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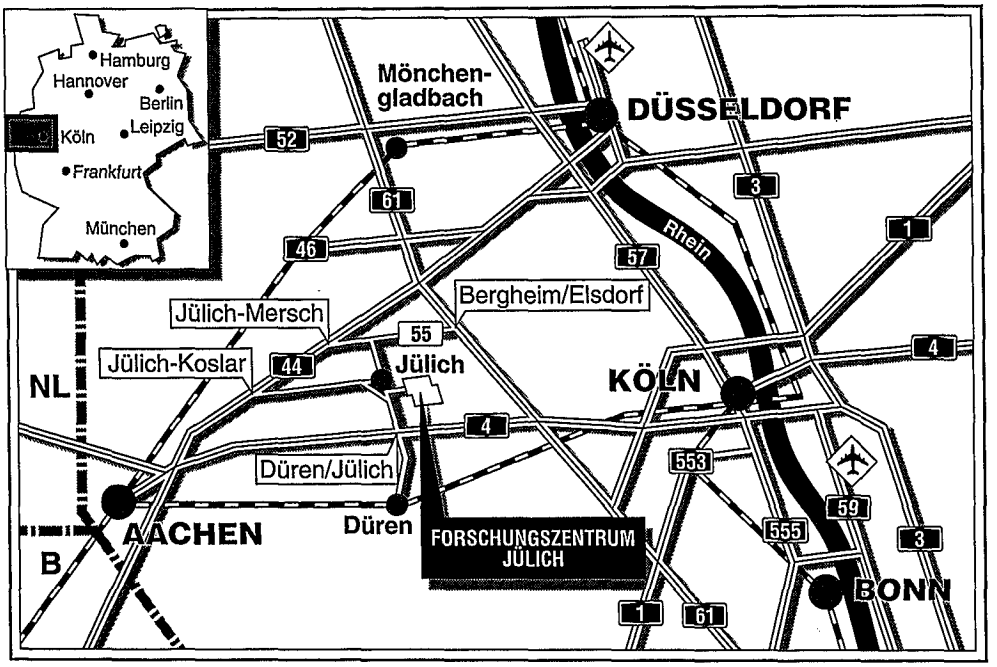

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Modellierung von Neutronenabsorbern in Hochtemperatur-Reaktoren mit kombinierten Transport-Diffusions-Methoden

von

V. Fen, M. Lebedev, V. Sarytchev, W. Scherer

KURZFASSUNG

Die neutronenphysikalische Beschreibung starker Neutronenabsorber zur Regelung und Abschaltung von Kernkraftwerken erfolgt heute unter Verwendung von kombinierten Transport- und Diffusionsmethoden. Für den Hochtemperaturreaktor werden hier zwei derartige Wege beschrieben und miteinander verglichen.

Die Methode der äquivalenten Wirkungsquerschnitte wurde im Institut für Sicherheitsforschung und Reaktortechnik der KFA Jülich entwickelt und für die deutschen HTR-Konzepte mit Erfolg eingesetzt. Im Institut für Kernenergietechnik in Obninsk, UdSSR, wurde die Response-Matrix-Methode erarbeitet. Sowohl die äquivalenten Wirkungsquerschnitte als auch die Responsematrix sind Ergebnisse von Transportrechnungen, die dann in mehrdimensionalen Diffusionsprogrammen zur Darstellung der Absorbergebiete verwendet werden.

Anhand von typischen Beispielen für den modularen Kugelhaufen-HTR konnte die gute Übereinstimmung der Ergebnisse beider Methoden untereinander und gegenüber reinen Transportverfahren, wie S-N und Monte-Carlo, nachgewiesen werden.

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Modelling of Neutron Absorbers in High Temperature Reactors by combined Transport-Diffusion-Methods

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ABSTRACT

Today, the neutron-physical description of strong neutron absorbing materials for control and shut-down of nuclear power plants is performed using combined transport and diffusion methods. Two of these approaches are described and compared in this paper.

The method of equivalent cross-sections has been developed at the KFA-Jülich Institute for Safety Research and Reactor Technology (ISR) and was widely used for all german HTR reactor concepts. The Obninsk Institute for Nuclear Power Engineering (OINPE) in USSR has worked out the response-matrix method for rodded regions. Both, equivalent cross-sections and response-matrix are results of transport calculations. They are subsequently used in multidimensional diffusion type reactor codes for the representation of absorber regions.

The analysis of typical examples related to the modular pebble-bed HTR showed a good agreement of the results of both methods relative to each other and relative to pure transport calculations such as S-N and Monte-Carlo.

Preface

This work was carried out within a scientific cooperation between

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IN PEACEFUL APPLICATIONS OF NUCLEAR ENERGY

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1.0 Introduction and Description of Problem

Traditional diffusion calculations for nuclear reactor cores in two or three dimensional geometries usually meet some difficulties connected with regions of great absorbers. Inside and in the vicinity of such regions the diffusion theory generally is not valid and so there is no accurate description of absorbers using pure diffusion programs. However, transport methods are time- and memory-consuming and normally not useful for full-core calculations. Additionally, for graphite-moderated systems like the high-temperature reactor most of the transport methods either show very slow convergence (S-N) or take not acceptable running times (Monte-Carlo).

To overcome these difficulties combined transport-diffusion methods have been developed in recent years. Basic idea of such methods is to describe the absorber region and optionally some part of its vicinity using transport theory, extracting some macroscopic data for this region from the transport calculation and to use these data in a subsequent 2-D or 3-D diffusion calculation for the whole reactor.

There are some ways known to follow these basic ideas. Most popular is the method of inner boundary conditions. In this case the ratio of net current to flux at the boundary of the absorber region is transferred from the (mainly) 1-D transport calculation to the diffusion code. No azimuthal asymmetry is taken into account, the difference between transport- and diffusion-flux is neglected and the finite mesh structure of the diffusion code often raises problems related to the discretization errors. Earlier calculations and comparisons with experiments /1,2,3/ have shown that this method is inadequate in many cases. Especially in combination with mesh-centered diffusion codes major problems may arise.

The search for better models of absorber regions has brought up two methods which seem to overcome at least partly the difficulties mentioned above. The method of equivalent cross-sections (MECS) can be adapted to the special diffusion code used for the reactor calculation. Earlier analyses have shown, that transport results can be reproduced by this method nearly exactly on a cell level /2/. However, azimuthal effects for absorbers outside the core are neglected here, too.

The response-matrix method (RMM) /4,5,6/ is capable to overcome this problem. A 2-D transport calculation for the absorber region retaines the main information on non-isotropic flux shape relative to the absorber center. Naturally, the calculational effort is larger in this case. Modern programming techniques and program optimization help to minimize this disadvantage.

Both methods seemed to have some lack of validation with respect to the requirements of the now discussed conceptions of modular pebble-bed high-temperature reactors, such as the german MODUL-HTR and the soviet VGM. In these reactors with a relatively small core diameter of about 3 m the control and shut-down elements - control rods and small absorber spheres in special shaped cylindrical borings - are located in the side reflector outside the pebble-bed core. The neutron flux therefore has some gradient and buckling in the absorber region leading to effects of asymmetry.

Besides the detailed description of the methods mentioned, it was the aim of this study to choose some example, representative for the special situation of modular HTRs, and to check the combined methods against themselves and against some pure transport codes. The latter were chosen to be DOT-IV /7/ as representative for the S-N method and MORSE-CG /8/ as a Monte-Carlo code.

The combined transport-diffusion methods are explained in chapters 2 and 3. A detailed description of the test examples and the calculation results are given in chapter 4. Finally, a summary is presented in chapter 5.

2.0 The Method of Equivalent Cross-Sections

2.1 General Description of the Method

Following the principle of combining transport and diffusion methods for the representation of strong neutron absorbers in nuclear reactors the task is divided into several parts:

1. Unit cell calculation for the heterogeneous absorber region using (e.g. S-N) transport theory

   - fine group energy discretization (e.g. 123 groups)
   - fine spatial discretization (e.g. 50 mehspoints)
   - detailed S-N and scattering matrix representation (e.g. S-8/P-3)

2. Calculation of equivalent macro-cross-sections

   - homogenization of macro cross-sections with respect to average cell flux
   - condensation of cross-sections to few group structure (e.g. 4 groups)
   - evaluation of few-group leakage at absorber region boundary
   - calculation of equivalent diffusion constants using transport leakage and diffusion code solution method

3. multidimensional reactor calculation using diffusion theory and the equivalent macro-cross-sections for absorber region

A detailed description of these tasks has been given previously /2,3/. In this frame the basic principles shall be recalled and some considerations concerning the special situation of the modular HTRs shall be added. The method of equivalent cross-sections is based on some general assumptions, namely:

1. Volume-preserving modelling of the absorber region into the R-ϕ-Z- or X-Y-Z-mesh of the diffusion code

2. Assumption of equivalence between transport- and diffusion-flux at some point outside the absorber region surface

3. Assumption of flux-equivalence at points of equal distance from the absorber region surface independent of absorber geometry

4. Consideration of the diffusion code's solution method in calculating equivalent cross-sections

The volume-preserving model is used mainly to maintain the material balance outside the absorber region. Otherwise for absorbers inside the core the fissile material density, for reflector rods the moderator density would have to be adjusted. Exept these restrictions other models like surface-preserving ones are possible, too.

