FIXED SOURCE PROBLEM SOLVER FOR BATAN-2DIFF AND -3DIFF NEUTRON DIFFUSION CODES

Liem Peng Hong

Abstract

FIXED SOURCE PROBLEM SOLVER FOR BATAN-2DIFF AND -3DIFF NEUTRON DIFFUSION CODES. Subprograms for solving fixed source problem of the multigroup neutron diffusion have been successfully developed and integrated into Batan's standard multidimensional, multigroup neutron diffusion codes, Batan-2DIFF and -3DIFF. The inner-outer iteration method has been adopted to solve the fixed source problem. The spatial variables are approximated with finite difference method and the difference equations obtained are solved with the successive over relaxation method. The calculation results of the codes have been verified with other generic diffusion codes. The relative differences of the group neutron fluxes and neutron balances were in the order of $10^{-4}$, while for the maximum power peaking factors were in the order of $10^5$ which proved the validity of the codes.

INTRODUCTION

One of the research objectives of Reactor Physics Division (Center for Multipurpose Reactor), for the first five years of the Second Long Range Development Plan (1994-1999) is the capability of National Atomic Energy Agency (BATAN) to self-develop code systems required for neutronic and thermal-hydraulic designs as well as safety analyses of critical assemblies and research and power reactors.

In the first phase of the efforts to meet the objective, the completion of BATAN standard codes for neutron diffusion is considered to be of the highest priority. Batan-1DIFF [1], -2DIFF [2] and -3DIFF [3] neutron diffusion codes have been successfully developed and they join the line-up of Batan's standard codes. These codes are now being used in reactor design practices in Batan, for example in the on-going Isotope Production Reactor Project.

These Batan's standard diffusion codes have certain features which are not always available in other generic diffusion codes, i.e. they can treat the $(n,2n)$ neutron scattering and directional diffusion constants. Common options for vacuum (zero flux) and reflective (zero gradient flux) external boundary conditions are available. In the codes, the spatial variables are treated with the finite difference method while the standard source iteration method is used to solve the eigenvalue problem of neutron diffusion. In the inner iteration, the codes use the successive over-relaxation (SOR) method.

* Center for Multipurpose Reactor - BATAN
Those diffusion codes were originally developed to solve the regular and adjoint eigenvalue problems (reactor criticality problems) of the multigroup neutron diffusion. However, as the codes are being used by Batan's users for analysing the MPR-30 or in designing an Isotope Production Reactor, several important additional capabilities were demanded. These included the capability of the codes to solve the fixed source or inhomogenous neutron diffusion problems. The inhomogeneous neutron diffusion problems are commonly solved to determine the neutron source strength required during reactor start-up. In the reactor design phase, the strength and the position of the neutron source relative to the start-up detector(s) have also to be surveyed by solving fixed source problems. The fixed source neutron diffusion problems (in modified forms) may also occur in designing and analyses where the neutron source is expressed in a specified source volume or neutron current at the media boundaries.

In this paper, the completion and verification results of subprograms for solving the inhomogeneous or fixed source problem of multigroup multidimensional neutron diffusion is reported. The subprograms were successfully integrated in the existing Batan's standard multidimensional diffusion codes, i.e. Batan-2DIFF and -3DIFF.

**FIXED SOURCE MULTIGROUP NEUTRON DIFFUSION EQUATIONS**

**Neutron Balance Equations**

The fixed source problem of multigroup neutron diffusion can be written in the form (see for e.g. Refs.(2,3))

\[
- \nabla D_g(r) \cdot \nabla \phi_g(r) + \Sigma_{t,g}(r)\phi_g(r) = \sum_{g'=1}^{G} \Sigma_{s,g'\rightarrow g}(r)\phi_{g'}(r)
\]

\[
+ \chi_g \sum_{g'=1}^{G} \Sigma_{f,g'}(r)\phi_{g'}(r) + Q_g(r), \quad (1)
\]

where all notations are commonly used in the standard reactor physics textbooks and will not be discussed here. When \((n,2n)\) neutron scattering is involved, Eq.(1) has to be modified (See Appendix of Ref.(2)).

