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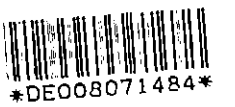

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Abstract

In this paper a short description of the ECN nuclear analysis code system is given with respect to application in HTR reactor physics calculations.
First results of calculations performed on the PROTEUS benchmark are shown.
Also first results of a HTGR benchmark are given.

1. INTRODUCTION

As part of a recently started ECN project on HTRs, a reactor physics analysis will be carried out with emphasis on the safety related physics parameters, like reactivity coefficients and dynamic behaviour during transients.
Until recently no HTR reactor physics calculations had been performed with the ECN code system. Therefore, benchmarking of our code system with computational and experimental benchmarks was needed. To this end ECN has started calculations on the HTR-PROTEUS benchmark.
The ECN reactor physics analysis code system covers the whole range from nuclear model calculations upto whole-core calculations. Within this code package different calculational approaches can be used to solve the same problem, but with distinct computational methods, e.g. deterministic versus Monte Carlo.
The ECN code system has mainly been applied for criticality calculations, shielding analyses, MTR calculations, and cell and assembly calculations, in which reactivity coefficients and burnup effects for PWR, BWR, and Candu geometries were calculated. Also sensitivity and uncertainty analysis for shielding applications can be carried out in this code system.
In section 2 a brief description of the ECN reactor physics code system is given.
In section 3 first results of cell calculations of the first configuration of the PROTEUS benchmark are presented. An intercomparison is made between results of the deterministic codes WIMS-E and SCALE-4, and the Monte Carlo code MCNP4A. Detailed calculations of the Dancoff factor by MCNP for use in SCALE are reported.

Apart from these benchmark calculations coupled neutronics and thermal-hydraulics calculations will be performed in order to analyse transients in HTR type reactors.
At ECN the codes VSOP (suitable for analysing pebble-bed cores) and PANTHER (suitable for analysing cores with prismatic fuel) were installed recently.
The code PANTHER, in conjunction with the WIMS-E code system, is currently applied in a MHTGR computational benchmark. First results are reported in section 4.
In section 5 some conclusions are given.

2. REACTOR PHYSICS CODE SYSTEM at ECN

The ECN nuclear analysis code system covers the whole range from nuclear model calculations up to whole-core calculations.
The following table shows an overview of the most important codes and the fields in which they are applied.

Application	SCALE	WIMS-E	SAS6	DORT,TORT	KENO	MONK-5W	MCNP	SUSD	Bold-Venture	PANTHER	NJOY	ECIS94
Reactivity	x	x	x	x	x	x	x		x	x		
Criticality	x	x		x	x	x	x		x			
Burnup	x	x	x			x			x	x		
Shielding	x			x	x		x					
Sensitivity				x				x				
Dynamics										x		
Fusion neutronics	x			x	x		x					
Nuclear data											x	x
	Deterministic				Monte-Carlo							

The SCALE code system, which includes the 3-D groupwise Monte-Carlo code KENO, is a general purpose system for reactivity, criticality and burnup calculations.
The codes WIMS-E and MONK-5W are general purpose codes for reactivity, criticality and burnup calculations.
The SAS6 code comprises SCALE, WIMS-D and modules developed by ECN and IRI/TU-Delft. SAS6 performs accurate depletion and reactivity calculations in a 2-D geometry taking into account the complete nuclide inventory.
MCNP is a powerful 3-D point energy Monte-Carlo code capable of detailed reactivity, criticality, shielding, and fusion neutronics calculations.
SUSD performs sensitivity and uncertainty calculations for shielding problems. SUSD is extended by ECN for sensitivity and uncertainty analysis for reactors.
DORT and TORT are 2-D and 3-D transport codes, used at ECN mainly for shielding, sensitivity and fusion neutronics calculations.
BOLD-VENTURE is a 3-D diffusion code, specially made for fuel management analysis.
PANTHER combines static or time dependent 3-D, 2-group neutron diffusion with channel thermal hydraulics.
Evaluated nuclear data are processed with NJOY (version 91.91) to obtain data libraries for the reactor codes.
The ECIS94 code calculates nuclear cross sections from nuclear models.
At the moment all reactor codes (except WIMS-E and MONK-5W) use nuclear data libraries generated at ECN based on the Joint Evaluated File JEF-2.2.