The flux at the absorber-nearest mesh point in the diffusion calculation must not suffer from a transport correction, i.e. it has to be equal in diffusion and transport theory. This is normally true for distances of some 5 cm in graphite. It is up to the user to manage the diffusion mesh structure in a corresponding proper way.

Because the transport calculation is normally performed in 1-D geometry only, azimuthal non-isotropy effects in the X-Y- or R-ϕ-plane related to geometry effects either of the absorber region or the overall reactor arrangement cannot be accounted for. This is the reason for assumption no. 3 in the above listing. If 2-D transport calculations are foreseen, this restriction may be cancelled. The error that is produced by neglecting these effects may be of some importance for the reflector controlled modular HTRs. That gave reason (among others) for this analysis.

Finally the solution method of the diffusion code is important to achieve equalization of the absorber region leakage rates in both transport and diffusion calculation. As the equivalent diffusion constants will directly be affected by this method it is neccessary to find a simple analytical approach for the geometry under consideration. For the diffusion code CITATION /9/ which is widely used in the KFA-ISR and elsewhere, a special code was developed to perform this task assuming that there is only one inner mesh point in the absorber region diffusion representation. Allowing for more inner meshpoints increases the difficulties to find an analytical expression for determination of the proper diffusion constants.

The method of equivalent cross-sections can also be used for absorbers in large cavities as for example in the upper cavity of pebble-bed reactors with the so called OTTO /10/ loading scheme like in the soviet VG-400 or the german HTR-500. Test calculations using this method and comparison against 2-D transport calculations have successfully been performed earlier /11/.

2.2 Adaption to the Computer Codes XSDRNPM and CITATION

The Institute for Safety Research and Reactortechnology (ISR) at KFA Jülich widely uses the Oak Ridge AMPX-II Code System /12/ for preparation of basic cross-sections and generation of fine and broad group system dependent nuclear data. Reactor neutronics calculation normally are performed using the Oak Ridge diffusion code CITATION /9/. Both have been reviewed, corrected and modified for the special requirements of the gas-cooled, graphite moderated high temperature reactor.

Starting with a 123-energy-group nuclear library based on ENDF/B-3 and /B-4 data together with own compiled graphite matrices resonance absorption is treated in the AMPX-module NITAWL. Using the S-N module XSDRNPM which is similar to the well known ANISN /13/ code cell calculations e.g. for absorber regions are performed. A special module, RODCIT was written at KFA-ISR for the purpose of generating the equivalent cross-sections for the diffusion code CITATION. RODCIT uses the fluxes and macroscopic cross-sections of XSDRNPM and performs the homogenization over the absorber region that shall be represented in the diffusion calculation. Depending on the CITATION mesh structure in and around this region the proper diffusion constant is calculated in order to match the absorber region leakage rate in the diffusion calculation to the transport code result. Finally all the cross-section macro data necessary for CITATION are printed in the correct CITATION format. Additional precautions have been made for special situations, such as treatment of reflecting boundary conditions inside the absorber region or treatment of two inner mesh points.

The geometrical shape of the absorber region may be quadratic or quasi-quadratic (in R-ϕ-Z geometry) or more generally rectangular. In the latter case, two flux reference points are used to describe the 2-dimensional partial leakages for both directions in the plane. It should be mentioned, that for strongly variing conditions in the Z-direction the equivalent cross-sections can be calculated for several axial regions.

As an example the geometric situation for a control rod is shown in Fig. 1 on page 7. The cylindrical structure on the top is treated in XSDRNPM. The inner shaded part is seen to be the absorber region that shall be transferred to CITATION. On the bottom this representation in the CITATION code is given. The four adjacent meshes are pairwise identical. The distances of their inner flux points to the surface of the absorber region are used to define the reference flux points in the XSDRNPM calculation. Leakage rates are normalized to these fluxes when calculating the diffusion constants. Because of the capability of CITATION to treat non-isotropic diffusion constants in

principle two different leakage rates with two different diffusion constants for each direction are possible. In the moment an average diffusion constant is used, however.

For example the leakage in X-Y geometry is calculated within the CITATION code in the following way:

$$L = \frac{F_y(\Phi_x - \Phi_a)}{\frac{\delta_x}{D_a} + \frac{\Delta_x}{D_o}} + \frac{F_x(\Phi_y - \Phi_a)}{\frac{\delta_y}{D_a} + \frac{\Delta_y}{D_o}} \tag{2.1}$$

The symbols used herein are:

$\Phi_x = \Phi$(x-neighbour); $\Phi_y = \Phi$(y-neighbour)

$\Phi_x \cong \Psi(R_x)$; $\Phi_y \cong \Psi(R_y)$

$\Psi =$ transport-flux from 1-D S-N calculation

R_x, R_y = radii in S-N calc. corr. to dist.: neighbour-mesh to absorber

$\Phi_a = \overline{\Phi}$(absorber region) $\cong \overline{\Psi}_a$

$F_y =$ surface to x-neighbour $= 2pa$

$F_x =$ surface to y-neighbour $= 2a$

$\delta_x =$ distance from absorber center to x-surface $= 1/2\, a$

$\delta_y =$ distance from absorber center to y-surface $= 1/2\, pa$

$\Delta_x =$ distance x-surface to x-neighbour center $= 1/2\, p_x a$

$\Delta_y =$ distance y-surface to y-neighbour center $= 1/2\, pp_y a$

a, p, p_x, p_y = parameters of CITATION mesh geometry description (c.f. Fig. 1)

$D_a =$ Diffusion constant in absorber region

$D_o =$ Diffusion constant in neighbours

Making this leakage equal to the transport leakage and assuming

$\Phi_x = \Psi(R_x)$ and $\Phi_y = \Psi(R_y)$

leads to an equivalent diffusion constant as follows:

$$1/D_a = 2p\frac{\Phi_x - \Phi_a}{L} + \frac{2}{p}\frac{\Phi_y - \Phi_a}{L} - \frac{p_x + p_y}{2D_o} + \sqrt{R} \tag{2.2}$$

with

$$R = \left(2p\frac{\Phi_x - \Phi_a}{L} + \frac{2}{p}\frac{\Phi_y - \Phi_a}{l} + \frac{p_x - p_y}{2D_o}\right)^2 + 4p(p_y - p_x)\frac{\Phi_x - \Phi_a}{L}\frac{1}{D_o}$$

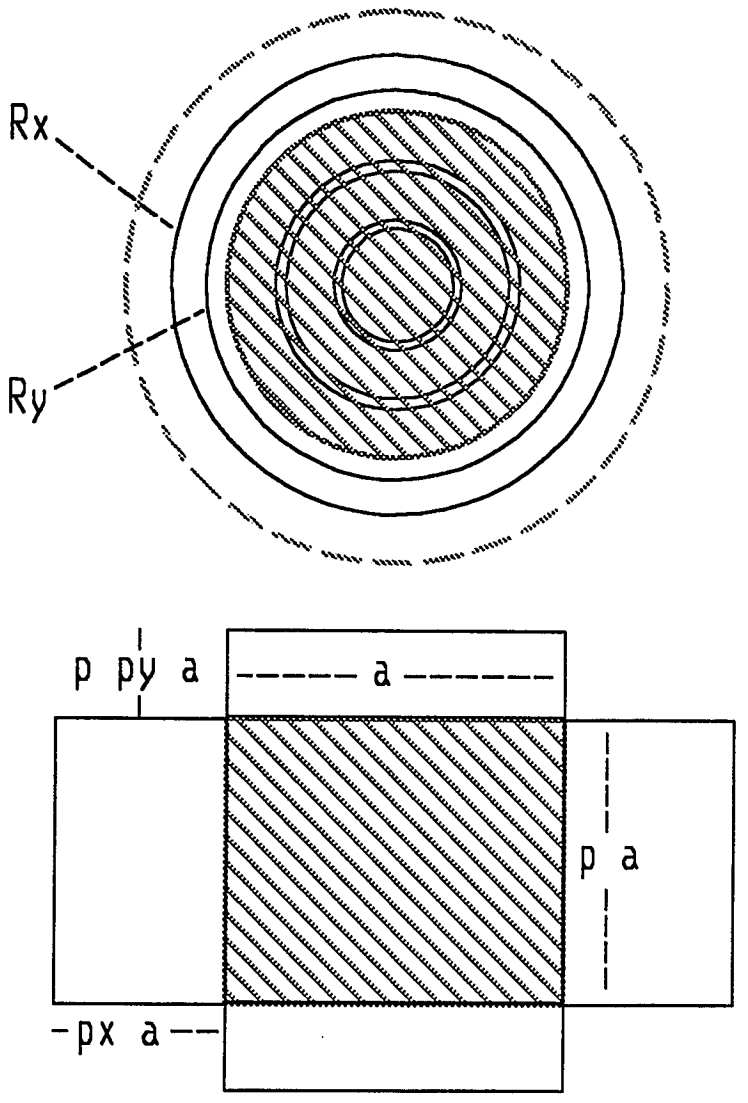

Fig. 1. Equivalent Cross-Sections: Geometric relations for XSDRNPM and CITATION

In the same way an equivalent inner boundary condition $C_s = -\frac{D_o}{\Phi_s}\frac{\partial \Phi}{\partial r}\Big|_s$ for the absorber region can be derived. In that case the CITATION leakage is given by:

$$L = \frac{F_y \Phi_x}{\frac{1}{C_a} + \frac{\Delta_x}{D_o}} + \frac{F_x \Phi_y}{\frac{1}{C_a} + \frac{\Delta_y}{D_o}} \tag{2.3}$$

This leads to the equivalent boundary condition

$$1/aC_a = \frac{\Phi_y + p\Phi_x}{L} - \frac{p_x + pp_y}{4D_o} + \sqrt{S} \qquad (2.4)$$

with

$$S = \left(\frac{\Phi_y + p\Phi_x}{L} + \frac{p_x - pp_y}{4D_o}\right)^2 - p(p_x - pp_y)\frac{\Phi_x}{L}\frac{1}{D_a}$$

This boundary condition differs from the standard evaluation corresponding to

$C_s = \frac{j_s}{\Phi_s}$ j = net current at the absorber region surface

from the 1-D transport calculation. Like the equivalent diffusion constant it reflects the mesh structure and the solution method of the CITATION code and improves the absorber model.

That completes the methods used today at KFA-ISR to describe absorber regions within the CITATION diffusion code.

3.0 Response-Matrix Method

3.1 Description of the Method

The main idea of the Response-Matrix method (RMM) as applied to regions with strong neutron absorbing materials in neutronics calculations is rather simple. The distribution of the neutron flux all over core and reflector is calculated by means of a diffusion code (e.g. the finite element code DTFEM) , but regions with great absorption and significant changes in the diffusion coefficient are separated from the main calculation and their influence is taken into account by means of nonlinear boundary conditions at the surface of separated sub-zones. These boundary conditions are the response matrices prepared by some computer code based on solving the transport equation. The calculation domain Ω (Fig. 2 on page 10) is subdivided into cells in such a manner, that every region with great absorption occurs in a separated cell. All over the domain Ω the standard diffusion equation is solved:

$$-\nabla D^g \nabla \Phi^g + \Sigma^g \Phi^g = Q^g \tag{3.1}$$

where

D^g - diffusion coefficient in group g;
Φ^g - neutron flux in group g;

$\Sigma^g = \Sigma_a + \sum_{j=1; j\neq g}^{G} \Sigma^{g\to j}$ - removal cross-section from group g;

$Q^g = 1/k_{eff}\chi^g \sum_{j=1}^{G} (\nu\Sigma_f)^j \Phi^g + \sum_{j=1; j\neq g}^{G} \Sigma^{j\to g}\Phi^g$ - neutron source in group g;

χ^g - fission neutron spectrum;
G - number of neutron energy groups.

The nondiffusion region ABCD inside the whole calculation area Ω is separated and nonlinear boundary conditions are put along the surface Γ:

$$0.25\,\Phi - 0.5\,D\,\nabla\Phi \cdot \vec{n}\,|_{\Gamma} = J^{+}(\Gamma) \tag{3.2}$$

where
$\vec{n}$ - vector of external normal to the Ω region;
Γ - surface of the ABCD region;
J^+ - current, coming into the domain Ω from the ABCD nondiffusion area.

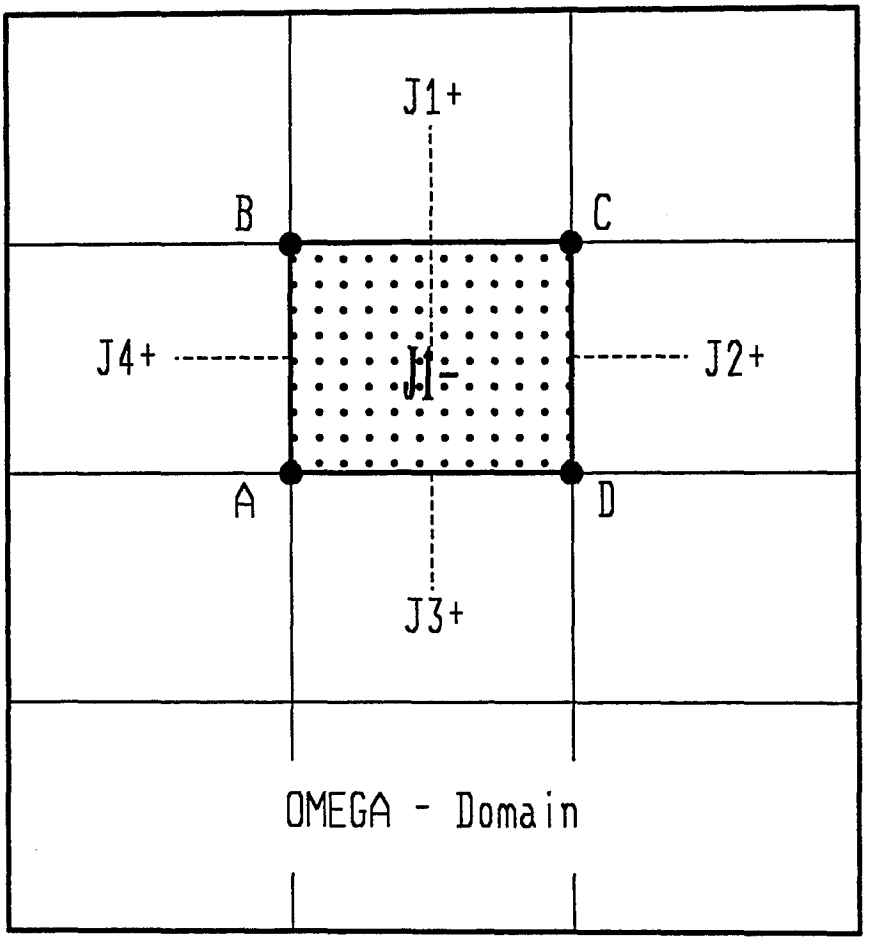

Fig. 2. Response Matrix Method: Geometric relations

Currents J_n^+ depend from J_m^- currents and can be determined with the help of the response matrix. This matrix characterizes the properties of the medium in the closed region.

We define the response matrix as :

$$A_{nm}^{jk} = \frac{J_n^{+j}}{J_m^{-k}} \text{ , (j,k = 1...G, n,m = 1...N),} \tag{3.3}$$

where

N - number of spatial intervals along the surface Γ;

J_n^{+j} - external current in energy group j coming into the domain Ω from the ABCD area through the spatial interval n on the surface and averaged over the interval ; these currents in every of N spatial intervals and in every energy group are the results of the outgoing current of neutrons in spatial interval m and averaged over this interval in energy group k, J_m^{-k}

Thus, it is shown in Fig. 2 on page 10 that in the case of one energy group and 4 intervals on the surface we have 4 currents coming into the domain Ω as a result of the outgoing current J_1^- in the first interval. In case of several energy groups, we must carry out the whole series of calculations for every energy group using the outgoing currents J_m^{-k}.

3.2 The Response-Matrix Code MCCG

The program MCCG (Method of Characteristics in Complex Geometry) is a two-dimensional multi-energy-group computer code, designated for calculations of the spatial and energy dependent neutron distribution in a nuclear reactor core or reactor shielding for irregular geometry /14/. The method of characteristics is similar to that one, described in /15/. An universal geometry block, used in the program, provides the use of arbitrary geometry cells or domains, that can be described by short-length lines and by parts of circles. This provides opportunities, comparable with opportunities of the Monte-Carlo method in describing of geometry cells. Scattering is taken into consideration in transport approximation P_0.

The program is written in Fortran and works in IBM MVS/CMS systems. It uses 2 Megabytes operational memory.