The left and right hand sides of Eq.(1) represent neutron loss and production terms, respectively. The first and second terms at the left hand side of Eq.(1) denote neutron loss from leakage and total removal from energy group \(g\), while the first and second terms at the right hand side denote...
neutron production from scattering into energy group g and fission neutrons born in energy group g, respectively. The external, group and space-dependent neutron source is represented by $Q_g(r)$.

**Boundary Conditions**

In Batan-2DIFF or -3DIFF code, two kinds of boundary condition commonly used in reactor design, i.e., reflective (zero flux gradient) and vacuum (zero flux) boundary conditions, are available. The reflective boundary condition is expressed as,

$$\nabla \phi_g(r) = 0, \quad (2)$$

while the vacuum boundary condition is expressed as

$$\phi_g(r)\bigg|_{r=0.7104 \lambda_g} = 0. \quad (3)$$

The latter boundary condition forces the group neutron fluxes to be vanish at the extrapolation distance from the reactor media ($0.7104 \lambda_r$).

**SOLUTION OF THE FIXED SOURCE PROBLEM**

**Source Iteration Method**

Batan-2DIFF and -3DIFF codes solve the fixed source neutron diffusion problems to obtain the group neutron flux distributions with an iteration technique quite similar to the source iteration method (or the inverse power method) used in solving the eigenvalue problem. For simplicity without loss of generality, the iteration method will be applied to the directly coupled, multigroup diffusion equations with no up-scattering [4], although the subprograms developed can solve general cases. Now, the inhomogeneous multigroup neutron diffusion equations are written out in more detail as

$$-\nabla D_1(r) \cdot \nabla \phi_1(r) + \sum_{r,1}(r) \phi_1(r) = \chi_1 S_f(r) + Q_1(r)$$

$$-\nabla D_2(r) \cdot \nabla \phi_2(r) + \sum_{r,2}(r) \phi_2(r) = \sum_{s,1-2}(r) \phi_1(r) + \chi_2 S_f(r) + Q_2(r)$$

$$-\nabla D_g(r) \cdot \nabla \phi_g(r) + \sum_{r,g}(r) \phi_g(r) = \sum_{s,g-1-2}(r) \phi_g(r) + \chi_g S_f(r) + Q_g(r)$$

$$-\nabla D_G(r) \cdot \nabla \phi_G(r) = \sum_{r,G}(r) \phi_G(r) = \sum_{s,G-1-2}(r) \phi_G(r) + \chi_G S_f(r) + Q_G.$$
where

$$\Sigma_{r,g} = \eta_{r,g} - \Sigma_{s,g \rightarrow g},$$

$$= \Sigma_{a,g} + \Sigma_{s,g \rightarrow g+1}.$$  \hspace{1cm} (5)

The fission source (which is spatial, but not energy group, dependent) is defined as

$$S_f(r) = \sum_{g'=1}^{G} \Sigma_{f,g'}(r) \phi_{g'}(r).$$ \hspace{1cm} (6)

The iteration is initiated by, first, guessing the initial flux distributions for all groups,

$$\phi_g(r) = \phi_g^{(n=0)}(r),$$ \hspace{1cm} (7)

where \(n\) denotes the source iteration index. Using the guessed flux, the initial fission source term can be calculated.

$$S_f^{(0)}(r) = \sum_{g'=1}^{G} \Sigma_{f,g'}(r) \phi_{g'}^{(0)}(r).$$ \hspace{1cm} (8)

If it is assumed that one has finished the \((n-1)\)-st outer iteration, then the neutron flux distributions in each group for the \(n\)-th iteration are sequentially computed beginning in the top (highest energy) group, \(g=1\).

$$- \nabla D_1(r) \cdot \nabla \phi_1^{(n)}(r) + \Sigma_{r,1}(r) \phi_1^{(n)}(r) = \chi_1 S_f^{(n-1)}(r) + Q_1(r).$$ \hspace{1cm} (9)

It should be noted that for \(g=1\), no scattering term, but only fission source and external neutron source terms appear in the right hand side of Eq.(9).