The calculations described in this paper have been obtained with the codes WIMS-E, SCALE-4, MCNP4A, and PANTHER. The use of these codes is described in some more detail in the following sections.

3. PROTEUS BENCHMARK EXERCISE

In the framework of the IAEA Co-ordinated Research Programme (CRP) on "Validation of Safety Related Physics Calculations for Low-Enriched (LEU) HTGRs" calculational benchmarks are performed on the basis of LEU-HTR pebble-bed critical experiments carried out in the PROTEUS facility at PSI, Switzerland. ECN joins this CRP since early 1994 with the aim to validate the available reactor physics codes and nuclear data for application to this type of reactors. Of special interest is the treatment of the double heterogeneity of the fuel particles and the spherical fuel elements of these pebble bed core configurations. Also of interest is the proper calculation of the safety related physics parameters, like the effect of water ingress and control rod worth.

3.1. Problem description

The PROTEUS benchmark consists of a series of 6 graphite reflected 16.7% enriched uranium pebble bed systems, named LEUPRO-1 to LEUPRO-6 [1].
There are 3 different types of geometrical arrangement of the pebble bed lattice (with packing fractions 0.74, 0.60, and 0.62) and 2 different moderator to fuel pebble ratios (1/2 and 2/1, respectively).
The pebble diameter is 6 cm. Each fuel pebble contains about 9400 coated particles (in total about 1 g U235) in a graphite matrix. The diameter of the fuel zone is 5 cm.
The coated particles (TRISO type) consist of a fuel grain with a diameter of 500 μm and 4 layers of different coatings, total diameter 915 μm.
In the PROTEUS facility the cylindrical core, which is filled with fuel and moderator pebbles, has a diameter of 125 cm, and is surrounded by radial and axial graphite reflectors. The control and shutdown rods are situated in the radial reflector.
The core is filled until criticality is reached.
The benchmark exercise comprises unit-cell calculations and whole-core calculations for the 6 different pebble-bed configurations.
At this stage of our study the unit cell of LEUPRO-1 (hexagonal closed-packed configuration with M/F ratio of 1/2) has been completed. Emphasis is put on the intercomparison of the results obtained by the deterministic codes WIMS-E and SCALE-4, and the Monte Carlo code MCNP4A.

3.2. Unit cell calculations

In the unit cell the fuel zone is represented by a sphere of 5 cm diameter surrounded by a 5 mm thick graphite shell, which together represent the fuel pebble. Around this fuel pebble sphere a spherical shell with outer diameter of 7.592 cm (for the LEUPRO-1 unit cell) represents the homogenized mixture of moderator pebbles and lattice void.
The unit cell calculations were performed for the single heterogeneous model and the double heterogeneous model. In the single heterogeneous model the coated particles and the matrix material in the fuel pebbles are homogenized over the fuel zone of 5 cm diameter. In the double heterogeneous model the fuel grains are explicitly modelled inside the fuel zone of the fuel pebbles.
Two different cases are studied: the no-leakage (k_{inf}) case and the critical ($k_{eff}=1$) case.

3.3. General aspects of the applied codes

3.3.1. SCALE-4 calculations

In the unit-cell calculations the SCALE version 4 is used with the 172-group library ej2-xmas version Oct.94.
The SCALE system uses the Nordheim method for the resolved resonance treatment and the Bondarenko method for the unresolved resonance treatment. In both methods a Dancoff factor is applied in the treatment of the spatial heterogeneity.
The unit cell is modelled by a 3 zone 1-D spherical geometry. These zones are: a fuel zone, a zone with the graphite shell of the pebble, and a zone representing the moderator pebbles with air in between.
The reactivity calculations of the unit cells are performed by XSDRN using S_{32} quadrature and P_3 scattering order, with reflective boundary condition.

In the single heterogeneous unit cell calculations a Dancoff factor of Df=0.30835 as specified in the benchmark is used.

The SCALE sytem cannot treat the double heterogeneity of the fuel pebbles explicitly. Therefore this calculation is split up in two parts. First, the homogenized cross sections are calculated in a "micro cell". Second, the reactivity is calculated in a single heterogeneous unit cell using the cross sections from the micro-cell calculation for the fuel zone.
The micro cell consists of four zones:
(1) the fuel of a single grain,
(2) the grain coatings,
(3) a zone representing the matrix material and
(4) an extra zone with matrix material to adjust the total micro-cell contents to the correct value of the M/F ratio of LEUPRO-1 unit-cell.
In the resonance treatment the heterogeneity is taken into account with a Dancoff factor of Df=0.2551, which was calculated by MCNP (see section 3.4).
The unit-cell calculation is performed in a 3-zone unit cell as described above. In this calculation the relevant cross sections, weighted over the first 3 zones of the micro cell, are used for the fuel zone.