The program solves the transport equation:

$$(\vec{\Omega} \cdot \nabla)\Psi^g(r,\Omega) + \Sigma_{tr}^g - \Sigma_{tr,s}^{g \to g}(r)\Phi^g(r) =$$
$$= \sum_{g'=1;\, g' \neq g}^{G} \Sigma_s^{g' \to g}(r)\Phi^{g'}(r) + \frac{\chi^g(r)}{k_{eff}} Q(r) + S^g(r,\Omega) \qquad (3.4)$$

where

G – number of neutron energy groups;

$\Psi^g(r, \Omega)$ – angle flux in energy group g at point r in direction $\vec{\Omega}$;

$\Phi^g(r) = \int_\Omega \Psi^g(r, \Omega) d\Omega$ – scalar flux in energy group g at point r;

$Q(r) = \sum_{g=1}^{G} \nu\Sigma_f^g(r)\Phi^g(r)$ – source of secondary neutrons;

$S^g(r, \Omega)$ – external surface- or spatially distributed source;

$\Sigma_{tr}^g(r)$ – transport cross section in energy group g at point r;

$\Sigma_{tr,s}^{g \to g}(r)$ – in-group scattering cross-section;

$\Sigma_s^{g' \to g}(r)$ – scattering cross-section from g' into group g;

$\nu\Sigma_f(r)$ – neutron production cross-section at point r in energy group g;

$\chi^g(r)$ – neutron fission spectrum;

k_{eff} – effective neutron multiplication factor.

After that the fluxes obtained are used to form the response matrix as defined in eq. (3.3) by integration and averaging procedures.

3.3 The Finite-Element Diffusion Code DTFEM

The DTFEM program /16,17/ is a two dimensional finite element multi-energy-group diffusion program designated for solving eigenvalue problems and problems with external sources of neutrons in R-Z, X-Y, R-ϕ geometries. The program has the ability of installing different boundary conditions: reflection, zero-flux, albedo, vacuum and, - what is the most important in our case - , response matrix on the surfaces of regions with great absorption. The program solves the standard multi-group diffusion equation (3.1), as described before. The code has been developed in Obninsk Institute of Nuclear Power Engineering (USSR). It demands 600 Kilobytes of operational memory in overlay regime and is installed on a personal computer IBM PC/AT. The program needs 6 logical units for input-output and working areas, which are allocated on hard-disk.

The program DTFEM has been tested in comparison with the MMKFK /20/ and FINELM programs /18,19/.

3.3.1 Sample problem with analytical solution

At first a comparison was made with the code FINELM. FINELM is a two-dimensional finite element code for multi-group diffusion calculations. The code has

been extensively tested. We took the simple test from /18/, where the results of calculations by means of different codes and analytical results are available /21/. The geometry of the test case is shown in Fig. 3 on page 13. The cross sections are given in Tab. 1 on page 13.

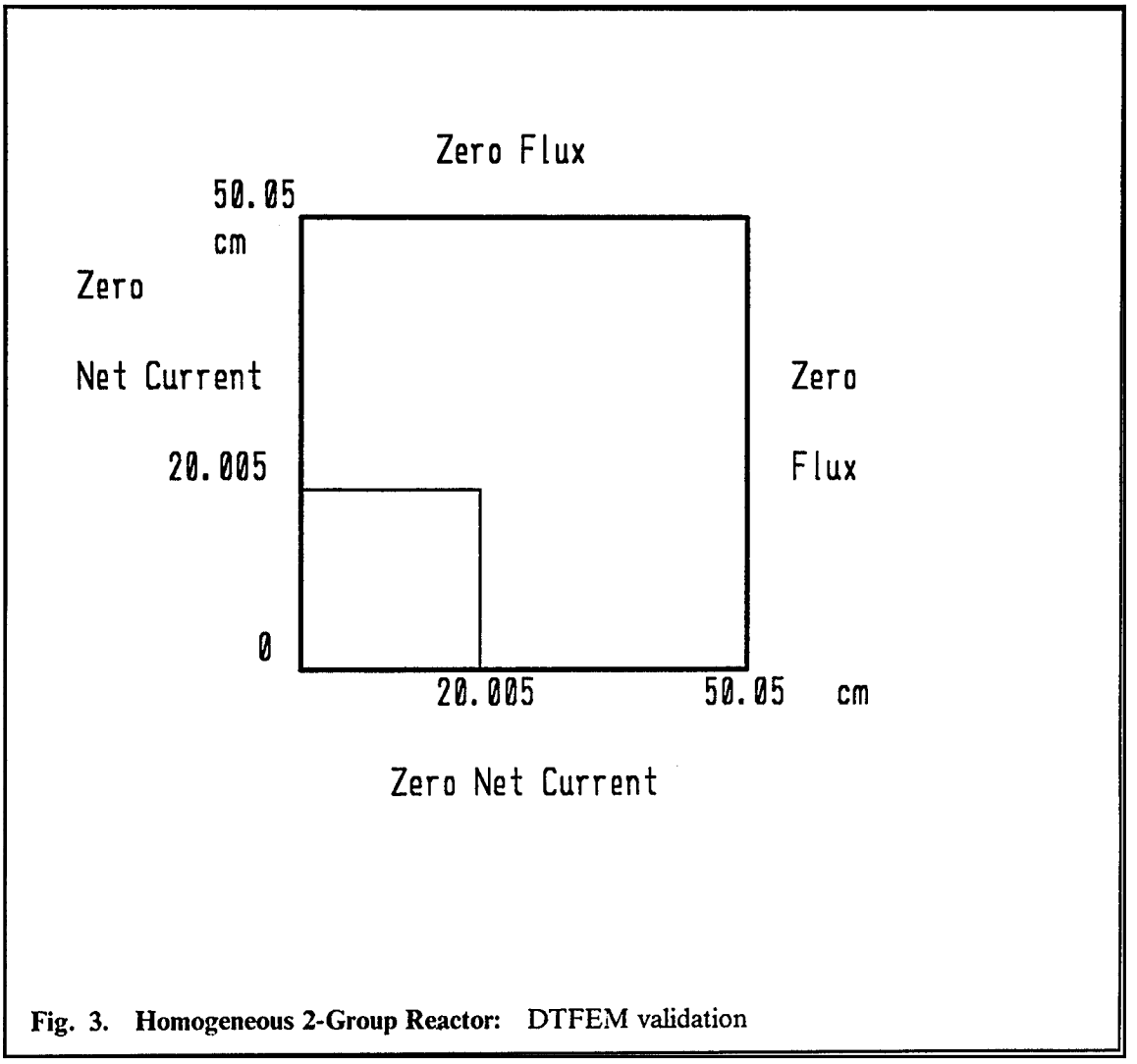

Fig. 3. Homogeneous 2-Group Reactor: DTFEM validation

Grp.	D	Σ_a	$\nu\Sigma_f$	$\Sigma^{1\to2}$	ν
1	2.6800045	5.4577038	0.030834	4.079207	.575
2	1.5787672	1.4496088	0.025200		.425

Tab. 1. 2-Group Homogeneous Reactor: Macroscopic cross sections

The simple two-group homogeneous k_{eff} problem has been chosen by the fact that k can be determined analytically and also because the computed values for k determined by the

finite element code HOD (High Order Diffusion) have been published by the Argonne National Laboratory /21/.

The results obtained by the different methods are sumarized in table Tab. 2.

Method	Nodes	k_{eff}
analytic		1.4653874
FINELM-2D	169	1.4653872
HOD (Argonne)	127	1.4653869
HOD (Argonne)	271	1.4653873
DTFEM		1.46538**

Tab. 2. DTFEM Validation (1): k_{eff} for 2-group homogeneous reactor

It should be noted that we used DTFEM release 1.1 with common computational accuracy. Nevertheless, errors are obtained only in the 6-th figure after the decimal point in the k-value, as compared to the above mentioned codes.

3.3.2 A two group problem related to a real modular HTR.

Two series of calculations were performed with a geometry similar to a real modular HTR as the second test. The first series was a one-dimensional model of the modular HTR with homogeneous core and reflector. The second one was a model with control rods in the reflector and the worth of the control rods was determined. Here, the response matrix method was used. It should be noticed, that these tests were performed in two energy groups and that they used cross-sections specially prepared in Kurchatov's Institute in Moscow and used for HTR calculations in USSR /22/. The geometry of these tests is presented in Fig. 4 on page 15 and cross-sections are summarized in Tab. 3 on page 16.

The first case (cf. case A of Fig. 4 on page 15) represents a sector of 7.5 degrees, i.e. 1/48 part of the reactor with two homogeneous zones, core and reflector.

In the second case all data are equal to case 1 except the core radius, which was changed from 150 cm to 140 cm.

