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Validation of Calculational Procedures for the Design of Light Water Tight Lattice Reactors (LWTLR) with Epithermal Spectrum.

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1. Introduction.

As shown in preceeding papers<1,2> of this seminar, the prediction of the behaviour of nuclear reactors in use nowadays can be performed with good accuracy. The applied calculational procedures, however, are not the same for the different reactor types e.g. for fast or thermal reactors. Since about 10 years increasing interest can be observed concerning the design of light water reactors with very tight lattices (LWTLR). The incentive of these investigations is to increase the conversion ratios from fertile into fissile fuel (CR) in the established light water reactors LWR in order to obtain an improvement of the fuel utilization in a closed fuel cycle. Both the thorium/U233- cycle <3,4> and the uranium/plutonium- cycle are investigated. Most of these investigations are concentrated on the U/PU- cycle. The main reason for this choice is the availability of the technical facilities for this fuel-cycle and the availability of plutonium from spent LWR fuel. This paper only will deal with the U/Pu-cycle.

Tightening the lattice in a light water reactor leads to a strong epithermal neutron spectrum. This spectrum hardening enables, together with the use of plutonium, good conversion ratios and a significant improvement of the fuel utilization. Especially Pu241 has very good fissile properties, characterized by the number of fission neutrons per absorption, ETA (Figure 1).

However, the use of plutonium fuel in a harder neutron spectrum not only improves the conversion ratios, but also leads to a change in the behaviour of the reactor core after coolant density changes. It is well-known, that fast reactors have a positive void-effect with consequences for the licensing procedures and for the safety-systems. It has been found that in the case of using realististic fuel in a LWTLR, e.g. plutonium from PWR-fuel after 33 GWD/THM mean burnup and about 10 years off-core and reprocessing time, similar problems may occur<5>. Because one of the main objectives of the LWTLR- development is to maintain Light Water Reactor (LWR)- characteristics<6>, also e.g. a sufficiently negative reactivity-effect after decrease of the coolant density, the voiding effects must be

predictable with a sufficient accuracy. This means, that both cores with an epithermal and a fast neutron spectrum have to be predicted accurately with the same calculational procedures, starting from the same nuclear data base.

In the next chapters brief discussions of the LWTLR- characteristics, of calculational procedures for fast and thermal reactors, of the development of new procedures for the LWTLR at the Kernforschungszentrum Karlsruhe (KfK) and of validation investigations for LWTLR- systems will be presented. More detailed information may be found in reference<7>.

2. Characteristics of LWTLR- systems.

The main purpose of the LWTLR- development is to increase the conversion ratio of a light water reactor (LWR) by hardening the mean neutron spectrum in the core. This means that the properties of a thermal reactor move into the direction of the characteristics of fast breeder reactors (FBR). From this point of view it seems significant to compare the main properties of LWR and FBR and to try to estimate the potential of the LWTLR within this range. Most of the LWTLR- investigations concern pressurized water reactor (PWR)- modifications (advanced PWR , APWR). Table 1 shows some comparisons of an APWR with LWR and FBR. Typical numbers for mean enrichment, for burnup and for conversion ratios are given.
In our opinion the lower two rows characterize the main problems for an APWR: possible positive void reactivity and recriticality configurations after severe malfunctions. Until now we did only a few estimates for recriticality cases. The void problems, on the other hand, have been analyzed in more detail. It has been found, that the void behaviour of an APWR is very sensitive to the special design considered<5>.

Concerning the calculational tools for PWRs and FBRs it is known that both reactor types can be predicted very well, but with different approximations for the solution of the basic equations for the calculation of zone- averaged cross sections. One of the first tasks starting LWTLR- investigations must also be to check the capability of the available calculational procedures for this reactor, which requires both the predictibility of the operating case (with epithermal neutron spectrum) and the voided configuration (with very hard fast neutron spectrum).
The Figures 2 and 3 give an impression of differences between the reactor types PWR, APWR and FBR. In Figure 2 a comparison of the flux per unit energy is shown. The calculations have been carried out in the 69 WIMS group structure<8>. The influence of the large plutonium resonances below 4 eV is observed clearly in a Pu-fueled APWR and FBR. This picture illustrates the importance of a proper treatment of the resonance absorption in an APWR; an APWR has its flux maximum there. In Figure 3 the differences between the reactor types are shown in another way. The cumulative reaction rates for capture and fission demonstrate the different importance of the thermal, the resonance and the fast energy regions for each reactor type. For these reasons it was not sure in advance that one of the established methods for LWR and FBR are useable for design calculations of an LWTLR, particularly with respect to the investigations

for voided configurations.

3. Analysis of calculation procedures.

The neutronic reactor calculations are usually separated into a few main tasks:

a) Preparation of mean macroscopic cross section sets for all reactor zones, both for the operating reactor and for the voided case.

b) Calculation of time-, space- and energy- dependant neutron fluxes in the reactor, including the associated reactivity values.

c) These calculations are often followed by evaluations, e.g. for the determination of (zone dependant) conversion ratios or power density maps.

Only the first task shows large differences between the solution procedures for LWRs and FBRs. For whole-core calculations the same methods can be applied for both reactor types, e.g. two- or three-dimensional diffusion theory in a coarse multigroup scheme.

The zone group constants for these reactor calculations, e.g. for capture, fission, scattering etc., are determined by preserving the reaction rates in these zones. The group constants for the common reaction types can be obtained by simple flux weighted averaging. The problem is, that no proper general flux guess can be made for this averaging "a priori". Using specific properties of the reactor types, different approaches have been established for this flux guess in FBR- and LWR- investigations.

3.1 Approximations for fast reactor calculational procedures.

For a fast reactor with hard neutron spectrum and large mean neutron free path it is assumed, that the space- und energy-dependant flux may be separated into the product of space- and energy- dependant components. This simplifies the problem in such a way, that the weighting flux becomes only energy- dependant. A further assumption is usually a weak energy- dependance of the collision density, flux times total cross section. In a standard calculation all materials of a reactor zone are homogenized and group constant sets may be calculated straight forward with the exception of the resonance region.

The calculation of the group constants in the resonance region is a complex problem because of the strong space- and energy-dependant selfshielding effects.

In our standard FBR- method spatial selfshieldung is not taken into account. For the energetic selfshielding several approximations have been introduced. In the FBR- solution at KfK the narrow resonance (NR) approximation is applied, stating a slowly varying collission density in the neighbourhood of the resonances. Lumping- and shadowing- effects are not important and therefore

they are not taken into account. The effective group constants in the resonance region are obtained from interpolations in precalculated resonance tables following the so called σ- concept as introduced by Abagyan et.al.(9). More detailed information about the standard FBR- cross section calculation code GRUCAL may be found in ref. (10).
For the LWTLR- investigations this FBR- methods have been modified on the basis of equivalence theorems for heterogeneous and homogeneous reactor fuel zones. These modifications concern the determination of the σ- value for the interpolation in the resonance tables. The corrections applied are well-known in the literature, see e.g. reference (11):

a) the lumping effect of the fuel rod by the volume-to-surface correction.

b) shadowing effects of the fuel rods by the Dancoff- correction as proposed by Sauer(12) and Williams and Gilai(13).

