been investigated.

A number of modifications has been introduced in the plant improving the safety of its operation.

.In the reactor core and in the reflector and thermal column channels test fuel elements, absorbing elements and structural materials are being irradiated. The B-3 beam irradiation of biological objects (blood, seeds, animals) are being conducted.

CRITICAL FACILITIES.

Durung the past year the investigation of neutron and physical parameters of LMFBR has been carried out at critical facilities. The goals of this reseach were as follows:

-to investigate the main functionals determining the neutron balance (k_{eff}, average capture-to-fission ratio, central reactivity worth ratio);

-to investigate sodium void effects, hydrogen introduction effects of power distribution and some problems of neutron streaming into the radial blanket of a special composition;

-to complete the experiment SFINKS (measurement of the average ^{238}U capture-to-^{239}Pu fission ratio in the thermal colomn) and to prepare for the joint β_{eff} measuring experiment.

Three critical assemblies were mounted at the BFS-55. The BFS-55-1 assembly consisted of metallic plutonium and uranium, sodium and steel and was a single-zone system. The blanket was made of metallic uranium (10cm) and uranium dioxide. The BFS-55-2 assembly had two zones. Into the central zone ($\sim$60% of the total volume).a great amount of zirconium was added and the fuel enrichment changed.

The BFS-55-1A assembly core composition was identical to that of the BFS-55-1 assembly but its blanket was all-uranium dioxide.

The investigation of these assemblies allowed us to make certain couclusions about the BNAB-78 neutron data accuracy in the field of spectra and compositions characteristic of LMFBR.

New data on the zirconium influence on neutron chracteristics of such fast reactors have been obtained.

The investigation of the BFS-50 critical assembly at the BFS-2 facility had been accomplished by the end of 1987.

The latest BFS-50-5 configuration is a three-zone model of the BN-800 reactor in the mean-stationary state with the low enrichment zone containing metallic uranium and plutonium.

The main objective of the BFS-50 research programme was to investigate the possibility of simulating fast plutonium reactor characteristics in the uranium model. For this reason at the stage of BFS-50-5 investigation the analysis of the research data oncharacteristics of the BFS-50-5 modifications (plutonium fuel in the low enrichment zone) and that of the BFS-50-4 modifications (uranium fuel) was performed. The comparison of the results shows that the k_{eff} prediction can ve successful (BFS-50-5 and BFS-50-4 are critical at the same geometrical parameters) and has the same uncertainty ($\sim$0.5% Δ k/k) for both modifications with allowance for inadequacy of a critical assembly and its design model.
Experimental distributions of fission rates within experimental errors coincide for both modifications. The value of the "design-experiment" difference in control rod worths for the BFS-50-5 modification appeared to be twice as high as that for BFS-50-4, the experimental values, as they are, being in good agreement for both modifications.

Upon the completion of the BFS-50 programme at the BFS-2 facility since the end of 1987 the research of the BFS-52 assembly has been initiated.

At the "KOBRA" facility during 1988-1989 two critical assemblies, KBR-14 and KBR-15, with central test zones of U-235 and chromium were investigated.The experiment aimed at verification of chromium group constants accuracy accepted in our BNAB-78 library.

The difference between these assemblies was in various degrees of neutron spectrum hardness in the central test zone, caused by the proportion of the U-235 and diluent (Cr, U-238) nuclei. For instance, if in KBR-14 the concentration ratio C_{cr}/C_{u-5} is equal to 81,5 and the ratio $C_{u-8}/C_{u-5}=4,2$, in the KBR-15 assembly these values are equel to 200 and 0,11, respectively. So, the neutron fraction absorbed by the chromium nuclei and thus, the valid information from the experiment on the chromium capture cross-section accuracy were different as well: about 16% in the KBR-14 assembly and 36% in KBR-15.

In the process of the experiment the following parameters were measured: the K_∞ values for the composition of central test zones, and central reactivity worth coefficients of separate elements (in particular, structural ones: iron, chromium and nickel). In experiments the oscillating method was employed together with the digital microcomputer-based reactimeter.

The experimentally obtained results differ significantly from the corresponding design values. For instance, the discrepancy of the K_∞ values was equel to 5,1% for the test zone composition of KBR-14 and 11% for KBR-15. The corresponding discrepancies were found for the chromium central reactivity worth coefficients. The experimental values ρ_{cr}/ρ_{u-5} in KBR-14 are approximately by 27% smaller than the design value and in KBR-15 this discrepancy reaches almost 50%, thus completely accounting for the discrepancy in the K_∞ values mentioned above.

The fact, that in the design the chromium group constants, compiled in the Nuclear Data Centre (IPPE) were used instead of the cross-section values from the BNAB-78 library, made it possible to reduce the discrepancy in the K_∞ values to 2% and 3% for KBR-14 and KBR 15, respectively. However the discrepancy remained requires further analysis of chromium group cross-sections.