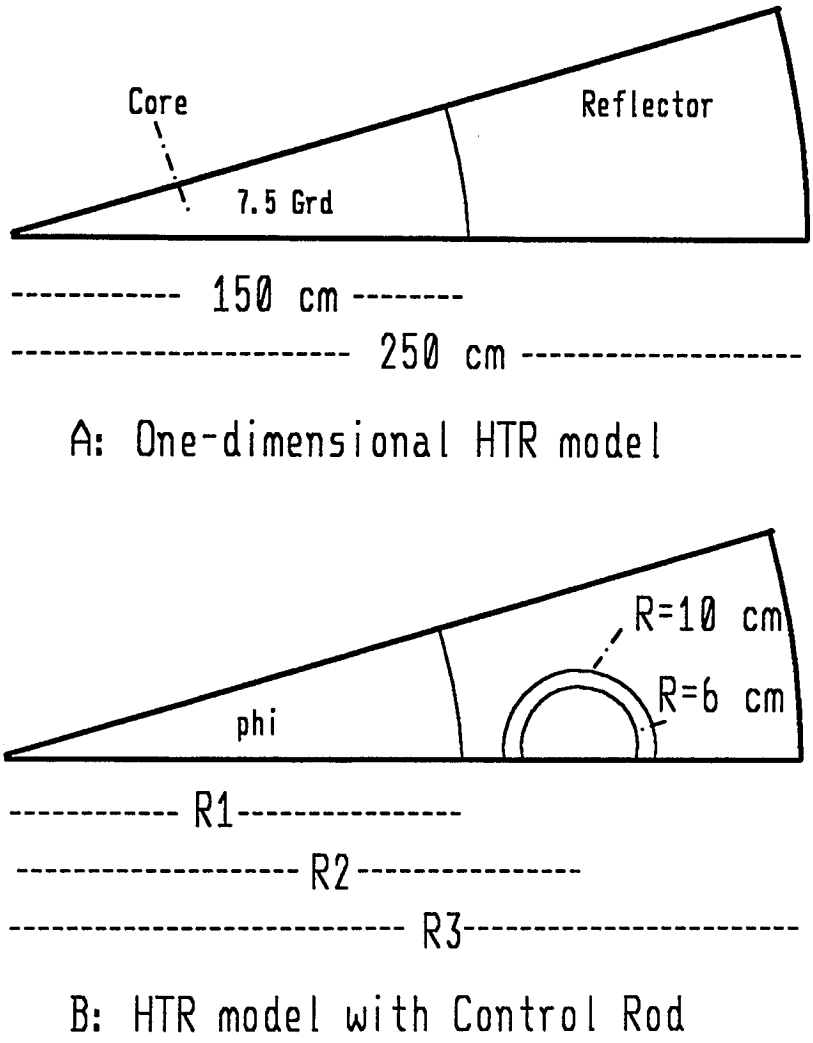

Fig. 4. 2-Group HTR Models: Preliminary test calculations

The DTFEM calculation for these configurations has been tested by means of the MMKFK program /20/. The MMKFK program is a Monte-Carlo computer code, designed for reactor calculations and based upon Academician A.D. Frank- Kameneckij ideas. It was developed at the Physical Energetic Institute, Obninsk, USSR.

CRX	Core	Refl.	Steel	B_4C	Helium
Σ^1_a	2.1063E-3	2.78E-3	4.43E-3	2.4391E-1	5.054E-4
Σ^2_a	2.06E-3	1.49E-4	1.446E-1	1.9674E1	5.03E-4
Σ^1_{tr}	2.1929E-1	3.3003E-1	1.02043E-0	5.247E-1	7.04E-4
Σ^2_{tr}	2.3981E-1	3.4614E-1	1.3146E-0	1.9915E1	8.73E-4
Σ^1_s	2.1718E-1	3.2725E-1	1.0160E-0	2.808E-1	2.01E-4
Σ^2_s	2.3775E-1	3.4599E-1	1.1700E-0	2.414E-1	3.24E-4
$\Sigma^{1\to2}$	1.68E-3	2.64E-3	1.662E-3	2.514E-3	2.4E-6
$\nu\Sigma^1_f$	9.13E-5				
$\nu\Sigma^2_f$	2.87E-3				

Tab. 3. HTR 2-Group Cross Sections: Kurchatov Institute, Moscow

The results are shown in Tab. 4.

CASE	DTFEM	MMKFK
I	1.02139	1.0204±0.0039
II	1.00959	1.0094±0.0043

Tab. 4. DTFEM Validation (2): k_{eff} for homogeneous HTR-problem

So, the DTFEM computer code is in a good accuracy for our purpose, and there will be no great errors in RMM calculations, connected directly with DTFEM computational errors.

The next cases represent a HTR's model with control rods in the reflector. Generally, reflector rods may be positioned near the boundary to the core or in the middle of the reflector. In the second case the rods are used as control system only and KLAKs may be positioned near the core and used as shut-down system. According to these possibilities several variants have been looked at. The general geometry is shown in case B of Fig. 4 on page 15.

1. ϕ = 10 deg., (18 rods), R_1 = 140 cm, R_2 = 155 cm, R_3 = 240 cm
2. ϕ = 10 deg., (18 rods), R_1 = 140 cm, R_2 = 190 cm, R_3 = 240 cm
3. ϕ = 7.5 deg., (24 rods), R_1 = 150 cm, R_2 = 165 cm, R_3 = 250 cm

Some of these variants were tested by the MMKFK program. All results are given in Tab. 5 on page 17.

CASE	DTFEM		MMKFK	
	no rods	*with rods*	*no rods*	*with rods*
1	1.00559	0.92557		0.9067 ± 0.0043
2	1.00583	0.98114		
3	1.00707	0.91280	1.0165±0.0059	0.9198±0.0042

Tab. 5. DTFEM and MCCG validation: k_{eff} for heterogeneous HTR-problem

For case 1 there is a difference of about 2 Nile for the rodded configuration between the Monte-Carlo-Code and the RMM-method. Unfortunately the clean situation has not been calculated with the MMKFK code in this case, so a comparison of the control rod worth is not possible here. In case 3 Monte-Carlo results are larger than the RMM values by about 1 Nile but the control rod reactivity, which is $\rho = 10.3$ Nile differs only by about 1 % of its value. With respect to the control rod worth this has to be looked at as an excellent result.

4.0 Benchmark Type Calculations for Modular HTR

4.1 Geometry and Cross-Section Basis

As it was mentioned before, the aim of this analysis is to illustrate the reliability of different calculation methods with respect to reactor safety. In this connection, we chose the model for our detailed calculations close to a real HTR design. All benchmark-like calculations were performed in four energy groups as they are usually used in HTR reactor design studies. The reactor design is close to both the german MODUL-HTR and the russian VGM and was defined some years ago between KFA-ISR and Kurchatov Institute /23/. A VSOP /24/ calculation generated the equilibrium core composition and therefrom macroscopic cross-sections were taken for a 3-D diffusion calculation using the CITATION code. For our purposes here a horizontal 2-D slice was extracted from these data and incorporated into a R-ϕ calculation scheme. Radially variing axial bucklings have been deduced from the 3-dimensional CITATION calculation. The geometry of this model is shown in Fig. 5 on page 20.

The Core in the model consists of six homogeneous radial annular regions with different macroscopic cross-section sets due to different burnup and temperature. Six different graphite type annuli define the radial reflector. The outermost reflector region which in the original design study was heavily boronated material has been exchanged by normal graphite to avoid additional uncertainties related to the use of diffusion theory in this region. For simplicity the same material as for the fifth reflector zone was taken for the outer sixth reflector zone. The four-group cross sections for core and reflector zones are given in Tab. 6 on page 21.

The reactor control system in our calculation model is represented by eighteen control rods and eighteen KLAK systems. So, a sector of twenty degrees (ϕ = 20°) was chosen as the region of the reactor's symmetry. Here, both control rod and KLAK borings are represented as circles with a radius of 6 cm placed at symmetric azimuthal positions but at different radii, i.e.

- KLAK at ϕ = 5 °, r = 155 cm ;
- Control Rod at ϕ = 15 °, r = 160 cm.