3.3.2. WIMS-E calculations

WIMS-E version 5a is used for the LEUPRO unit-cell calculations. WIMS-E contains a special module, PROCOL, which handles the double heterogeneity associated with granular fuel in HTR fuel elements, both in the resonance-shielding treatment by the subgroup method and the subsequent multi-group neutron transport calculation. PROCOL was designed for spherical fuel grains embedded in an annular structure of matrix material, which is the correct geometry for the MHTGR benchmark. PROCOL is not capable to treat grains in spheres, so we have used an equivalent cylinder model.

The PROCOL module calculates (region-to-region) collision probabilities for systems containing spherical (fuel) grains (shell substructure definition possible) embedded in an annular structure of matrix material. For the resonance-shielding calculation this module is preceeded by the PRES module, which calculates subgroup cross-sections for the required nuclides. The PROCOL module then calculates the collision probabilities associated with these subgroup cross-sections (resonance-energy range only) and the double-heterogeneous

geometry. Subsequently, the RES module uses these collision probabilities to calculate subgroup fluxes to weight cross sections. In the subsequent neutron-transport calculation again the PROCOL module is employed to calculate collision probabilities associated with the same double-heterogeneous geometry, but this time for the entire neutron energy range. Finally, the PIP module uses these collision probabilities to calculate the fluxes and the multiplication factor.

The equivalent cylinder model
An equivalent cylinder model is used as an approximation for the spherical heterogeneous unit cell.
In the equivalent cylinder model the outer radius of the fuel zone is given by R_1= 1.6666 cm, based on the equivalence of the chord length of the spherical fuel zone. This approximately preserves the escape probability in the fuel zone.
The outer radius of the graphite annulus (R_2= 2.19 cm) and the outer radius of the moderator/void mixture (R_3= 3.12 cm) are based on volume conservation.
In order to check this approach, MCNP calculations were performed for both models.
A good agreement between equivalent cylinder and spherical unit cell is observed. The deviation in k_{inf} is within the statistical uncertainty (~0.1%). The deviation in spectral indices is less then 3%.

3.3.3. MCNP4A calculations
Monte Carlo calculations were performed with MCNP4A. This code offers the possibility to take into account all details of the 3-D HTR-PROTEUS geometry. Fuel grains (and the surrounding coating shells) in the unit cell can be modelled, so the double heterogeneity is treated explicitly, see fig. 1.
In this model the fuel zone of the fuel pebble was filled with a rectangular lattice of cubic microscopic unit cells, which contain the fuel grains inside the matrix.
The dimension of this micro-cell is such that the fuel sphere (see fig. 1) contains the specified number of 9394 fuel grains.
Continuous-energy cross sections (processed by NJOY91.91) are used in MCNP, bases on the JEF-2.2 evaluation.
Modifications to MCNP4A were applied in order to make white boundary conditions possible. Also, an extra option was added for the calculation of Dancoff factors in arbitrary geometry.

In the k_{inf} case the spherical unit cell has isotropic reflecting boundary conditions.
For the k_{eff} = 1 case criticality is reached in the deterministic calculations by buckling search to account for the proper leakage. In MCNP criticality is achieved by stacking cubic macroscopic unit cells in a rectangular lattice in a large sphere with vacuum boundary condition and radius $R_{crit.}$ The value of R_{crit} was determined by the condition k_{eff} = 1.
The cubic macroscopic unit cell is the equivalence of the spherical unit cell providing volume conservation of the LEUPRO-1 unit cell (cube edge = 6.119 cm).
Spectral indices are determined in a volume at the centre of the critical sphere, containing 5x5x5 unit cells. The dimension of this cube is still small compared to the dimension of the large critical sphere (D_{cube}= 30.6 cm, R_{crit} > 100 cm). The shape of the neutron flux spectrum is assumed not to change over this central cube.