3.2 Approximations for thermal reactor calculation procedures.

For a thermal reactor the mean neutron free paths are smaller than the reactor zones and it is assumed that the flux guess may be obtained from the space- and energy- dependant flux distribution in a representative cell of this zone, placed in an infinite array of such cells. Usually a one dimensional many group calculation in transport approximation (discrete ordinates method Sn or collission probability method CPM) is performed to generate this weighting flux.

The thermal reactor codes available for our investigations were HAMMER(14) and WIMS/D(8). Both systems have code-own data libraries, containing older nuclear data; they predict thermal systems rather well. The resonance treatment in these codes is different. Whereas HAMMER calculates the weighting spectrum iteratively as proposed by Nordheim(15), WIMS/D applies resonance tabulations with interpolation schemes based on equivalence theorems, similar to our modified FBR- methods. However, compared to ours, the WIMS/D interpolation scheme is more refined (intermediate resonance approximation IR, Bell- factor).

Although the program HAMKOR(16), developed from HAMMER, may calculate the depletion of fuel in a PWR satisfactory compared with experiments, we observed that this system is not suited for reactivity investigations of APWR- systems. For that reason only a few comparative calculations have been performed with this program.

The WIMS/D code showed good agreement for the tight lattice experiments we have investigated (see chapter 5 and ref.(17)), but calculations for the voided case, starting from the same data base, have large discrepancies compared with the results of reliable FBR- codes(18) (Figure 4). Because these FBR-codes have been checked intensively with results of experiments with hard neutron spectra(19), we believe WIMS/D is not adequate for studying voided core configurations. Therefore one has to be sceptic with the interpretation of publications, using WIMS/D for such purposes.

4. KfK- code development for LWTLR investigations.

Benchmark investigations in collaboration with the University of Braunschweig (TUBS) and with Kraftwerkunion (KWU)<17> and further extensive analysis of calculational procedures showed that at KfK no adequate tools were avalaible to calculate both normal and voided LWTLR- systems with sufficient accuracy with one code system.

The modified FBR- codes for homogenized reactor zones cannot be applied for the normal reactor condition because the plutonium resonances in the low eV. energy region cause strong space-dependant shielding effects. For this reason the basic assumption for the homogenization, separation of space- and energy- dependance, is not valid. Figure 5 shows a comparison of the fluxes in the unit-cell of a FBR- zone and an APWR- zone. Whereas the synthezation of the space- and energy- dependance for the FBR-cell obviously is valid, in the case of the APWR-cell it is certainly not allowed.

As pointed out before, the available thermal code WIMS/D was not able to predict the voided lattices with sufficient accuracy.

Two alternatives for the improvement of our calculational tools for LWTLR- applications were discussed:

A) improvement of the data library in a thermal code, e.g. WIMS, or
B) establishment of thermal code features in the FBR- system.

Because our FBR- codes, collected in the program system KAPROS <20>, are very powerful and flexible we decided to introduce the main advantages of the WIMS/D code into KAPROS.

These improvements are:

a) establishment of a 69 group constant library with the energy group structure of WIMS, but based on the newest nuclear data at KfK. This group structure includes the treatment of upscattering.

b) introduction of a slightly modified version of the one-dimensional collission probability method (CPM) from WIMS for cell-calculations.

The characteristics of these developments are:

4.1 The 69 group libraries G69COLD and G69HOT.

a) Energy structure: 69 WIMS groupboundaries<8>. One of the advantages of this structure is the good resolution of the first resonances of Pu239, Pu240 and Pu241 up to 4 eV. Selfshielding for fuel with significant Pu- content can be described satisfactory.

b) Scattering matrices: in the thermal region below 4 eV the original WIMS/D matrices are used, otherwise new calculated data from KEDAK-4<21> with the help of the standard FBR- codes using the collission density spectrum of a typical APWR- lattice for the weighting (MIGROS3<22>, MITRA<23> and GRUMA<24>).
The upscatter matrices for hydrogen and oxygen at reactor opera-

tion temperatures were calculated with the code NJOY(25), starting from an ENDF/B-V library(26).

c) All other group constants are recalculated from KEDAK-4 with the procedures mentioned above. For a large number of fission products, group constants have been provided by ECN- Petten(27), which are comparable in quality or even better than ENDF/B-V evaluations.

d) The data storage has the same format as the FBR- libraries, that means separate data types are used for the different degradation and emission cross sections as elastic and inelastic scattering and n-2n reactions. This data separation is advantageous for the calculation of systems with hard neutron spectra, where these reactions are important. They may be changed and adjusted separately if necessary. In most thermal codes, e.g. WIMS/D and HAMMER, only one tranfer matrix is applied for all degradation processes.
Other advantages of the use of the FBR- storage mode are that existing FBR- codes for the calculation of zone dependant macroscopic cross sections can be applied with only small changes for the better treatment of the heterogenities in these zones and the possibility to use the numerous available evaluation programs.

A disadvantage at the beginning was that the modified FBR- codes could not treat temperature dependant transfer matrices. It has been observed, that at least two matrices for the moderator materials must be used for cold and hot lattices, especially for burnup studies.
Two libraries which only have differences in the WIMS thermal energy region are available.

A) The library G69COLD contains matrices at about 300K below 4 eV. The group cross sections for the heavy nuclides are determined at 300K. This library is used for the recalculation of experiments at room temperature.

B) The library G69HOT contains the matrices for the mean temperatures at operational level (580K for hydrogen and 700K for oxygen) and cross sections for the heavy nuclides at 900K. This library is used for the APWR- design calculations at nominal power.

In the meantime the standard FBR- code GRUCAL for the calculation of macroscopic group cross sections has the capability to treat temperature dependant upscattering. At the present time, the generation of a 69-group library with temperature dependant scattering matrices is in progress.

4.2 The one dimensional collission probability program WEKCPM.

The commonly used cell program in WIMS is a one dimensional collission probability program derived from the THESEUS code(28). Only a few modifications were necessary to adopt the WIMS program in the KAPROS system. These modifications in WEKCPM are related to the cross section preparation for the interface files and to the handling of the program input. The data transfer of the results to other KAPROS- programs is organized in the same way as established before in the discrete ordinates program ONETRA, derived from

the Los Alamos code ONETRAN<29>. In this way comparisons between WEKCPM and ONETRA can be performed easily, e.g. for verification purposes.

4.3 Burnup calculations.

One of the most important LWTLR- parameters is the achievable mean burnup of discharged fuel, determined by the number of reloads for the fuel and the number of full power days per cycle. The prediction of the crucial parameter "full power days/cycle, FPD/cycle" showed large discrepancies in the first APWR- publications<7>, see also Table 2. In the meantime a trend to more uniform, accurate results for the burnup calculations can be observed<7>. These results are in good agreement with our published data<5,7>, based on calculations with the KAPROS- module BURNUP<30>.