As for the reactivity worth coefficients of other structural elements-iron and nickel, we should point out a rather good agreement of the experimental and calculated results for KBR-14 (within 5%) and a significant overestimation of the calculated values in KBR-15: 17% for nickel and 22% for iron. Wheras the nickel data discrepancy can be accounted for by inaccurate cross-section values for this element in the BNAB-78 system (this fact is confirmed by the experiments at the previous assemblies), the results for iron somewhat unexpectedly appeared to run out beyoned the interval ± 10%, which implies the experiment-design discrepancies for all the six assemblies investigated earlier.

Summing up the latest investigation results we can conclude that nickel and chromium cross-section values compiled in the BNAB-78 library require significant correction in the direction of reducing the capture cross-section values. Iron cross section values can be considered satisfactory for the reactor design.

DESIGN AND EXPERIMENTAL INVESTIGATIONS OF THE BN-1600-TYPE REACTOR HETEROGENEOUS CORE MODEL.

Investigations of the BFS-46 critical assembly simulating the BN-1600-type reactor heterogeneous core, which had been performed earlier at the BFS-2 facility, showed expediency of further study of heterogeneous core physics with regard to development of more adequate models and application of more advanced design methods and codes.

For this reason in 1987 the investigation of the BFS-52 series critical assemblies was initiated, which simulated a rather representative annular-type core model of the BN-1600 reactor.

The BFS-52-1 critical assembly was the simpliest one among all the assemblies of the BFS-52 series. It was a one-zone assembly, its low enrichment zone composition was identical to thatof the following assemblies of the BFS-52 series, it contained neither simulated control and safety rods systems, nor internal breeder zones.

The subsequent assembly BFS-52-2A had: a low enrichment zone (LEZ),a high enrichment zone (HEZ) and three annular internal breeder zones (IBZ).The change-over from the BFS-52-2A assembly to BFS-52-2B/2 was carried out by means of introduction of a central breeder zone in the centre of the core and subsequent transfer into a critical state caused by the rupture of the first and the third rings of IBZ, with IBZ modules being replaced by LEZ module (for the first ring) or by HEZ modules (for the third ring).

For all these modifications were made design and experimental investigations of critical parameters and the reaction rates distribution along the core radius and height. For the BFS-52-2B/2 assembly the isolated mock-up worth of control and safety rods systems was investigated both in the centre of the core and in its periphery.

For the most simplified homogeneous one-zone BFS-52-1 assembly the discrepancy of calculated and experimental data in K_{eff} is equal to 0,65-0,82% Δ k/k when using the design packages in two-dimentional (R,Z) geometry.

Upon changing-over to a more complex heterogeneous BFS-52-2A critical assembly with application of the same design packages this discrepancy appeared to be 1,04-1,19% Δ k/k, as for BFS-52- 2B/2 assembly, which has a more contiguous structure because of the ring "ruptures", here the discrepancy between the design and experimental values of K_{eff} is the greatest, that is 2,1-2,2% Δ k/k. Design of this critical assembly in three-dimensional hexagonal geometry enabled to take into account the real structure of the assembly and thus made it possible to obtain the differency from the experimental K_{eff} value being equel to 1% Δ k/k.

The design K_{eff} values in all the cases underestimate the experimental values, this is characteristic of the BFS facility critical assemblies.

The agreement between the experimental and calculated results can be considered quite satisfactory if we take into account that the error due to the experimental conditions is equal to $\pm$ 0,30% Δ k/k for the given heterogeneous assemblies, and the constant component of the design error is $\sim$ 1,5% Δ k/k.

The experimental and the design values of the U-235, U-238 and PU-239 fission rate distribution for the BFS-52-1 assembly showed a good agreement (along the core radius the discrepancy does not exceed 3,5%).

For the BFS-52-2A and BFS-52-2B/2 assemblies this discrepancy increases significantly, for instance for U-235 and PU-239 in HEZ it reaches 30-35%, the design values being below the experimental ones.

Such discrepancies are caused by the application of diffusion design methods when analyzing these assemblies at the given stage of investigations.In the future more accurate design methods will be applied.

The design values of the simulated control and safety rods system worth were determined by the codes in three-dimensional hexagonal geometry.

For simulated boron rods located in the central part of the core the design and experimental values of their worth showed a good agreement,within 5%. For comparison, if we take the worth of the eccentrically located simulated control rods,the discrepancy will increase up to 12%, the design values being below the experimental ones.

At present the design and experimental investigations of the BN-1600-type heterogeneous core with annular heterogeneity are still in progress.

IMPROVEMENT OF DESIGN METHODS AND CODES.

To make systematic investigations in the field of group constants preparation from the evaluated nuclear data files and investigations of the errors of reactivity effects calculation (due to the errors of group constants preparation),a package of codes "GRUKON-SPEKTR/ZERO", based on the"GRUKON" codes,was developed in 1988 in the Institute of Physics and Power Engineering. This package consist of two main parts-the"GRUKON-SPEKTR" package and the code "ZERO". The first part is intended for calculation of the group microconstantsfrom the files of the nuclear data evaluated. As the averaging spectra we can use either the standart spectrum or the spectrum resulted from the solution of the energy-detailed equation of neutron moderation in infinite medium ore of the energy-detailed set of equations in P_1-approximation. Besides, there is opportunity to get the energy-detailed spectrum of the value function and,accordingly, bilinearly-averaged group constants.