3.4. Dancoff factor calculations

In the resonance self-shielding calculations in SCALE the spatial self-shielding due to the heterogeneity of the pebble bed lattice and the heterogeneity of the fuel grains in the fuel zone is taken into account by using the Dancoff factor (Df). This Dancoff factor is defined as the probability that a neutron leaving a fuel grain (or fuel region) will escape a collision in the moderator material in the fuel pebbles or the moderator pebbles in the surroundings and will enter again a fuel grain (or fuel region) in the same or in another fuel pebble. The Dancoff factor is calculated by MCNP as accurately as possible, both for the single heterogeneous case as well as for the double heterogeneous case.

In a real fuel sphere the fuel grains are distributed in the fuel zone in a stochastic sense. Therefore, it is required that the model used is equivalent to the stochastic model.
In order to check the assumptions made in the MCNP calculations in the double heterogeneous calculations (viz. rectangular lattice of cubic micro-cells), Dancoff factors were calculated in two simple micro-cells (containing one fuel grain without coating layers):

(1) A cubic micro-cell (with specularly reflecting boundary conditions) representing an infinite rectangular lattice of cubic microscopic unit cells. This model of the fuel zone is used in the double heterogeneous calculations.
(2) A spherical micro-cell (with isotropic reflection boundary conditions) representing an infinite medium filled with a stochastic distribution of fuel grains. This model represents the situation in the fuel zone of a real fuel pebble.
For this simple geometry the Dancoff factor can also be calculated analytically.

The resulting values of the Dancoff factor are:
Cubic micro-cell: 0.4007
Spherical micro-cell: 0.4006
Analytical calculation: 0.4005
Therefore, the rectangular lattice of cubic micro-cells is a good description of the fuel zone of a real fuel pebble.

The Dancoff-factor option in MCNP4A was also used to calculate the Dancoff factor to be used in the deterministic SCALE calculations of the double-heterogeneous composition. For specific grains inside the heterogeneous fuel-pebble Dancoff factors as function of radial distance from the centre of the fuel pebble were calculated, see fig. 2. The volume-averaged Dancoff factor over the fuel pebble was determined to be Df = 0.2551 for the LEUPRO-1 unit cell, and Df = 0.2313 for the LEUPRO-2 unit cell.

3.5. ECN results on PROTEUS benchmark

Table 1 shows the results of the single heterogeneous unit cell calculations.
Table 2 shows the results of the double heterogeneous unit cell calculations.

Good agreement of k-values and spectral indices for single heterogeneous and double heterogeneous unit cell calculations of the pebble bed HTR type reactor configuration is obained between the deterministic and Monte Carlo type of codes. These results are also in good agreement with IRI results [2] and in reasonable agreement with the results of other participants in the PROTEUS benchmark [3].

4. MHTGR BENCHMARK EXERCISE

General Atomics has proposed a "nuclear physics benchmark" on MHTGR [4] to validate and verify the computational methods for the calculation of core physics parameters.
A series of core physics benchmark problems for a realistic 450 MWth MHTGR has been specified with increasing complexity to facilitate comparison of calculational results from different computing methods (diffusion theory, transport theory, Monte Carlo) to be performed by various organizations. These computational comparisons will be used to determine the accuracy of the computing methods used for MHTGR physics analysis as part of the code validation process.

4.1 Problem description

The computational benchmark problem as specified is representative for a 450 MW MHTGR at beginning of initial cycle, with an annular core containing 84 hexagonal columns. The active core is an annulus of three rows of colums with 10 elements per column, an inner reflector of 37 graphite columns, an outer radial reflector and axial reflectors above and below the core.
The annular core contains 12 control rods and 12 holes for reserve shutdown rods. The outer reflector contains 24 control rods for operation control.
The fuel is composed of 19.8% enriched fissile particles (of 350 μm diameter) and natural uranium fertile particles (of 500 μm diameter).
For increasing complexity, the benchmark includes 1-D radial, 1-D axial, 2-D hexagonal, and 3-D hexagonal models. All models include hot versus cold condition, without and with moisture ingress. The 2-D and 3-D problems include also all-rods-in cases.
Parameters to be evaluated are: k_{eff}, power distribution, water-ingress effects, control rod worths.