The module BURNUP has been developed for FBR- burnup calculations <31> from the KORIGEN- code<32>. Slight modifications enable the application for thermal and epithermal reactors. BURNUP treats all transitions between nuclides. The data for these transitions are stored on module-own libraries like in KORIGEN. During a depletion calculation BURNUP checks the availability of system dependant one-group cross sections. In this case these data are used, otherwise the data from the library will be taken. The number of fission products used is defined by the code-libraries. E.g. the same number as in KORIGEN (ca 1200) may be taken into account, but as the other extremum, also one pseudo fission product pair, as often done for FBR- investigations.

With these extensions the KAPROS system contains adequate calculation modules for all tasks during the neutronic design of fast, epithermal and thermal reactors, including depletion. Particularly the expected accuracy for the most important parameters seems to be sufficient for comparitive studies. For the detailed analysis of a final concept and for detailed evaluations of LWTLR-experiments it may become necessary to improve both the resonance treatment and the weighting procedures for the cross sections in the epithermal and fast groups.

A flexible new procedure,KARBUS, KArlruhe Reactor BUrnup System, enables the automatized execution of these tasks and organises the data transfer between the codes. With a minimum of user input the following calculations can be performed: determination of number densities in reactor cells and in homogenized reactor zones, calculation of zonedependant macroscopic cross sections with (modified) fast reactor methods or from homogenized results of cell calculations with the collission probability method CPM or the discrete ordinates method Sn, calculation of zero-, one- ,two- and three- dimensional neutron flux distributions, including upscattering, calculation of power maps, conversion ratios and absolute fluxes, depletion and fuel management investigations. A more detailed description of the contents of the KAPROS- system and of the procedure KARBUS may be found in the KAPROS-own program descriptions and in reference<33>.

The organisation of the use of fast computer storage and of the slower external one is very flexible, so only the computer capacity determines the limits of the possible complexity of the problems. At least ca 2 MB fast memory must be available.

5. LWTLR- validation investigations.

5.1 Joined KfK, TUBS and KWU Benchmark investigations.

Because of the expected new spectral properties of LWTLR- lattices, the first task of our LWTLR- work was to try to validate the available calculational procedures. In cooperation with the University of Braunschweig (TUBS) and the Kraftwerk Union (KWU) we intended to perform benchmark investigations on relevant experiments. However, at that time (ca 1978), no relevant experiments with mixed oxyde,UO2PuO2, (MOX) in tight light water lattices were available. Only experiments with small PuO2-contents in wider light water lattices could be found in the SAXTON(34) and plutonium Utilization Program (PUP)(35). For UO2- lattices, on the other hand, relevant experiments could be found in the ZPR-7 program(36). So we decided to investigate also a theoretical lattice with typical APWR- properties. This theoretical lattice was proposed by TUBS. Both normal and voiding conditions should be calculated.

These joined benchmark investigations could be completed in 1982. The results are described in reference(17). The most relevant findings were:

a) The experiments with UO2 in tight lattices and MOX-fuel with small PuO2-contents in wider lattices could be recalculated satisfactorily, both by thermal reactor methods and by modified fast reactor methods. The standard fast reactor methods without treatment of the cell-heterogenities are not usable for the recalculation of these experiments.

b) The results for the theoretical lattice showed large differences between the applied methods, especially for void-calculations. A main part of these discrepancies could be explained by data differences (e.g. for inelastic scattering of U238).

5.2 Recent LWTLR- lattice experiments.

The discrepancies in the benchmark results and the increasing interest for the LWTLR- development demonstrate the need for relevant experimental data for tight light water moderated lattices with MOX- fuel with PuO2-contents in the order of 10-20%. Several activities have been initiated in order to perform such experiments, e.g. in the PROTEUS-facility at Wuerenlingen(37), the SNEAK- facility at Karlsruhe(38), the ERASME experiment and the MELUSINE experiment in France(39).
Whereas the Phase I experiment in PROTEUS and the SNEAK-experiment had to be carried out with not well suited fuel (developed for special FBR- experiments), the French experiments and the Phase II experiment in Wuerenlingen will use relevant MOX-fuel(40).

Until now only the PROTEUS- results have been published detailed enough for code validation work, e.g. refs ⟨37,41⟩. The main part of these results are reaction rate ratios for several moderator conditions: H2O, dowterm (simulating about 42.5% H2O-voidage) and the dry lattice.

The fuel used in the PROTEUS Phase I and in the SNEAK experiments was MOX with about 15% PuO2- content. The Pu- composition was 80% Pu239, 18% Pu240 and 2% Pu241. These values are not typical for the available fuel for LWTLR: 57.8% Pu239, 26.6% Pu240, 9.5% Pu240 and 6.1% Pu242.
The PROTEUS experiments have been carried out in a lattice with a mixture of these fuel rods and of rods with depleted uranium. With the ratio MOX:UO2 equal 1:1, the mean fissile enrichment was 6%, with the ratio 2:1, it was 8%. The SNEAK experiment contained a testzone with a lattice of MOX-fuel with 12% fissile enrichment.
Reference⟨40⟩ gives a detailed analysis of several investigations on the PROTEUS Phase I experiment. In refence⟨38⟩ first results of the SNEAK- experiment are given. More detailed analysis are in progress now. Table 2 shows some preliminary C/E- values for Keff. The core 12B was a fast system. The cores 12F1 and 12F2 contained different tight lattice testzones.

5.3 Validation of the KARBUS- procedure.

The main purpose for the development of the KArlsruhe Reactor BUrnup System KARBUS was to assure reliable prediction of the criticality for the whole range of fission reactor types, including fuel depletion and reactor control, and of the most important reaction rates (e.g. CR). In order to reach this goal it is necessary to validate the system by calculating as much reactor types as possible. Two types of validations have been performed:

a) Recalculation of available experimental data. This could be done for fast, epithermal and thermal systems.

a) Comparison with results of established methods for the specific reactor type considered. Fast and thermal code systems were available for this work.

5.3.1 Validation for fast reactors.

Because of the great experience with fast reactors (including voided configurations) at KfK it was rather easy to verify KARBUS for these reactor configurations and in this way also for voided LWTLR-systems.

a) Experimental verification was performed for:
   i) a simplified PHENIX- model⟨42⟩,
   ii) a BZA- experiment⟨43⟩,
   iii) a SNEAK- experiment⟨44⟩.

b) Code comparisons were performed for:
   i) KFKINR- set validation work⟨45⟩,
   ii) NEACRP- LMFBR benchmark investigations for a fresh core⟨46⟩ and for burnup behaviour⟨31⟩.

All comparisons show satisfactory agreement.

5.3.2 Validation for thermal reactors.

These investigations started 1984 with two comparison calculations:

a) Analysis of Kernkraftwerk Obrigheim (KWO)- type fuel burnup as carried out for the KORIGEN- verification(32). The first comparisons show satisfactory agreement.

b) Participation in a NEACRP- benchmark on recycling of reprocessed uranium(47). After the correction of some data deficiencies not yet approved, also satisfactory agreement can be observed with the first preliminary results of participants from five laboratories. This comparison concerns both reactivity values and actinide number densities.