The next lowest energy group, i.e., \(g=2\), follows after the first group neutron flux is obtained,

$$- \nabla D_2(r) \cdot \nabla \phi_2^{(n)}(r) + \Sigma_{r,2}(r) \phi_2^{(n)}(r) = \Sigma_{s,1 \rightarrow 2}(r) \phi_1^{(n)}(r) + \chi_2 S_f^{(n-1)}(r) + Q_2(r).$$ \hspace{1cm} (10)

In calculating the scattering term for \(g=2\), the latest \((n\)-th\) first group neutron flux is used. This procedure can be continued to determine all of the group fluxes:

$$\phi_1^{(n)}(r), \phi_2^{(n)}(r), \ldots, \phi_G^{(n)}(r).$$ \hspace{1cm} (11)
After the new fluxes in all group have been calculated, a new fission source distribution
\[ S_f^{(n)}(r) = \sum_{g=1}^{G} \Sigma_{f,g}(r)\phi_g^{(n)}(r). \] (12)
is obtained. Although it is not needed in the iteration, the effective multiplication factor can be derived from the neutron balance in the reactor,
\[ k_{\text{eff}}^{(n)} = \frac{\text{Production}^{(n)}}{\text{Absorption}^{(n)} + \text{Leakage}^{(n)}}. \] (13)

Neutron production, absorption and leakage at the \( n \)-th source iteration can be written as
\[ \text{Production}^{(n)} = \int \sum_{g=1}^{G} \Sigma_{f,g}(r)\phi_g^{(n)}(r) dV, \] (14)
\[ \text{Absorption}^{(n)} = \int \sum_{g=1}^{G} \Sigma_{a,g}(r)\phi_g^{(n)}(r) dV, \] (15)
and
\[ \text{Leakage}^{(n)} = \int \sum_{g=1}^{G} D_g(r) \cdot \nabla \phi_g^{(n)}(r) dA. \] (16)

It should be noted that this \( k_{\text{eff}} \) value is different from the eigenvalue of the system considered without external source.

The source iterations proceed for \( n=1,2,3,...,N \). Here, \( N \) is the maximum number of source iteration. The iteration process may stop before reaching \( N \) if the group neutron flux converges, i.e. the following condition is fulfilled
\[ \max(g,r) \left| 1 - \frac{\phi_g^{(n)}(r)}{\phi_g^{(n-1)}(r)} \right| < \epsilon_\phi. \] (17)

**Finite Difference Equation**

In the source iteration method discussed earlier, for a particular \( n \)-iteration, in order to obtain group neutron flux, the \( g \)-th neutron diffusion equation has to be solved.
\[- \nabla D_g(r) \cdot \nabla \phi_g^{(n)}(r) + \Sigma_r(g)\phi_g^{(n)}(r) = \Sigma_{a,g}(r)\phi_g^{(n)}(r) + \Sigma_{x,g}(r)S_f^{(n-1)}(r) + Q_g(r). \] (18)
It can be observed that the terms on the right hand side of the above
equation are not dependent on the $n$-th iteration, $g$-th group neutron flux,
therefore, have constant values.

$$-\nabla D_g (r) \cdot \nabla \phi_g^{(n)} (r) + \Sigma_{r,g} (r) \phi_g^{(n)} (r) = S_g^{(n)} (r),$$

where

$$S_g^{(n)} (r) = \Sigma_{s,g-1 \rightarrow g} (r) \phi_{g-1}^{(n)} (r) + \chi_g S_f^{(n-1)} (r) + Q_g (r).$$

The right hand side of the above equation (known value) represents the
source terms from down-scattering, fission, and external neutron source,
respectively.

If an iterative method is applied to solve the above equation, such as
when the reactor geometry is two or three dimensional (Batan-2DIFF or 
-3DIFF code), then the iterative process is commonly called as inner
iterations while the source iteration discussed previously is commonly called
as outer iterations. At this point, approximation of spatial variable has to be
considered. The finite difference method (FDM) has been chosen for
Batan-2DIFF and -3DIFF codes since FDM is relatively simpler to be
programmed and the reactor geometry commonly faced in practical reactor
design can be usually treated quite satisfactorily in $X-Y$, $R-Z$ or $X-Y-Z$
geometry.