4.2. The code PANTHER

Besides the WIMS-E code, also the general purpose 3-D reactor code PANTHER is used in the MHTGR benchmark calculations. PANTHER combines static or time-dependent 3-D, 2-group neutron diffusion with channel thermal-hydraulics. Although PANTHER was originally designed for PWR and AGR/Magnox simulation, which is especially reflected in the built-in thermal-hydraulics models for these reactor types, it is intended to use this code for static and transient combined neutron diffusion and thermal hydraulics calculations on HTRs, by connecting an external HTR thermal-hydraulics model to the code. As a first step in this development a comparison is made between the calculational results of PANTHER and the SNAP (3-D multigroup diffusion) module of the WIMS-E modular code-system for the first case of the first model of the MHTGR benchmark. The 2-group nuclear data required for a PANTHER calculation can be generated conveniently by the WIMS-E code system.

4.3. Results of calculations

Some preliminary calculations have been performed by the WIMS-E and PANTHER codes on the first model of the MHTGR benchmark: the 1-D radial case in the cold (300 K), unrodded state without water ingress. The geometry for this case consists of 8 annular

regions representing the central reflector (regions 1 and 2), the annular core containing the fuel compacts (regions 3, 4 and 5) and the outer reflector (regions 6, 7 and 8). The outer radius of this structure is 306.6 cm.

These calculations have been performed for two purposes:

- Calculation of the benchmark results. As the benchmark only concerns the reactor physics aspects (without thermal hydraulics) of the MHTGR, this could be done by using only the WIMS-E code system, employing its multigroup one-dimensional neutron transport (PROCOL/THESEUS) and three-dimensional diffusion modules (SNAP). However, as we intend to be able to perform both static and transient combined neutronics and thermal hydraulics calculations in the future, the second purpose of this exercise is:
- Intercomparison of WIMS-E (transport and diffusion) and PANTHER (two-group diffusion) calculations.

In the process of serving these two purposes the following steps were taken:

(1) The first step is a 69-group neutron transport calculation (main WIMS-E module: THESEUS; calculates multigroup collision-probabilities for annular systems) for the entire 1-D radial case, assuming homogenized (i.e. volume-weighted number densities of fuel compacts and graphite blocks) core regions (regions 3, 4 and 5). The resulting multiplication factor is k_{eff} = 0.87775. Condensation from 69 to 2 groups yields the 2-group flux distribution and 2-group macroscopic cross sections for the 8 core and reflector regions, to be used in subsequent 2-group neutron diffusion calculations. The resulting 2-group flux distribution is shown in fig.3. Note the strong decrease of the fast flux in the reflector regions. For the time being the energy boundary between the fast and thermal flux is set at 0.625 eV, as is the case for the PROTEUS benchmark. Most probably a higher value is more suitable for high-temperature systems. This aspect should be investigated further.

(2) The second step is a diffusion calculation (R-Z geometry) of the same structure by the SNAP module, employing the 2-group macroscopic cross sections generated in the first step. The resulting multiplication factor is k_{eff} = 0.87699. This corresponds quite well to the result of the 69-group transport calculation (first step): the difference in k_{eff} is -0.09%. In fig.4 a comparison is presented of the fast and thermal flux distributions, resulting from the 69-group transport calculation and the 2-group diffusion calculation. Note that the differences only become quite large in the reflector regions, where the flux is 2 to 4 orders of magnitude lower than in the core regions.

(3) The third step is also a 2-group (R-Z) neutron diffusion calculation for the entire 1-D radial system by the SNAP module, but in this case the 2-group macroscopic cross sections for the core regions (regions 3, 4 and 5) are obtained from a 1-D radial, 69-group neutron transport calculation (main WIMS-E module: THESEUS) on unit cells consisting of a fuel compact (outer radius: 0.6223 cm) and the associated part of the fuel block (outer radius: 1.3828 cm).

Actually, these unit cell calculations should have been performed employing the PROCOL module, taking into account the granular structure of the fuel compact. However, as PROCOL can not properly handle mixtures of two types of fuel grains (viz. fissile and fertile), it was decided, for the time being, to use the THESEUS collision probability module, assuming homogenized (volume-weighted) fuel compacts. This results in a significantly lower value of k_∞ for the unit cells than would have been obtained by a proper treatment of the granular structure. However, it is intended to repeat these unit-cell calculations with an improved version of the PROCOL module, enabling a proper calculation of the benchmark case.