5.3.3 Validation for epithermal reactors.

For this spectral region, LWTLR- type lattices have been analyzed.
As pointed out above both experiments and benchmark grids have been investigated. Until now KARBUS has been tested for a selected number of lattices:

a) the tightest single zone UO2- core (HIC-8) from(17),
b) the most representative MOX- experiment (core 7) from (17) and
c) the K-infinity value of the first LWHCR- experiment at EIR- Wuerenlingen(37). Because of the complex layout of the test-zone in PROTEUS with 2 different rodtypes, detailed reaction ratio comparisons are difficult to perform. Until now we did not establish a procedure for these calculations.

Table 3 shows a summary of the validation calculations.

With exception of the PHENIX- core all these results show satisfactory agreement for a wide range of reactor spectra with a tendency to slight underestimation of the reactivityof epithermal lattices. The initial conversion ratio (ICR) of the UO2- core can be recalculated much better than the results in (17). The 1.4% difference for Keff in PHENIX is related to the leackage from the reactor; the K-infinity values of the single zones show better agreement between the KFKINR and the G69HOT results. The origin of this effect is in investigation at the present time.

Although only a limited number of validation investigations have been performed until now, the results for reactor types with completely different spectra indicate that the present version of KARBUS is well suited for LWTLR- calculations. Also the depletion calculations, which showed large discrepancies in earlier publications, see table 4, have gained more confidence by the results of the NEACRP- benchmark calculations on recycling of reprocessed uranium, referenced before.

6. Summary and future work.

The LWTLR will have physics characteristics between the currently used LWR and FBR. The latter may be predicted with sufficient accuracy, but with different calculational procedures. Whereas LWRs do not have severe problems with negative voiding effects, the FBR has a positive void effect with consequences for the licensing procedures and for the safety-systems.
Because the LWTLR- designs in consideration now should be licensable with the current LWR- procedures, sufficiently negative void-effects must be guaranteed. For this reason both normal and voiding conditions of LWTLR- lattices must be predictable with good accuracy.

Analysis of available calculational procedures at KfK show, that neither the thermal reactor code WIMS/D, nor the modified fast reactor codes could fulfill these requirements.

The new developments at KfK combine advantages of established calculational procedures for thermal and fast reactors e.g.:

a) Groupboundary scheme with upscattering as in the thermal code WIMS/D.

b) Treatment of the degradation processes at high energies with separate data storage for elastic, inelastic, (n,2n)- processes, as usual realized in fast reactor codes.

Moreover, the large R/D- developments at KfK for the fast breeder project can be utilized for LWTLR- investigations. The new calculational procedure (KARBUS) is realized at KfK within the powerful FBR- calculational system KAPROS.

The validation-investigations for the KARBUS- procedure for fast, epithermal and thermal systems show satisfactory agreement for a selected number of comparisons with experimental data and with the results of established codes for special reactor systems. The burnup behaviour, which showed larger discrepancies in earlier studies, also seems to be predictable with good accuracy in KARBUS.
For these reasons we believe, at the present time KARBUS is a good tool for comparative LWTLR- investigations. For detailed calculations of a final design, further improvements may become neccesary.

For the near future the following improvements of our calculational procedures and of the experimental data are identified:

a) Refinements in the resonance treatment, e.g. by the introduction of a calculation module based on the Nordheim theory.

b) Refinements in the weighting procedures for the group constants in the coarse epithermal and fast energy groups.

c) Need for more representative experimental data. Desirable is a number of experiments with MOX-fuel with 10-20% PuO2-content and with a pitch-to-diameter ratio (p/d) from 1.10 to 1.30. The experimental setup should be as simple as possible. The measurements in wed lattices are more important than those in dry configurations. Lattice K-infinty values are important because one of the main problems for the LWTLR- design is the prediction of reactivity in the normal reactor condition.

7. References.

<1> W. Bernat
Data Validation for Thermal Reactors
KTG/ENS - International Seminar, October 1-2,1985 Bonn

<2> S. Pilate
Data Validation for Fast Reactors
KTG/ENS - International Seminar, October 1-2,1985 Bonn

<3> G.B. Webb, Editor
Conceptual Design of a Movable Thoria Finger Rod Controlled Light Water Breeder Reactor
WAPD-TM--1519 (1982)

<4> M.J. Schneider, Editor
Design Concept of an Advanced Light Water Breeder Reactor Controlled by Movable Fuel
WAPD-TM--1534 (1982)

<5> C.H.M. Broeders
Design-Einfluesse auf einige neutronenphysikalische Eigenschaften eines fortschrittlichen Druckwasserreaktors
Jahrestagung Kerntechnik Berlin, (1983)

<6> H.H. Hennies, H. Maerkl
Ueberlegungen zur Modifizierbarkeit eines LWR im Hinblick auf eine bessere Uranausnuetzung
Jahrestagung Kerntechnik Berlin, (1980)

<7> C.H.M. Broeders
Neutron Physics Investigations for Advanced Pressurized Water Reactors.
Nuclear Technology Vol. 71 (1985)

<8> J.R. Askew, F.J. Fayers, P.B. Kemshell
A General Description of the Lattice Code WIMS
Journal of the British Nuclear Energy Society (1966)

<9> L.P. Abagjan, N.O. Bazazjanc, I.I. Bondarenko, M.N. Nikolaev
Gruppenkonstanten schneller und intermediaere Neutronen fuer die Berechnung von Kernreaktoren
KfK-tr-144

<10> D. Woll
GRUCAL, Ein Programmsystem zur Berechnung makroskopischer Gruppenkonstanten
KfK 2108 (1975)

<11> A.F. Henry
Nuclear-Reactor Analysis
The MIT Press (1975)

<12> A. Sauer
Approximate Escape Probabilities
Nuclear Science and Engineering 16, (1963)

<13> M.L. Williams, D. Gilai
Incorporation of Clad Effects into Sauers Method for Computing Dancoff Factors
Annals of Nuclear Energy Vol. 9, (1982)

<14> J.E. Suich, H.C. Honeck
The HAMMER System, Heterogeneous Analysis by Multigroup Methods of Exponentials and Reactors
DP-1064 (1967)

<15> L.W. Nordheim
A New Calculation of Resonance Integrals
Nuclear Science and Engineering 28 (1962)

<16> M.A.S. Marzo
Untersuchungen zur Anwendung der Isotopenrelationstechnik bei Nachbestrahlungsuntersuchungen und der Überwachung von Spaltstoffen.
KfK 3264 (1982)

<17> H.D. Berger, A.W. Rowe, C. Broeders, M. Schatz
Überprüfung der Berechnungsverfahren für enge Reaktorgitter von Fortschrittlichen Druckwasserreaktoren (FDWR) an experimentellen Anordnungen
KfK 3389 (1982)

<18> C.H.M. Broeders
Neutronenphysikalische Untersuchungen zu engen H2O- moderierten Kernanordnungen.
Jahrestagung Kerntechnik Mannheim, (1982)