The spatial difference equations are obtained by integrating Eq. (19)
over the volume associated with each mesh point. The placing of the mesh
point is in the interior of the mesh. The Laplacian term is discretized with
five and seven point finite difference schemes for 2-D and 3-D cases,
respectively. Full derivations are not given here since they are too lengthy.
However, readers may find the detail derivation of the finite difference
equations for example in Refs.(4,5). The finite difference formulation of
neutron diffusion equation, Eq.(19), takes the final form of a system of
linear equations,

$$A_g \Psi_g = s_g,$$

$$g = 1,2,\ldots,G.$$

Here, $A$ is a symmetric, positive definite, block tridiagonal and block
pentadiagonal matrices for 2-D and 3-D cases, respectively, and the other
two notations are vectors. Since the dimensions of $A$ are usually large for
practical problems of reactor design, the successive over-relaxation (SOR)
method is adopted in Batan-2DIFF and -3DIFF codes to solve Eq. (21).
In the SOR method, the optimum acceleration parameter ($\omega$) will minimize the number of iterations in the SOR method. The optimum $\omega$ is in the low side of $1 < \omega < 2$ if the number of meshes is very small, but it approaches 2 if the number of meshes is increased. In reactor design practices, the number of meshes is usually large and the convergence rates of SOR method become slower and more sensitive to the value of $\omega$ selected. The calculation of the optimum $\omega$ which is usually a few hundred iteration cycles is strongly recommended. The optimum value of $\omega$ for a given problem can be found easily by a pilot run of SOR with a lower estimate of $\omega$. Then through ratio of the obtained group flux and the previous value the optimum $\omega$ is estimated [8]. If the same equation (coefficient matrix A) is solved repeatedly as in Eq.(21) with different source terms, the optimum $\omega$ found for the first problem remain the same for all other problems. This property is very advantageous for solving the fixed source problem of multigroup neutron diffusion. An option to invoke the optimum acceleration searching is available in the code.

CODES VERIFICATION

Subprograms for solving the inhomogeneous problem, i.e. the fixed source problem of multigroup neutron diffusion have been developed (written in FORTRAN-77 language) and integrated into the Batan's standard neutron diffusion codes, Batan-2DIFF and -3DIFF. These codes can be executed on both DEC VAX 8550 and DEC Alpha Work Station AXP-2100. In particular, the Batan-2DIFF code can also be executed in a personal computer with slower computation time and smaller problem sizes. The modularization of the Batan-2DIFF and -3DIFF codes turned out to be very useful for installing the new subprograms. The subprograms, themselves, have been extensively debugged and verified to exclude any possible error. Errors mostly occurred during the programming phase. The most effective way to check the validity of a code is by comparing the calculations results with another standard code. In the following, the verification results using a generic diffusion code 2DBUM are presented for two fixed source sample problems of neutron diffusion in a 2-D X-Y reactor geometry.

The reactor configuration and dimension for Sample Problem 1 is shown in Fig. 1. The reactor with X-Y geometry consists of two regions, i.e., core (consists of MTR-type silicide fuel elements) and reflector (light water) regions. The core is exactly in the center of the reactor and the external neutron source is homogeneously distributed over the reactor. This neutron source is not group-dependent and has a unity strength (1 neutron/cc.s). The same four group diffusion parameters for core and reflector regions with
the one listed in Refs. (2) and (3) were prepared with WIMS/D4 cell calculation code [6] (Table 1). All four sides have vacuum boundary conditions. With 50 meshes for both x- and y-directions, altogether 2500 meshes or control volumes were involved in the calculation. Therefore, the dimensions of the coefficient matrix and the group neutron flux vector are 2500 x 2500 and 2500 per group, respectively, which is categorized as a typical medium scale problem.

The fixed source problem is physically meaningful only if the effective multiplication factor, \( k_{\text{eff}} \), of the system is lower than unity (subcritical). For this sample problem the \( k_{\text{eff}} \) was 0.879455 (2DBUM calculation result). The calculation results of Batan-2DIFF and 2DBUM codes are compared in Table 2. The convergence criteria imposed on the group neutron fluxes were \( 10^{-5} \). It can be observed that the two codes produced almost identical results. The relative differences of the group neutron flux and neutron balance were in the order of \( 10^{-4} \), while the maximum power peaking factor and effective multiplication factor were in the order of \( 10^{4} \). Fig. 2 shows the group neutron flux distribution at the center of the reactor (in x-direction). It can be observed that the two codes produced identical profiles.