An extra problem is associated with the use of these 2-group macroscopic cross sections in SNAP, in view of the eventual comparison with the PANTHER code, namely: PANTHER can not handle upscattering. Therefore, it was decided not to use the full (2-by-2) scatter matrix in the SNAP R-Z calculations, but to use an upscatter-corrected downscattering cross-section, i.e. a downscattering cross section diminished by the upscattering cross section times the thermal-to-fast flux ratio. The upscattering cross section is consequently set to 0. This preserves the net downscattering rate, provided that the spectrum does not change too much. This upscatter-correction was tested by the comparison of two 2-group SNAP diffusion calculations for the same geometry, employing the full 2-by-2 scatter matrix in the first case, and only an upscatter-corrected downscatter in the second case. Comparison of the calculational results shows a good agreement between the two cases (difference in k_{eff} less than 0.05%).
Using the 2-group macroscopic cross sections for the core regions thus generated, we performed the -for the moment- most practical 2-group diffusion calculation by the SNAP module for the benchmark case, resulting in $k_{eff} = 0.902049$. As expected, this value is higher than the value obtained from the first step. A proper treatment of the granular fuel structure by the -improved- PROCOL module should yield an even higher value of k_{eff}.
(4) The fourth step was taken mainly to serve the second purpose: the intercomparison of WIMS-E and PANTHER. Beside the problem mentioned above (PANTHER can not handle upscattering) a further problem is associated with this intercomparison, namely: a PANTHER geometry can only consist of rectangular or hexagonal elements of equal size.
To overcome this problem, the annular structure of the 1-D radial benchmark case is approximated by a structure consisting of small hexagons, which can be modelled in both SNAP and PANTHER.
In table 3 the results of the SNAP and PANTHER calculations, using exactly the same geometry and macroscopic cross sections, are presented for three values of the side dimension of the hexagons. Note the excellent agreement between the SNAP and PANTHER results for the hexagonal cases. Also the results of the 1-D and hexagonal SNAP calculations agree quite well.

5. CONCLUSIONS

- Good agreement of k-values and spectral indices for single heterogeneous and double heterogeneous unit-cell calculations of the pebble-bed HTR-type reactor configuration is obained between the deterministic and Monte Carlo type of codes. These results are also in good agreement with IRI results and in reasonable agreement with the results of other participants in the PROTEUS benchmark.
- MCNP provides good reference data for complex geometries.
- The Dancoff factor in complex geometries can easily be calculated by using the option added to MCNP.
- As far as escape probabilities in the resonance region are concerned, the rectangular lattice of cubic unit-cells as used normally in MCNP calculations, is equivalent to a stochastic distribution of fuel grains over the fuel sphere.
- Calculations in the framework of the General Atomics MHTGR benchmark can most probably be performed quite well by the WIMS-E and PANTHER code systems. However, the problem with modelling mixtures of two types of fuel grains in the WIMS-E module still exist. For the moment a practical solution is to generate

homogenized two-group nuclear constants, necessary for reactor calculations with PANTHER, by appropriately mixing the results of PROCOL cell calculations for unit cells with single fuel grain types. The resulting k-value and homogenized nuclear constants should be verified against results of detailed MCNP unit cell and assembly calculations.

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Table 1: *Results for the single heterogeneous unit cell.*
(Spectral parameters are given for critical conditions)

	WIMS	SCALE	MCNP	IRI [2]	AVER [3]
k_∞	1.641	1.646	1.640	1.639	1.631
$k_\infty(B^2 = B^2_{cr})$	1.583	1.591		1.584	1.574
H (cm)	117	123	"116"	122	119
ρ^{28}	11.7	11.3	11.7	11.4	-
δ^{25} (10^{-2})	11.2	11.1	11.6	11.2	-
δ^{28} (10^{-4})	16.2	16.0	16.3	17.0	-
C^*	0.283	0.274	0.281	0.275	-

Table 2: *Results for the double heterogeneous unit cell.*
(Spectral parameters are given for critical conditions)

	WIMS	SCALE	MCNP	IRI [2]	AVER [3]
k_∞	1.729	1.726	1.729	1.721	1.716
$k_\infty(B^2 = B^2_{cr})$	1.679	1.678		1.673	1.669
H (cm)	112	118	"110"	118	113
ρ^{28}	7.66	7.68	7.57	7.66	7.77
δ^{25} (10^{-2})	11.0	10.9	10.9	11.0	11.4
δ^{28} (10^{-4})	18.5	18.4	18.5	17.0	15.4
C^*	0.192	0.193	0.191	0.192	0.194