<19> E. Kiefhaber
The KFKINR-Set of Group Constants; Nuclear Data Basis and First Results of its Application to the Recalculation of Fast Zero Power Reactors.
KfK 1572 (1972)

<20> H. Bachmann, G. Buckel, W. Hoebel, S. Kleinheins
The Modular Program System KAPROS for Efficient Management of Complex Reactor Calculations
Proceedings of Conference on Computational Methods in Nuclear Engineering, Charleston. CONF-750413 (1975)

<21> B. Goel, B. Krieg
Status of the Nuclear Data Library KEDAK4
KfK 3838 (1984)

<22> I. Broeders, B. Krieg
MIGROS-3: A Code for the Generation of Group Constants for Reactor Calculations from Neutron Nuclear Data in KEDAK Format
KfK 2388 (1977)

<23> H. Huschke, B. Krieg
MITRA- A Program for the Transformation of the Output of MIGROS-2 into Input for GRUMA
KfK 1810 (1973)

<24> D. Woll
Aufbau der Gruppenkonstantenbibliothek GRUBA und ihre Verwaltung durch das Programm GRUMA
KfK 3745 (1984)

<25> R.E. MacFarlane, D.W. Muir, R.M. Boicourt
The NJOY Nuclear Data Processing System,
Volume I: Users Manual
LA-9303-M, Vol. I (ENDF-324) (1982)

<26> B.A. Magurno, P.G. Young, Editors
ENDF-201 Supplement I, ENDF/B-V.2 Summary Documentation
BNL-NCS-17541 (ENDF-201) (1985)

<27> H. Gruppelaar
Private communication 1982

<28> A. Jonsson
THESEUS- A One Group Collision Probability Routine for Annular Systems
AEEW-R253 (1963)

<29> V. Brandl
User Manual for the Kapros Module ONETRA (based on ONETRAN, a one-dimensional SN Code by T.R. Hill, LASL, USA)
Interner Primärbericht 1980

T.R. Hill
ONETRAN, a Discrete Ordinates Finite Element Code for the Solution of the One-dimensional Multigroup Transport Equation
LA-5990 (1975)

<30> E. Stein, E. Wiegner, C. Broeders
Kurzbeschreibung des KAPROS-Moduls BURNUP zur numerischen Loesung der Abbrandgleichungen
Interner Primaerbericht 1982

<31> G. Palmiotti, M. Salvatores
Proceedings of the NEACRP Specialists Meeting on the "NEACRP LMFBR Benchmark Calculation Intercomparison for Fuel Burn-Up"
Cadarache, 28th to 30th April, 1982
To be published

<32> U. Fischer, H.W. Wiese
Verbesserte konsistente Berechnung des nuklearen Inventars abgebrannter DWR- Brennstoff auf der Basis von Zell-Abbrand-Verfahren mit KORIGEN.
KfK 3014 (1983)

<33> C.H.M. Broeders
Ueberlegungen zur Gestaltung eines Programmsystems fuer Untersuchungen zum nuklearen Brennstoffkreislauf
Unpublished report 1982

(34) E.G. Taylor
SAXTON Plutonium Program Critical Experiments for the SAXTON partial Plutonium Core
WCAP-3385-4 (1965)

(35) V.O. Uotinen, J.H. Lauby, L.C. Schmid, W.P. Stinson
Lattices of Plutonium-Enriched Rods in Light Water-Part I: Experimental Results
Nuclear Technology Vol. 15 (1972)

(36) A.R. Boynton, Q.L. Baird, K.E. Plumlee, W.C. Redman, W.R. Robinson, G.S. Stanford
High Conversion Critical Experiments
ANL-7203 (1967)

(37) R. Chawla, K. Gmuer, H. Hager, E. Hettergott, J.M. Paratte, R. Seiler, D. Stahel
Comparisons of Calculated and Measured Parameters for a Pu-Fueled LWHCR Lattice
EIR-Bericht Nr. 463 (1982)

(38) R. Boehme, F. Helm
Reaktorphysikalische Messungen fuer den Fortgeschrittenen Druckwasserreaktor an einem engen Plutonium-Uran-Mischoxyd-Stabgitter mit Polystyrolmoderator
Jahrestagung Kerntechnik Munich (1985)

(39) The CEA experimental program for validating the tight lattice reactor concept
NEACRP/A-635 (1984)

(40) R. Chawla
On the Current Status of the Comparisons between Calculated and Measured Integral Parameters relevant to the Physics Design of LWHCRs.
KTG/ENS - International Seminar, October 1-2,1985 Bonn

(41) R. Chawla, K. Gmuer, H. Hager, R. Seiler
Reactivity and Reaction Rate Ratio Changes with Moderator Voidage in a Light Water High Converter Reactor Lattice
Nuclear Technology Vol. 67 (1984)

(42) S. Pilate
Calculational model for PHENIX T=0
Private communication (1978)

(43) F. Kappler
Calculational model for BZA
Private communication (1982)

(44) J. Braun, E. Kiefhaber
Calculational Input for 62 Fast Reactor Test Problems
Private communication (1982)

〈45〉 L.G. LeSage et. al.
Proceedings of the NEACRP/IAEA Specialists Meeting on the International Comparison Calculation of a Large Sodium-Cooled Fast Breeder Reactor
ANL-80-78 (1980)

〈46〉 C. Golinelli, J.C. Guvot
Benchmark on Recycling of Reprocessed Uranium
NEACRP 690 (1984)

Characterization of some reactor properties.

	PWR	APWR	FBR
Mean enrichment (%)	3	5-15 1) 7-8 2)	15-20
Burnup (GWD/THM)	30-50	>50	100
Conversion ratio	0.6	0.85-1.00	1.20
Void- reactivity	negative	? ? ?	positive
Recriticality possible	no	? ? ?	yes

1) heterogeneous core
2) homogeneous core

PWR and FBR can be predicted with good accuracy, but with different own methods.

KfK INR

Table 1 Comparison of reactor types

Preliminary C/E- values for Keff in SNEAK- assemblies

CORE / METHOD	"MODIFIED FBR"	WEKCPM
	*)	*)
12B	1.0058±0.0010	0.9967±0.0010
12F1	1.0017±0.0003	1.0018±0.0003
12F2	1.0019±0.0006	1.0006±0.0005

*) estimated transport and control rod corrections as SNEAK 12F

Table 2 KARBUS results with G69COLD

		experiment	KFKINR	KARBUS
LMFBR- benchmark	Keff	--	1.0240	1.0223
PHENIX- model	Keff	--	1.0033	1.0174
BZA- experiment	Keff	1.000	0.9990	1.0077
SNEAK- experiment	Keff	1.001	1.0069	1.0012
		experiment	WIMS/D	KARBUS
HIC-8 (UO2)	Kinf	--	1.1988	1.1975
	Keff	1.000	1.0081	0.9926
	ICR	0.500	0.543	0.506
PLUT-7 (MOX)	Kinf	--	1.2150	1.2269
	Keff	1.000	1.0087	0.9968
	CR	--	0.581	0.517
EIR-1	Kinf	1.045±1%	1.0423	1.0407
			+)	
			mean value	KARBUS
NEACRP U236 recycle benchmark	Kinf fresh		1.268	1.2726
	Kinf after 33GWD/THM		0.941	0.9377

+) preliminary results from 5 participants

Table 3 KARBUS validation results

Data on APWR- burnup.