In Sample Problem 2, the neutron source are located only in the upper left quadrant of the core region (Fig. 3) so that the neutron flux distribution will be unsymmetrical against x- or y-axis, in contrast to the symmetric Sample Problem 1. This neutron source intensity is identical with the one of the Sample Problem 1, i.e. (1 neutron/cc.s). All four sides have vacuum boundary conditions. Group constant set, mesh definition and convergence criteria are also identical with Sample Problem 1, so that the effective multiplication factor was also 0.879455. The calculation results of Batan-2DIFF and 2DBUM codes are shown in Table 3. Similar to the results of Sample Problem 1, the relative differences of the group neutron flux and neutron balance were in the order of \( 10^{-4} \), while the maximum power peaking factor and effective multiplication factor were in the order of \( 10^{4} \). Fig. 4 shows the group neutron flux distribution at the center of the reactor (in y-direction). Again the two codes produced almost identical profiles.

Some remarks on using the 2DBUM code for solving the fixed source problem are given. The first limitation imposed on the code is that the system must contain fissile material (the effective multiplication factor must be larger than zero). This limitation is relaxed in the Batan-2DIFF and -3DIFF codes. Problem related to 2DBUM convergence appeared when the neutron source is concentrated and located in the non-fissile region. For example, if the neutron source in Sample Problem 2 is moved to the upper left of the reactor in the reflector region then 2DBUM will not converge. In general, the convergence rate of 2DBUM for fixed source
problems are worse than the one for the eigenvalue problem. There is no convergence problem in case of Batan-2DIFF and -3DIFF codes.

CONCLUSION

Subprograms for solving the fixed source problem of the multigroup neutron diffusion in 2-D and 3-D reactor geometries have been successfully developed and integrated into Batan's standard multidimensional, multigroup neutron diffusion codes, Batan-2DIFF and -DIFF. Verification of the results with other generic diffusion codes showed that the relative differences for the group neutron fluxes and neutron balances were in the order of $10^{-4}$, while the maximum power peaking factors were in the order of $10^{-5}$ which proved the validity of the codes.

ACKNOWLEDGMENTS

The author expresses his special gratitude to Ir. Tagor M. Sembiring for providing the macroscopic group constant with WIMSD/4 cell calculation code and for his advise on using the 2DBUM diffusion code. Keen interest and constant encouragement given by Ir. Hilman Ramli, Ir. Tagor M. Sembiring, Ir. Bakrie Arbie, Dr. Ir. As Natio Lasman, Ir. Iman Kunto, and all other staffs of the Reactor Physics Division and Critical Assembly Installation, Center for Multipurpose Reactor, are very helpful in the course of completing the work.

REFERENCES


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9. T.M. SEMBIRING, and B. ARBIE, private communications
Table 1. Macroscopic cross sections for code verification.

<table>
<thead>
<tr>
<th>g</th>
<th>$\Sigma_{f,g}$</th>
<th>$\nu\Sigma_{f,g}$</th>
<th>$\Sigma_{s,g\rightarrow g+1}$</th>
<th>$\Sigma_{s,g\rightarrow g+2}$</th>
<th>$\Sigma_{a,g}$</th>
<th>$D_g$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core (Silicide MTR fuel)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>1.7123E-3</td>
<td>6.1896E-4</td>
<td>7.2244E-2</td>
<td>4.0177E-4</td>
<td>1.0747E-3</td>
<td>2.4904</td>
</tr>
<tr>
<td>3</td>
<td>1.2629E-2</td>
<td>5.1879E-3</td>
<td>8.0215E-2</td>
<td>0.0</td>
<td>1.4650E-2</td>
<td>0.8019</td>
</tr>
<tr>
<td>4</td>
<td>1.7972E-1</td>
<td>7.3945E-2</td>
<td>0.0</td>
<td>0.0</td>
<td>1.0157E-1</td>
<td>0.2781</td>
</tr>
<tr>
<td>Reflector (light water)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>0.0</td>
<td>0.0</td>
<td>1.0465E-1</td>
<td>6.5165E-4</td>
<td>5.2009E-4</td>
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</tr>
<tr>
<td>2</td>
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<td>0.0</td>
<td>1.4625E-1</td>
<td>1.4299E-5</td>
<td>2.1743E-7</td>
<td>0.7878</td>
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<tr>
<td>3</td>
<td>0.0</td>
<td>0.0</td>
<td>1.4590E-1</td>
<td>0.0</td>
<td>9.7625E-4</td>
<td>0.5633</td>
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<tr>
<td>4</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>1.8944E-2</td>
<td>0.1531</td>
</tr>
</tbody>
</table>

$\chi_1 = 0.7452496$, $\chi_2 = 0.2545513$, $\chi_3 = 2.006703E-4$, $\chi_4 = 0.0$

*read as $1.7123 \times 10^{-3}$

Table 2. Verification Results for Sample Problem 1.