The second part of the package is meant for application of the file-based microconstants in order to prepare composition-dependent microconstants (if initial microconstants are obtained with the standart spectrum weight), microconstants and the design of the flux spectrum and importance of neutrons as well as their functionals in the zero-dimensional group model of the reactor.

The code package "GRUKON-SPEKTR/ZERO" enables to prepare group constants by different ways: based on the linear averaging, bilinear averaging and the averaging with spectrum perturbation weight. Accually the cross-sections being averaged can be presented both at the detailed level description and in the multigroup presentation. The latter opportunity can be realized by means of the 140-group library of the ZERO constants. Currently the code package described and the ZERO library are being tested.

The design-technique basis for the design analysis of the experiments at the BFS critical assemblies are under development. In order to take into account the influence of the spatial heterogeneity effects on the design reactor parameters, the "SORA" code was included into the code package "RADAR" to homogenize asymptotically macroconstants for the periodical slab lattices by the VPS method. By the spectial method "SORA" calculates efficient diffusion coefficients for the directions perpendicular to the lattice layers and along the layers of the slab lattice (D_{11}). This method results in the Bentz formula for D_{11}.

The effects of resonance heterogeneity are considered on the basis of the efficient use of equivalence correlations in asymetric multilayer cells.

The code package "RADAR" with the heterogeneous preparation of constants is widely used to evaluate the heterogeneity effects at the BFS critical assemblies. At present the identical method of cylindrical cells homogenization has been developed, the cylindrical version of the "SORA" code has been tested.

The design and theoretical investigations of hydrogen ractivity effects (HRE) in FR of the BN-1600 type were performed. Though the experience of the BN operation and design allows us to conclude that the hydrogen ingress is hardly possible, to investigate HRE is very important because the possible expected reactivity effects are rather considerable. Therefore the design investigations of HRE have been performed within a wide range of main core parameters changing: the core composition, isotopic plutonium composition, the number of fission fragments-at various localization of a hydrogenous substance in the core, significant regions of a positive effect, localized in the core periphery were found. The influence of the main neutron-absorbing nuclides on the reactivity effect was analyzed. We investigated the error of the design methods: applicability of the perturbation theory of the first order; one-dimensional design models, possibility of application of undisturbed axial backling values, influence of the group number. The physical model of a hydrogen reactivity effect was developed which allowed us to account for a great number of regularities revealed in numerical investigations.

IMPROVEMENT OF NUCLEAR DATA FOR FAST REACTOR DESIGN.

Critical analysis of the information contained in the first version of the evaluated neutron data library has been performed recently. For the on-line file correction a dialoque procedure editor of the estimated data files has been developed at the ES-1061 and RS/AT computers. The resonance parameters for all isotopes of chromium, wolfram and tin were revaluated. The influence of uncertainty in the U-238 neutron force function on the resonance structure calculation in the unresolved resonance region was estimated.

For the main reactor materials the evaluated neutron data files were transformed into the multigroup constants system by means of a code package NJOVEO and PPP GRUKON. By this constants system the main physical parameters were calculated for the assemblies included in the library of the integral experiments evaluated. The discrepancy was found between the design and the experiment which was twice or even three times as large as that according to the constants system BNAB-84 (modification of BNAB-78). At present the formation of the 28-group constants version (BNAB-89), totally based on the estimated data files is about to be over.

We obtained the power distribution constants in the 28-group BNAB break-down for 94 nonfissionable and 28 fissile nuclides. Moreover the constants take into account the power distribution of the neutron radioactive decay. Based on the results obtained a computer data library was formed concerning the local and total (with allowance for γ -quanta) power distribution at elastic, inelastic scattering, capture and fission for all the nuclei from the constants system BNAB-84 and corresponding group kerma-factors were formed.

The experimentally-based analysis of measuring neutron spectra of CF-252 spontaneous fission and U-235 and PU-239 fission neutron spectra was made. It was concluded that for the fission neutron spectra description it was better to use the Watta destribution. Consideration was given to the influence of different evaluations of fission spectra on the results of calculations of K_{eff} and other functionals for fast critical assemblies.The average energy for the PU-239 fission spectrum was found to be by 3% higher than in BNAB-78. It results in increasing the K_{eff} values of the fast breeder approximately by 0,6%. The average energy of the U-235 fission neutrons, affected by thermal neutrons turned out to be $\bar{E}$=1,98 $\pm$ 0,03 whereas in ENDF/B-$\underline{V}$ $\bar{E}$=2,03 Mev.

The constant component of the various parameters design error was assessed with allowance for fuel burn-up. Calculations were made for two models of fast reactors of the BN-1600 and BN-800 type. We calculated the group constants sensitivity factors and the errors of the following parameters: multiplication factors at various time of the reactor life-time, reactivity loss during the reactor life-time, breeding ratio, power density, safety parameters, such as sodium voids and Doppler temperature reactivity effects. We considered in detail the value of various components in calculation of sensitivity factors with fuel burn-up. It was shown that for the majority of reactor parameters the accuracy achieved is close to that required. As for the parameters of criticality and the reactivity loss during the reactor life-time, the accuracy of these values should be two or three times higher.