1) TU-BS, Nuclear Technology Vol.59 (1982)

   2.2% reactivity loss for 450 FPD,
   ==> 0.5E-4 ΔK/FPD.

2) GKSS , Nuclear Technology Vol.59 (1982)

   3.6% reactivity loss for 20 GWD/THM,
   ==> 0.5E-4 ΔK/FPD.

3) Uotinen, Edlund et. al. EPRI-NP-1833 (1981)

   6% "depletion allowance", (TABLE IV-16)
   for 350-396 FPD/Cycle,
   ==> (1.5 - 1.7)E-4 ΔK/FPD.

4) KFK results:

   ==> (1.0 - 1.6)E-4 ΔK/FPD.

KfK INR

Table 4 Burnup data 1983

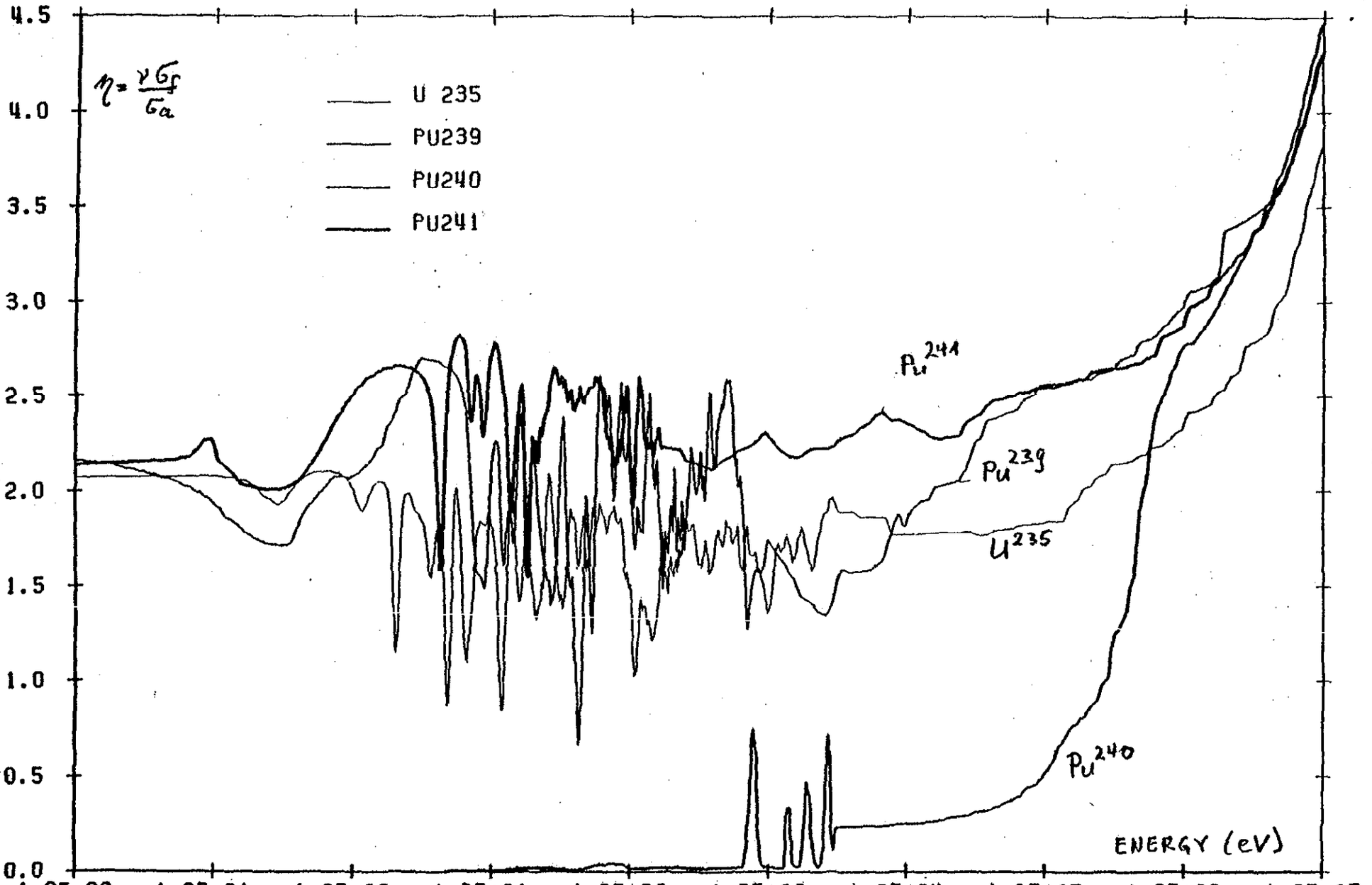

FIG. 1 ETA(E) FOR HEAVY ISOTOPES

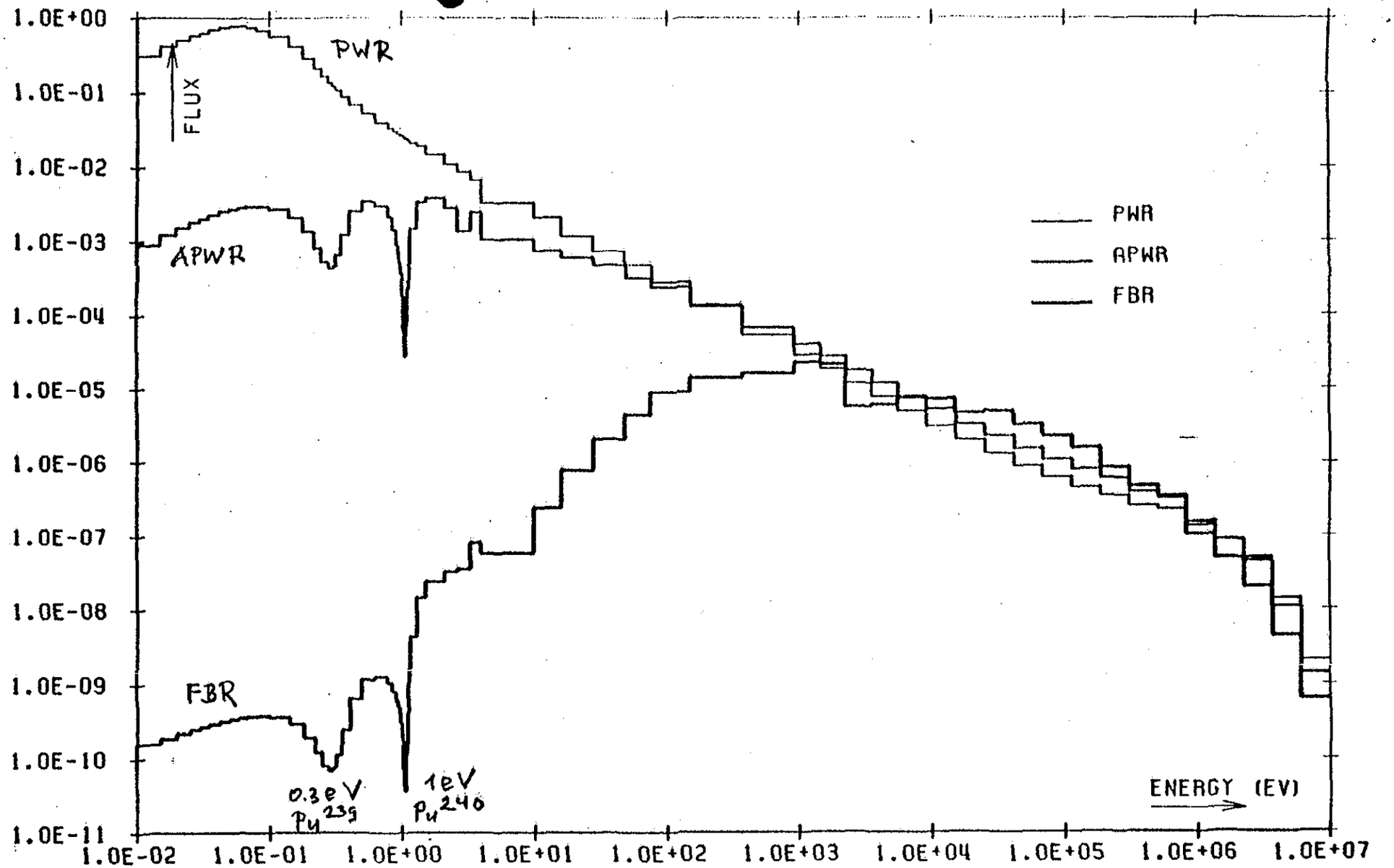

FIG. 2 FLUX PER UNIT ENERGY FOR DIFFERENT REACTOR TYPES