Table 3: *Intercomparison of SNAP and PANTHER results for different hexagon sides*

SNAP 1-D radial k_{eff}	0.902475		
Hexagon side dimension (cm)	1.5	2.5	10.0
SNAP hexagonal k_{eff}	0.903636	0.902502	0.895926
PANTHER hexagonal k_{eff}	0.903641	0.902502	0.895926

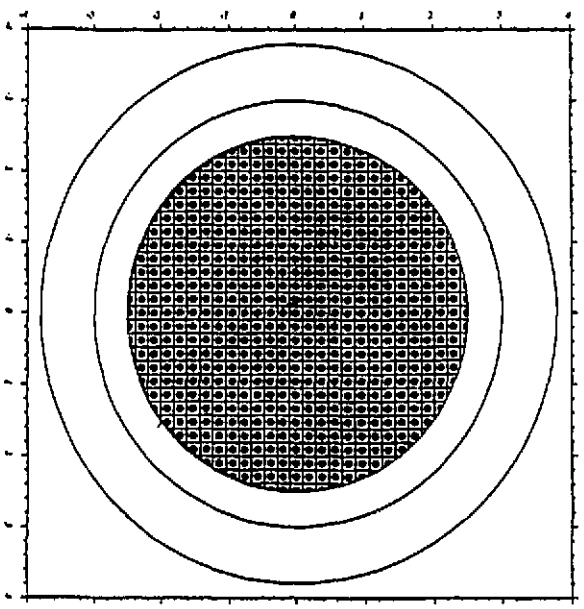

Fig. 1. *MCNP representation of the double heterogeneous unit cell of LEUPRO-1 taking into account all fuel grains (including coatings) in the fuel zone.*

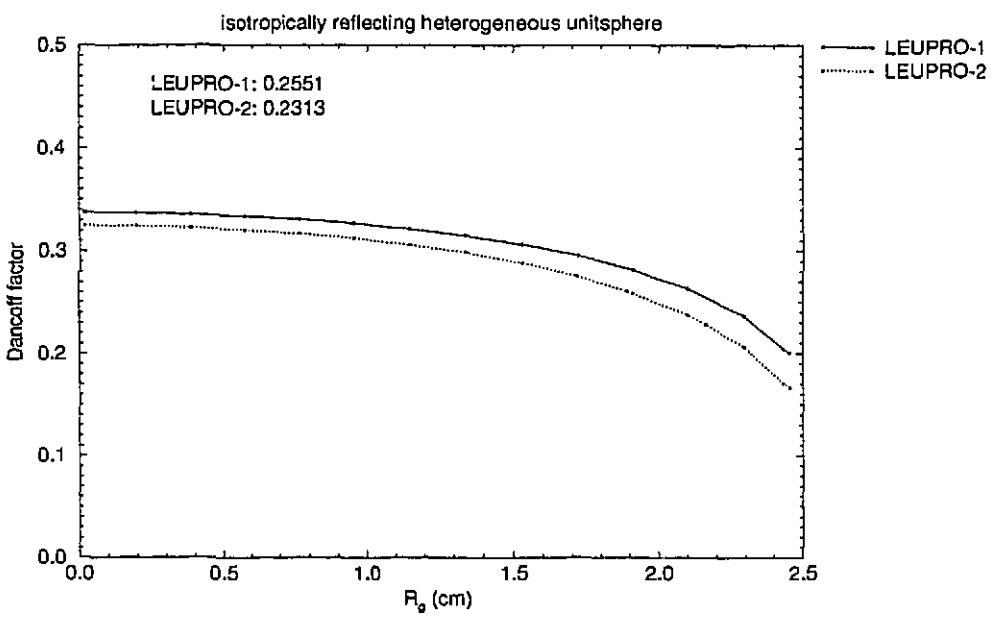

Fig. 2. *Radial dependence of Dancoff factor in the fuel zone of the LEUPRO-1 and LEUPRO-2 geometry as calculated with MCNP4A.*