In MECS (Method of Equivalent Cross-Sections) calculations control rod and KLAK are represented by equivalent pseudo-rectangles with the same area as a circle of 6 cm, because naturally the CITATION code doesn't allow to represent circles in the calcu-

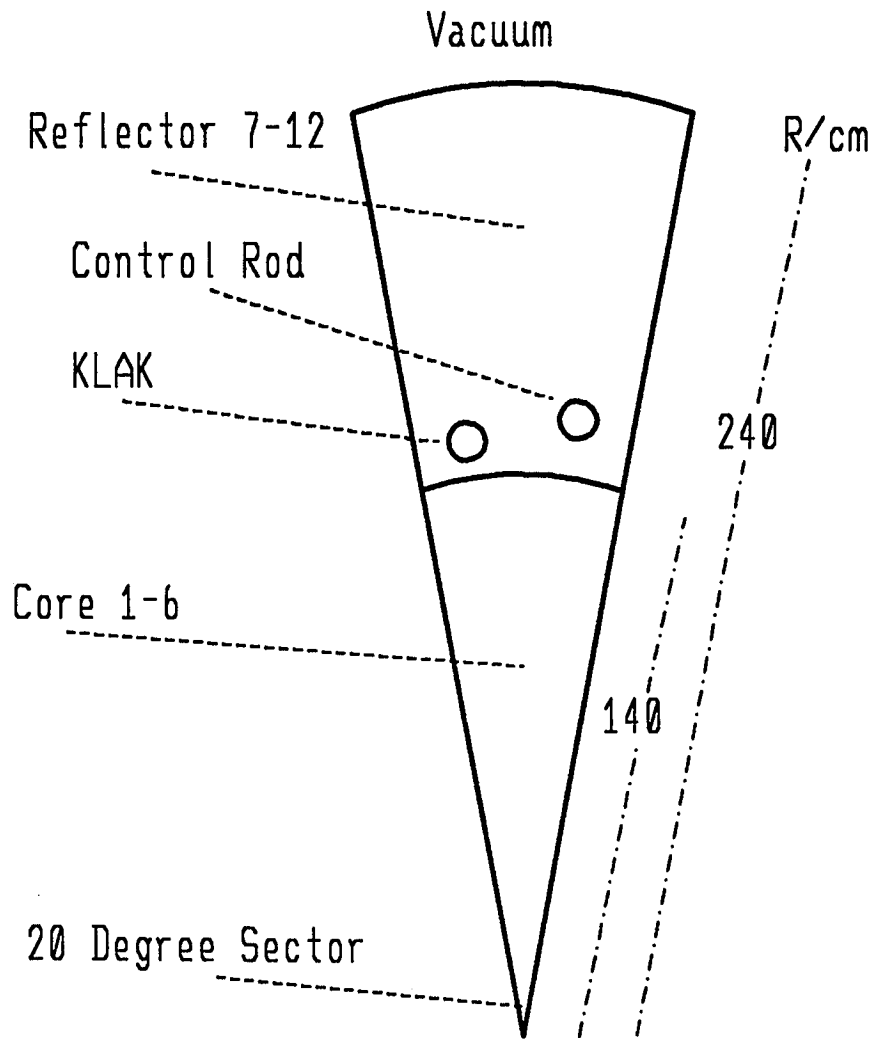

Fig. 5. MODUL Absorber Benchmark: Geometric overview

lation domain. So the control elements have been modified in the CITATION treatment and are represented by regions with limits given in Tab. 7 on page 23.

Grp.	D	Σ_R	Σ_a	$\nu\Sigma_f$	B^2
Core 1					
1	2.92712E-0	8.71259E-3	3.21705E-5	4.13047E-5	-6.51238E-6
2	1.69547E-0	4.08351E-3	4.59084E-4	1.14489E-4	-1.61179E-6
3	1.67152E-0	1.27101E-2	2.56893E-3	4.14177E-4	-2.45649E-6
4	1.67090E-0		2.14420E-3	3.11700E-3	8.56749E-6
Core 2					
1	2.92712E-0	8.71259E-3	3.21705E-5	4.13047E-5	-6.50829E-6
2	1.69547E-0	4.08351E-3	4.59084E-4	1.14489E-4	-1.60414E-6
3	1.67152E-0	1.27101E-2	2.56893E-3	4.14177E-4	-2.45089E-6
4	1.67090E-0		2.14420E-3	3.11700E-3	8.58606E-6
Core 3					
1	2.92712E-0	8.71259E-3	3.21705E-5	4.13047E-5	-6.47194E-6
2	1.69547E-0	4.08351E-3	4.59084E-4	1.14489E-4	-1.59272E-6
3	1.67152E-0	1.27101E-2	2.56893E-3	4.14177E-4	-2.43546E-6
4	1.67090E-0		2.14420E-3	3.11700E-3	8.68273E-6
Core 4					
1	2.92712E-0	8.71259E-3	3.22101E-5	4.13836E-5	-6.44741E-6
2	1.69546E-0	4.08352E-3	4.59374E-4	1.15349E-4	-1.62554E-6
3	1.67152E-0	1.27101E-2	2.56799E-3	4.16245E-4	-2.37825E-6
4	1.67093E-0		2.13855E-3	3.11628E-3	8.96065E-6
Core 5					
1	2.93719E-0	8.48883E-3	3.25598E-5	4.18510E-5	-6.51343E-6
2	1.69537E-0	3.79863E-3	4.34618E-4	1.11317E-4	-1.84771E-6
3	1.67039E-0	1.24348E-2	2.55492E-3	4.17248E-4	-2.08380E-6
4	1.67259E-0		2.13434E-3	3.12156E-3	9.06803E-6
Core 6					
1	2.93716E-0	8.48892E-3	3.25127E-5	4.17069-5	-7.99585E-6
2	1.69535E-0	3.79867E-3	4.34243E-4	1.10651E-4	-2.25621E-6
3	1.67038E-0	1.24349E-2	2.55321E-3	4.13696E-4	-2.14509E-6
4	1.67282E-0		2.09423E-3	3.05655E-3	8.31876E-6

Macroscopic Cross-Sections for Diffusion Calculation

Grp.	D	Σ_R	Σ_a	$\nu\Sigma_f$	B^2
Reflector 7					
1	1.52410E-0	1.76845E-2	6.48953E-6		-7.44644E-6
2	8.38938E-1	1.13624E-2	0.00000E-5		-2.58328E-6
3	8.33712E-1	2.72139E-2	2.00180E-5		-1.84545E-6
4	8.22467E-1		1.89496E-4		6.36891E-6
Reflector 8					
1	1.49568E-0	1.80204E-2	6.61283E-6		-6.12556E-6
2	8.23298E-1	1.15783E-2	0.00000E-5		-2.21198E-6
3	8.18168E-1	2.77309E-2	2.03983E-5		-1.24314E-6
4	8.07134E-1		1.93095E-4		5.48071E-6
Reflector 9					
1	1.53565E-0	1.60530E-2	8.65412E-6		-3.99339E-6
2	8.34655E-1	4.08757E-3	0.00000E-5		-9.71203E-7
3	7.93470E-1	4.34625E-3	1.47852E-5		8.19690E-8
4	8.26503E-1		5.30023E-5		5.12101E-6
Reflector 10					
1	1.97449E-0	1.24851E-2	6.73067E-6		-1.87092E-6
2	1.07318E-0	3.17908E-3	0.00000E-5		1.36466E-7
3	1.02022E-0	3.38026E-3	1.14991E-5		9.19357E-7
4	1.06269E-0		4.12222E-5		4.99816E-6
Reflector 11					
1	1.53565E-0	1.60530E-2	8.65412E-6		-7.99822E-7
2	8.34655E-1	4.08757E-3	0.00000E-5		7.73539E-7
3	7.93470E-1	4.34625E-3	1.47852E-5		1.31833E-6
4	8.26503E-1		5.30024E-5		4.98359E-6
Reflector 16					
1	1.53565E-0	1.60530E-2	8.65412E-6		-7.99822E-7
2	8.34655E-1	4.08757E-3	0.00000E-5		7.73539E-7
3	7.93470E-1	4.34625E-3	1.47852E-5		1.31833E-6
4	8.26503E-1		5.30024E-5		4.98359E-6
Macroscopic Cross-Sections for Diffusion Calculation					

Tab. 6. Macroscopic Cross-Sections (1): Core and reflectors

Figure	Control Rod	KLAK
inner Radius/cm	155.3	150.0
outer Radius/cm	165.9	160.6
left angle/°	13.097	3.032
right angle/°	16.903	6.968
RODCIT Parameters:		
a/cm	10.670	10.670
p	0.993	0.993
p_x	0.768	0.768
p_y	0.500	0.500

Tab. 7. Method of Equivalent Cross-Sections: CITATION absorber representation

Four group cross-section sets for the different absorber materials have been condensed from a 123 group library for the hot state of the reactor, using the AMPX-II system 1-D S-N module XSDRNPM /12/. In these calculations the control rod is modelled as shown in Fig. 6 on page 24.

In the unit cell the rod's guide hole is surrounded by some 6 cm of graphite and an additional region of a mixture of core material and reflector graphite. For the KLAK system a homogeneous mixture of B_4C and C with a filling factor of 0.6 is assumed to fill the hole. No double heterogeneity effects have been accounted for.