KfK INR

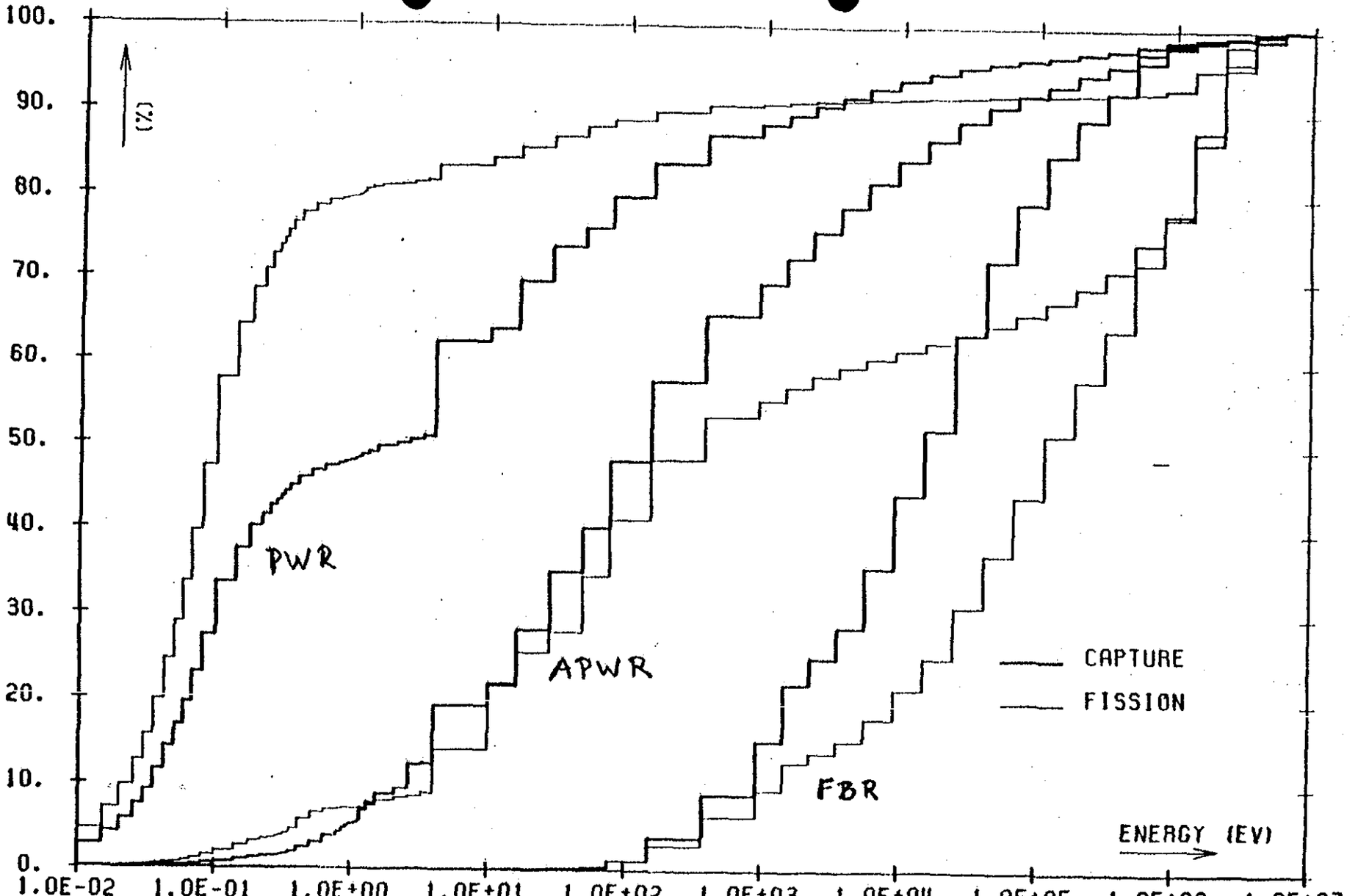

FIG. 3 CUMULATIVE REACTION RATES FOR REACTOR TYPES

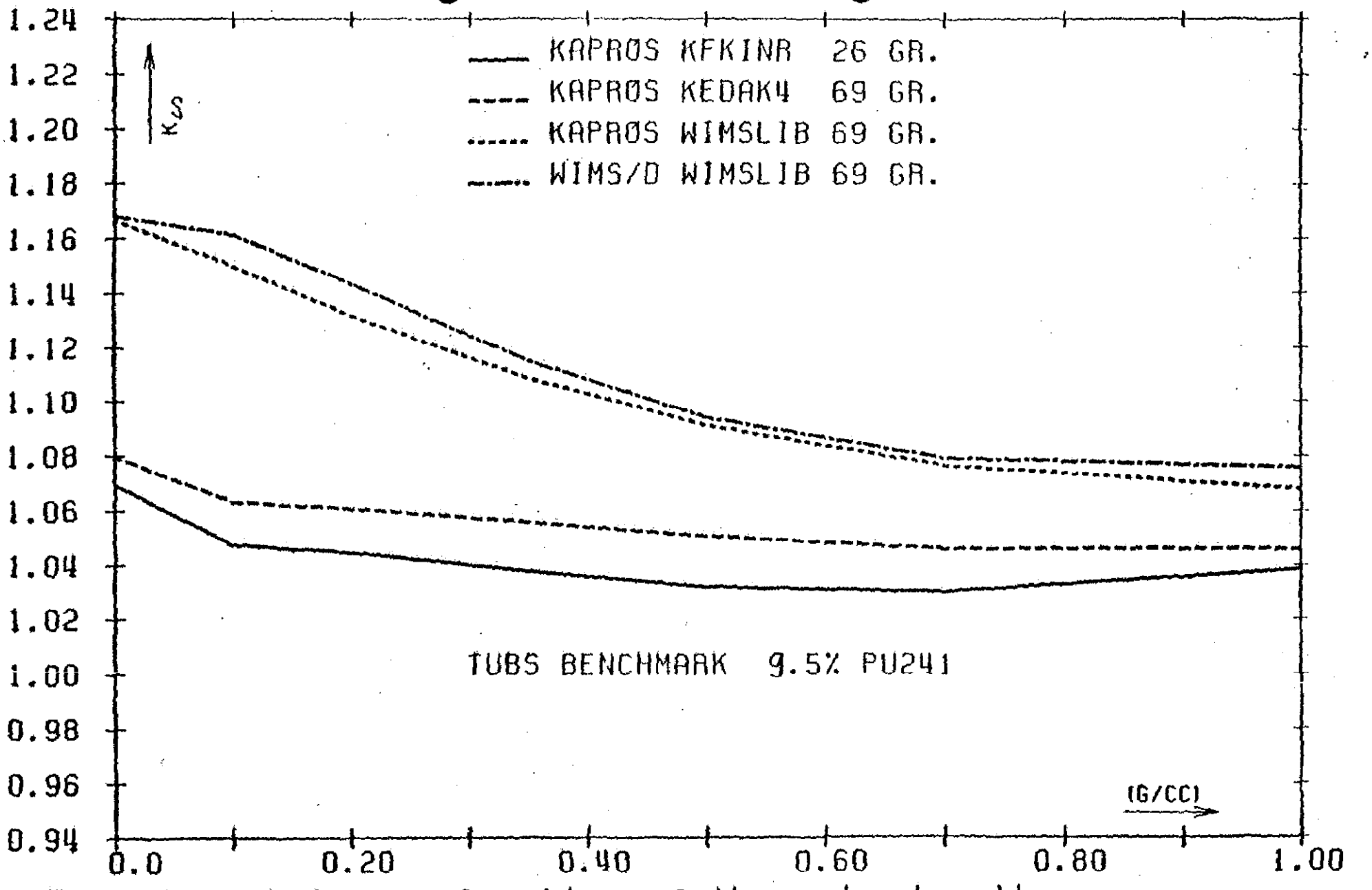

Fig. 4 Kinf as a function of the waterdensity

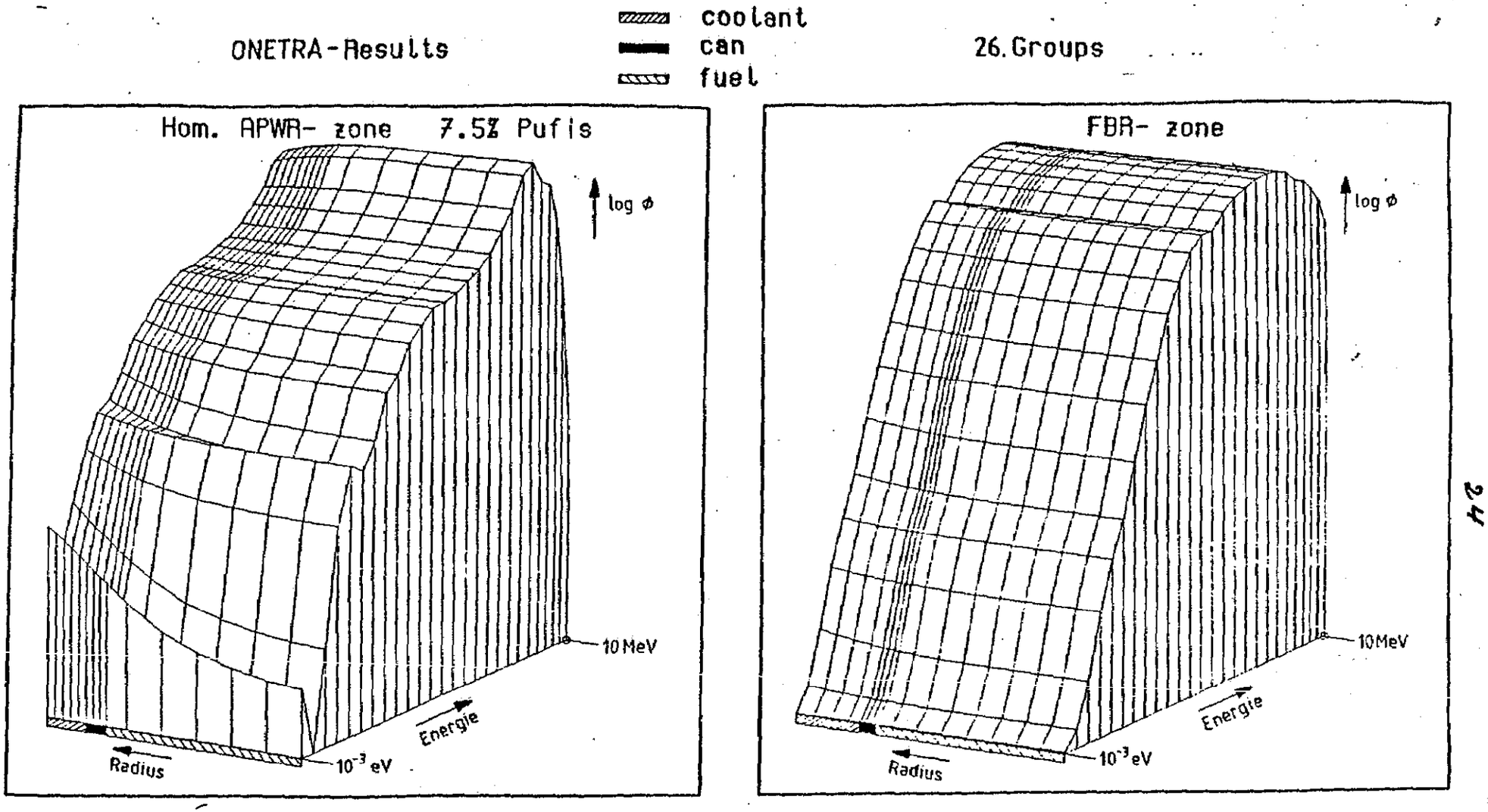

Fig. 5 Neutronflux distributions in APWR and FBR fuel cells