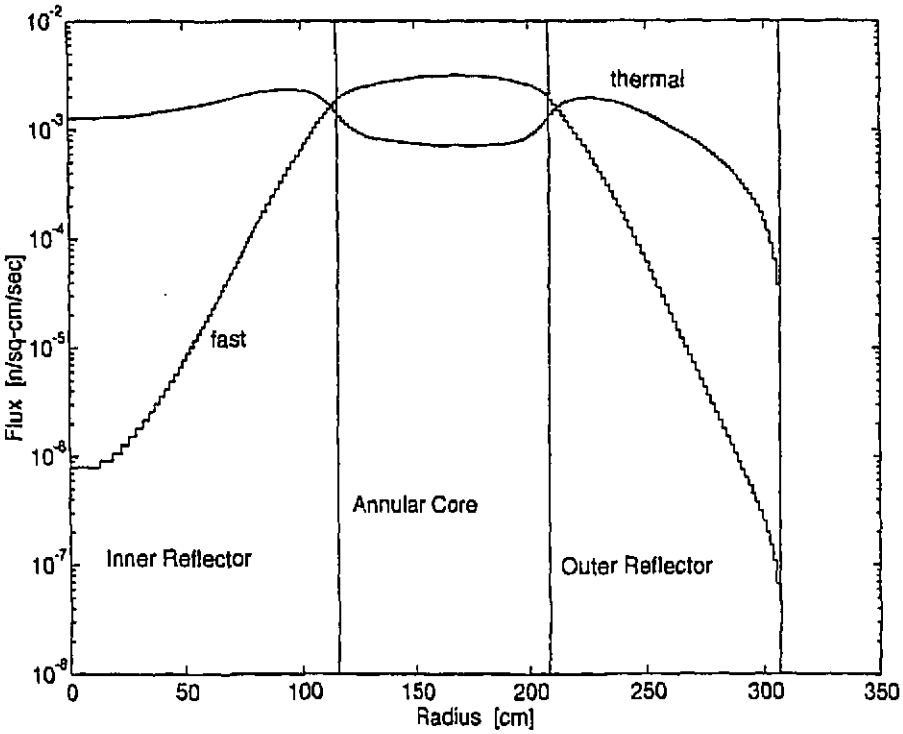

Fig. 3. *WIMS 2-group flux distribution in 1D Radial Benchmark*

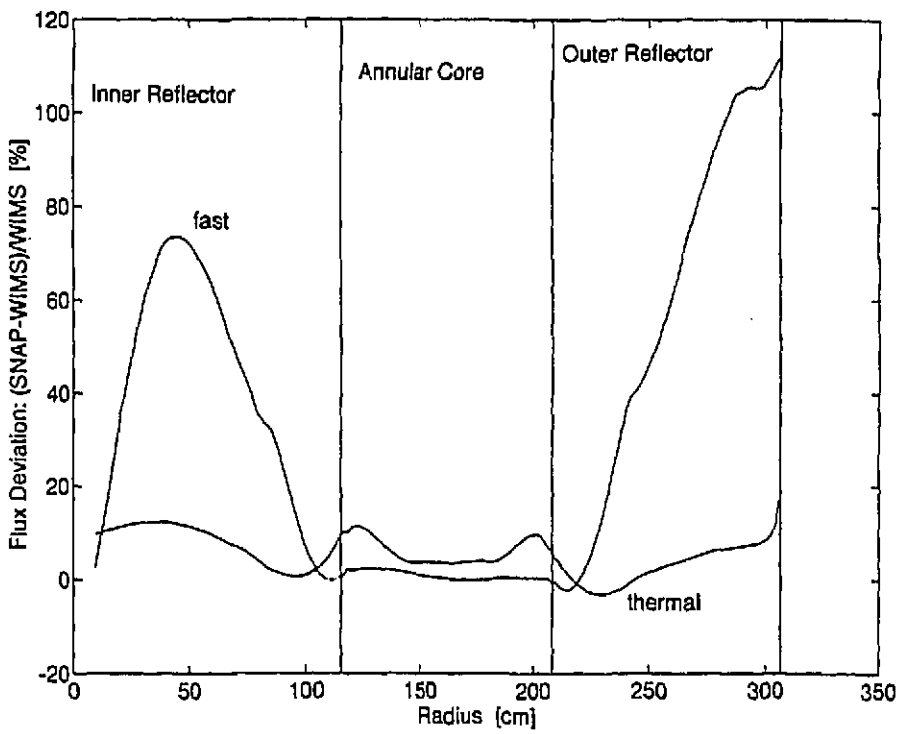

Fig. 4. *Comparison of SNAP and WIMS 2-group Flux Distributions*