<table>
<thead>
<tr>
<th>Calculated parameter</th>
<th>Batan-2DIFF</th>
<th>2DBUM</th>
<th>Relative difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flux at the center of the reactor (10^2/cm.s):</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1-st group</td>
<td>4.77495</td>
<td>4.77674</td>
<td>$3.7 \times 10^{-4}$</td>
</tr>
<tr>
<td>2-nd group</td>
<td>5.17691</td>
<td>5.17866</td>
<td>$3.8 \times 10^{-4}$</td>
</tr>
<tr>
<td>3-rd group</td>
<td>4.32706</td>
<td>4.32867</td>
<td>$3.7 \times 10^{-4}$</td>
</tr>
<tr>
<td>4-th group</td>
<td>3.42458</td>
<td>3.42582</td>
<td>$3.6 \times 10^{-4}$</td>
</tr>
<tr>
<td>Core power peaking factor (maximum)</td>
<td>1.5492</td>
<td>1.5491</td>
<td>$6.4 \times 10^{-5}$</td>
</tr>
<tr>
<td>Reactor neutron balance (10^4/s):</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total production</td>
<td>2.7378</td>
<td>2.7388</td>
<td>$3.7 \times 10^{-4}$</td>
</tr>
<tr>
<td>Total absorption</td>
<td>6.1989</td>
<td>6.2000</td>
<td>$1.8 \times 10^{-4}$</td>
</tr>
<tr>
<td>Total leakage</td>
<td>0.5388</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>$k_{eff}$(eigenvalue problem)</td>
<td>0.879471</td>
<td>0.879455</td>
<td>$1.8 \times 10^{-5}$</td>
</tr>
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Table 3. Verification results for Sample Problem 2.

<table>
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<tr>
<th>Calculated parameter</th>
<th>Batan-2DIFF</th>
<th>2DBUM</th>
<th>Relative difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flux at the center of the reactor ((10^2)/cm.s):</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>1-st group</td>
<td>7.72918</td>
<td>7.73192</td>
<td>3.5 x 10^{-4}</td>
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<tr>
<td>2-nd group</td>
<td>8.50224</td>
<td>8.50518</td>
<td>3.5 x 10^{-4}</td>
</tr>
<tr>
<td>3-rd group</td>
<td>7.18358</td>
<td>7.18602</td>
<td>3.4 x 10^{-4}</td>
</tr>
<tr>
<td>4-th group</td>
<td>5.99782</td>
<td>5.99970</td>
<td>3.1 x 10^{-4}</td>
</tr>
<tr>
<td>Core power peaking factor (maximum)</td>
<td>1.8544</td>
<td>1.8545</td>
<td>5.4 x 10^{-5}</td>
</tr>
<tr>
<td>Reactor neutron balance ((10^3)/s):</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total production</td>
<td>4.0677</td>
<td>4.0692</td>
<td>3.7 x 10^{-4}</td>
</tr>
<tr>
<td>Total absorption</td>
<td>4.4660</td>
<td>4.4677</td>
<td>3.8 x 10^{-4}</td>
</tr>
<tr>
<td>Total leakage</td>
<td>0.001474</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>(k_{eff}) (eigenvalue problem)</td>
<td>0.879471</td>
<td>0.87945</td>
<td>1.8 x 10^{-5}</td>
</tr>
</tbody>
</table>
Figure 1. Core-reflector configuration for Sample Problem 1 (the neutron source is distributed uniformly over the reactor).

Figure 2. The y-directional group neutron flux distributions at the center of the reactor calculated with Batan-2DIFF code (solid lines) and with 2DBUM code (dots) for Sample Problem 1.
Figure 3. Core-reflector configuration for Sample Problem 2 (the neutron source is located in the upper left quadrant of the core).

Figure 4. The y-directional group neutron flux distributions at the center of the reactor calculated with Batan-2DIFF code (solid lines) and with 2DBUM code (dots) for Sample Problem 2.