Using the 123-group spectrum cross sections are then condensed and homogeneized over the appropriate region. For the MECS this was the hole itself exept for the diffusion coefficient, which is calculated by the RODCIT code as described in chapter 2. The equivalent cross sections as they were used in the CITATION diffusion calculation are listed in Tab. 8 on page 24.

For the RMM code MCCG cross sections were produced for the different material zones of the absorber regions including an additional surrounding zone comparable to the MECS procedure. These later are taken by the MCCG code to produce the response matrices. The 4-group cross-section sets with P_0 scattering for both absorbers used for these purposes are summarized in Tab. 9 on page 27.

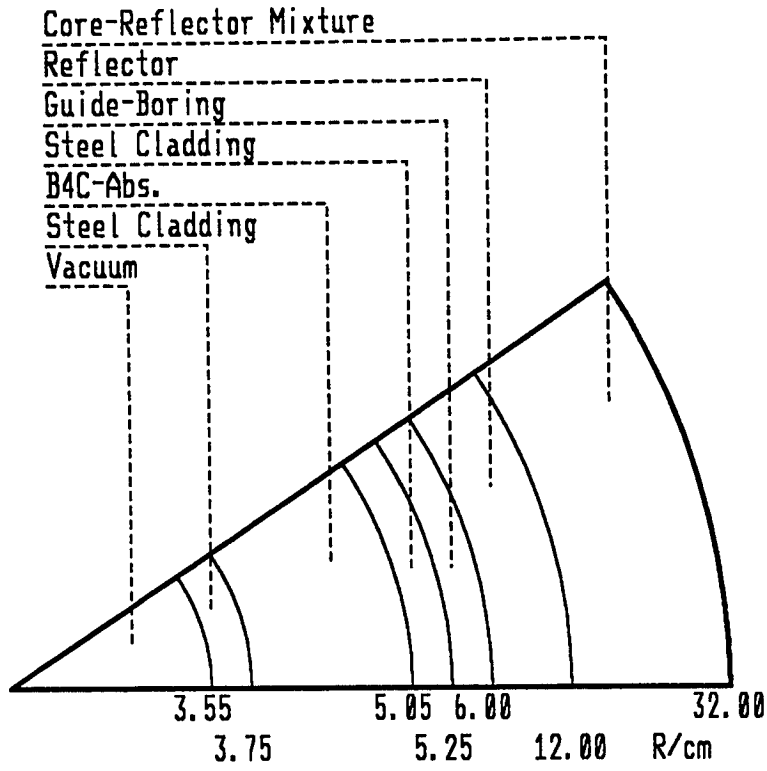

Fig. 6. S-N Cell Geometry for XSDRNPM: Equivalent cross section method

Grp.	D	Σ_R	Σ_a
Control-Rods			
1	2.00128E-0	5.49209E-3	3.41992E-3
2	2.26507E-0	6.25047E-4	5.48200E-2
3	1.55530E-0	3.02282E-3	2.87362E-1
4	1.45740E-0		3.46340E-1
Small Absorber Spheres (KLAK)			
1	2.20234E-0	9.79120E-3	6.66570E-3
2	2.76597E-0	6.34140E-4	1.01940E-1
3	2.16677E-0	6.95629E-3	1.72620E-0
4	1.91834E-0		1.63430E+1

Tab. 8. Macroscopic Cross-Sections (2): Equivalent cross-sections for absorbers

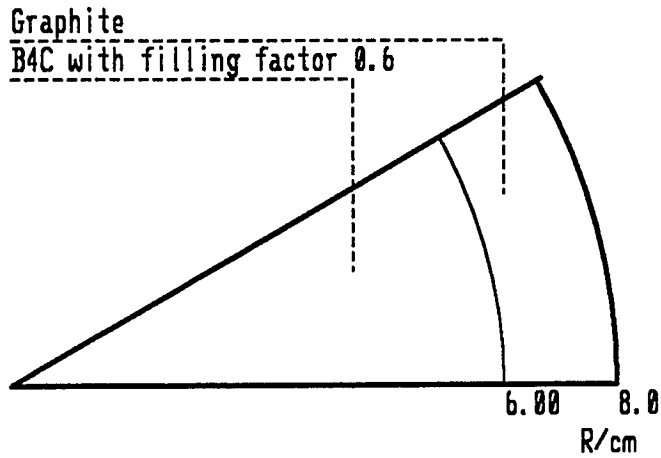

Fig. 7. Cell Geometry for RMM KLAK System: Response matrix method

In the RMM calculations control rods and KLAK are circular nondiffusion regions with a radius of 8 cm. Increasing the radius to more than the physical bore hole seems to improve the calculation scheme, because adding some part of the surrounding reflector allows to calculate the partial currents in one and the same material medium, and not on the boundary between different materials. Thus we avoid errors, connected with some effects, taking place on the surface of two media, and errors, connected with great gradients of flux near the surface of the control element. The response matrix calculations using the MCCG code represented a special part of the work. Response matrices for the control rod and the KLAK system have been calculated in advance of the 2-D diffusion calculations, according to the definition given in chapter 3. Geometrical structures of the control rod and KLAK were taken into consideration in this step as shown in Fig. 8 on page 26 and Fig. 7.

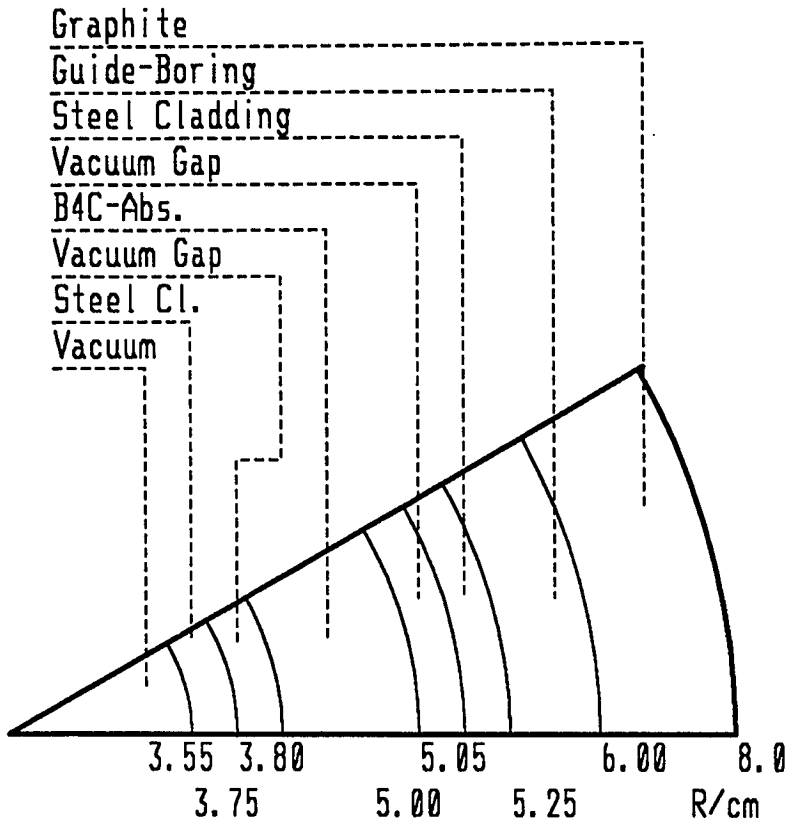

Fig. 8. Cell Geometry for RMM Control Rod: Response matrix method

Grp.	Σ_a	Σ_t	$\Sigma_{G\to G}$	$\Sigma_{G\to G+1}$
Control Rod Inner Steel Cladding				
1	1.8095E-3	3.3379E-1	3.2701E-1	4.9693E-3
2	4.0892E-3	1.1869E-0	1.1824E-0	3.7385E-4
3	1.4038E-2	1.1560E-0	1.1418E-0	1.3581E-4
4	4.3932E-2	1.1688E-0	1.1249E-0	
Control Rod Boron Carbide Absorber				
1	1.1187E-2	2.6259E-1	2.3417E-1	1.7237E-2
2	2.0513E-1	5.8008E-1	3.7335E-1	1.5982E-3
3	2.8306E-0	3.2199E-0	3.7787E-1	1.1428E-2
4	1.8782E+1	1.8865E+1	8.3000E-2	
Control Rod Outer Steel Cladding				
1	1.8100E-3	3.3451E-1	3.2757E-1	5.1278E-3
2	7.5964E-3	1.1990E-0	1.1886E-0	2.7944E-3
3	2.1730E-2	1.1633E-0	1.1285E-0	1.3069E-2
4	1.5314E-1	1.2806E-0	1.1275E-0	
KLAK Boron Carbide Absorber				
1	6.6657E-3	1.3394E-1	1.1748E-1	9.7912E-3
2	1.0195E-1	4.1590E-1	3.1332E-1	6.3414E-4
3	1.7262E-0	1.9870E-0	2.5384E-1	6.9563E-3
4	1.6343E+1	1.6472E+1	1.2940E-1	

Tab. 9. Macroscopic Cross-Sections (3): Detailed cross-sections for absorbers in RMM

4.2 Reference S-N Transport and Monte Carlo Calculations

Both methods to describe strong absorbers in HTR reactor design and safety calculations use combinations of transport and diffusion theory. This facilitates the incorporation of these methods into commonly used diffusion based large reactor life following code systems as for example the VSOP or the HTR-2000 /25/ programs. Naturally some discrepancies are to be expected between the combined methods and purely transport theory based calculations. To get an estimation on the errors in this respect a comparison has been made using the 2-D S-N transport code DOT-IV /7/ and the Monte-Carlo code MORSE-CG /8/.

In the DOT code the geometry was modelled in the same way as in the CITATION code exept for finer mesh specifications. Especially the absorber holes were represented as

pseudo-rectangles as in CITATION. The 4-group macroscopic cross sections remained unchanged in principle. The transport cross section was assumed to define the total reaction cross section and the in-group scattering cross section was calculated by regarding the total neutron balance.

The technique of pointwise rebalancing was used to accelerate convergence. However, convergence was not optimal, which is normal in using S-N methods for large, low absorbing graphite reactor systems.

The MORSE code allows to use the same data base as the DOT program. With respect to geometry the combinatorial package has been used. Mirror reflection was used at the symmetry surfaces of the region under consideration. No time kill, splitting or energy kill option was used, but russian roulette was played to avoid moving around of neutrons with very low statistical weights.

Nevertheless these calculations were very hard to accomplish. Despite of running times in the order of one hour on the KFA IBM ES/9000 mainframe the statistical error could not be decreased under about half a percent. This is by far to bad for an exact comparison with the combined methods. The only result of interest was to see, wether the error margin covered the other results or not.

So, it was decided to define the DOT results as some sort of reference for the test of the other methods.

The results of the pure transport calculations are summarized together with the other methods in Tab. 10 on page 29, Tab. 11 on page 30 and Tab. 12 on page 31.

4.3 Results of the Different Methods and Discussion

All eigenvalue results received are presented in Tab. 10 on page 29. From these reactivity effects for the different configurations of the absorbers were deduced as shown in Tab. 11 on page 30. The differences in the calculated absorber reactivity values for both combined methods are put together in Tab. 12 on page 31.

Basing on these results a comparison of the different methods can be made. The calculations performed have included five different situations to determine the efficiency of the control rod and KLAK system.

	Transport		Combined	
Example	**MORSE**	**DOT-IV**	**Equival. Cross-Sections**	**Response Matrix**
no absorbers, no holes	1.0298±0.0037	1.03015	1.03274	1.03323
no absorbers, with holes	1.0230±0.0056	1.02265	1.02564	1.02547
with Control-Rods	0.9125±0.0043	0.91121	0.91871	0.91466
with KLAK	0.8870±0.0031	0.88533	0.89179	0.88935
with KLAK + Control-Rods	0.8632±0.0039	0.86110	0.86979	0.86392

Tab. 10. Results of Benchmark Calculations (1): k_{eff} for different methods

The one-dimensional situation without any non-diffusion regions in the reflector goes first. According to these results all codes are in a good agreement between each other and they stay inside the Monte-Carlo error limits.

The second situation is a case when both control rod and KLAK system are removed and we have two holes in the reflector. In MECS all reaction cross sections in the holes are set to zero. The diffusion coefficient has to be calculated for this case in a special way. For large symmetric cavities a theory has been derived at KFA-ISR some years ago /11/. It was shown previously, that this theory is not valid for the situation here under discussion. By comparison with 2-D S-N calculations for the german HTR-MODUL design a relationship has been established and was used here, too.

Two response matrices had to be calculated for this case. They represent themselves as graphite circles of an inner radius of 6 cm and an outer one of 8 cm with vacuum inside. The surface of the response matrix region was dissected in 6 spatial intervals because it intersects a material boundary in the reactor model. The full response matrix in four groups and 6 intervals for the KLAK system was an array of 576 figures. But for the control rod a symmetry is available and it is enough to make calculations only for one interval, for the rest it will be the same. This symmetric matrix is represented only by 96 figures.

As a result of this case, the effect in k_{eff} of opening two holes in the reflector was compared with the different codes relative to the one-dimensional case. If we accept the DOT

calculation as a reference, the difference in the 'hole'-effect for both combined methods is in the range of 5% from the reference and we may accept this as a good result.

	Transport		Combined	
Example	**MORSE**	**DOT-IV**	**Equival. Cross-Sections**	**Response Matrix**
Holes	0.680±0.93	0.750	0.710	0.776
Control-Rods	11.05±0.99	11.144	10.693	11.081
KLAK	13.60±0.87	13.732	13.385	13.612
KLAK + Control-Rods	15.98±0.95	16.155	15.867	16.155

Tab. 11. Results of Benchmark Calculations (2): Reactivity (%) for different methods

The situation three is a case where the data for the hole at the place of the control rod is replaced by the data for the control rod region. Either equivalent cross sections or the appropriate response matrix is used. Thus, the control rod reactivity is determined and compared for the different methods. The excellent agreement of RMM with the reference calculations should be noticed. The difference between the results is less than one percent. The MECS result is a little worse, but 4% difference is still a very good agreement.

The fourth case describes the situation 'KLAK system activated.' The model here is somewhat fictitious, because this situation would normally occur in a cold reactor. For simplicity in our calculations the same database was used as for the control rod insertion case, i.e. the macroscopic reactor cross sections were not changed. The KLAK were represented by their equivalent cross sections or by their response matrix.

The KLAK reactivity is somewhat larger compared to the control rods. This is mainly due to the smaller distance of the KLAK position to the core-reflector boundary.

Again, there is optimal coincidence between RMM and the reference DOT calculation. The MECS result is even better than in the previous case.

The last situation represents the maximum of absorbers in the reactor. It is a very complicated case for diffusion theory, but using RMM and MECS methods allow us to produce results close to the reference in this case, too. The difference for MECS is better than 2% and for RMM it is practically zero.

Example	Equival. Cross-Sections	Response Matrix
Holes	-5.3	+3.5
Control-Rods	-4.0	-0.6
KLAK	-2.5	-0.1
KLAK + Control-Rods	-1.8	0.0

Tab. 12. Results of Benchmark Calculations (3): Errors in reactivity (%)

In total the analysis has shown, that the combined transport/diffusion methods discussed here are able to determine the absorber reactivity in high temperature reactors with very high accuracy even in the case of pure reflector control with large flux gradients and bucklings at the absorber location. The general uncertainty was shown to be in the order of 5% for MECS and about 1% for RMM. Finally it should be recognized, that there is a general tendency to underpredict the absorber efficiency by the combined methods. This adds confidence to these methods with respect to safety analyses.

5.0 Summary

Today, the neutron-physical description of strong neutron absorbing materials for control and shut-down of nuclear power plants is performed using combined transport and diffusion methods. Two of these approaches have been described and compared in this paper.

The method of equivalent cross-sections (MECS) has been developed at the KFA-Jülich Institute for Safety Research and Reactor Technology (ISR) and was widely used for all german HTR reactor concepts. The Obninsk Institute for Nuclear Power Engineering (OINPE) in USSR has worked out the response-matrix method (RMM) for rodded regions. Both, equivalent cross-sections and response-matrix are results of transport calculations. They are subsequently used in multidimensional diffusion type reactor codes for the representation of absorber regions.

The analysis done here showed a validation of these two combined transport-diffusion methods for predicting the neutronic reactivity efficiency of strong absorbers in a high temperature reactor of the modular design type.

Defining some benchmark problem close to realistic HTR designs and comparing the combined methods with pure transport descriptions and codes such as the S-N code DOT-IV and the Monte-Carlo program MORSE-CG showed very good consistency and accuracy of the combined methods. The differences for absorber reactivities relative to pure transport methods are in the range of 5% for MECS and about 1% for RMM.

Now we may say that the results given by those methods can be believed in. This was proven by comparison of two quite different approaches implemented into computer codes at different places in the world independently.

In principle, both methods can be used for threedimensional problems. In fact, the method of equivalent cross sections together with the CITATION code is beeing used at the ISR of KFA Jülich for all questions of HTR safety that involve the determination of reactivity values.

Future development of the DTFEM code with the response matrix method to three-dimensional geometry is going on now at OINPE Obninsk. At appropriate time a similar comparison of the methods in three-dimensional geometry will be done.

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