The Convenient Monte Carlo Code MULTI-KENO for Criticality Safety Analysis of Transport Casks

K.OSADA[1], Y.NAITO[2] and H.OKUNO[2]

[1]Kubota Corporation, Osaka, Japan

[2]Japan Atomic Energy Research Institute, Tokai, Ibaraki, Japan

Introduction

In criticality safety analysis of transport casks, an infinite array of fuel rods using a homogeneous cell is normally assumed, since the calculation configurations of transport casks are, in many cases, complicated. Criticality calculations are performed with the homogeneous cell. In the case of a transport cask, using the homogeneous cell is not adequate, since there are many fuel rods contacting reflectors in comparison with the total number of fuel rods. We developed a Monte Carlo code MULTI-KENO (Naito et al. 1983) at Japan Atomic Energy Research Institute (JAERI) in 1983. MULTI-KENO was developed to express a configuration exactly without performing the cell calculation by introducing the concept of 'SUPER BOX'. In this paper, we show some features of MULTI-KENO and an example of calculations by MULTI-KENO.

Features of MULTI-KENO

MULTI-KENO was developed at JAERI in 1983 as an improvement to KENO-IV (Petrie and Cross 1975) developed at Oak Ridge National Laboratory (ORNL). MULTI-KENO is now included in the criticality safety evaluation code system JACS (Katakura et al. 1982) developed at JAERI. MULTI-KENO has the following functions.

SUPER BOX

A system can be divided into any number of sub-systems named SUPER BOX. A super box can contain some boxes named BOX TYPE. A box is a rectangular lattice, which is also used in KENO-IV. By introducing the super box concept, the size of boxes in each super box can be selected independently. By means of several super boxes, we can specify the calculation configuration easily. Examples of configurations of MULTI-KENO are shown in Fig.1 and Fig.2. The system shown in Fig.1 is divided into two super boxes. The upper super box consists of four boxes. The lower one consists of twenty boxes. The size of the box contained in the upper super box can be determined independently of that of the box in the lower one. An array of super boxes is enclosed by a 'CORE BOUNDARY' (CORE BDY). The system shown in Fig.2 is also divided into two super boxes. The upper super box consists of eight boxes. The lower one consists of four boxes. An array of boxes is enclosed by a 'CELL BOUNDARY' (CELL BDY). The cell boundary of each super box has a double cylinder enclosure, and each outer cylinder is enclosed by a parallelepiped. A cell boundary defines an outline of an array of boxes, a core boundary defines an outline of an array of super boxes.

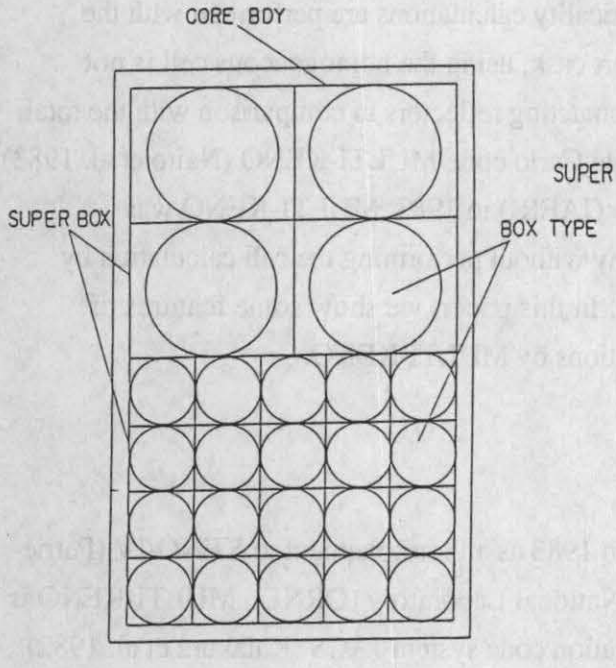

Fig.1 Example of SUPER BOX of MULTI-KENO (1).

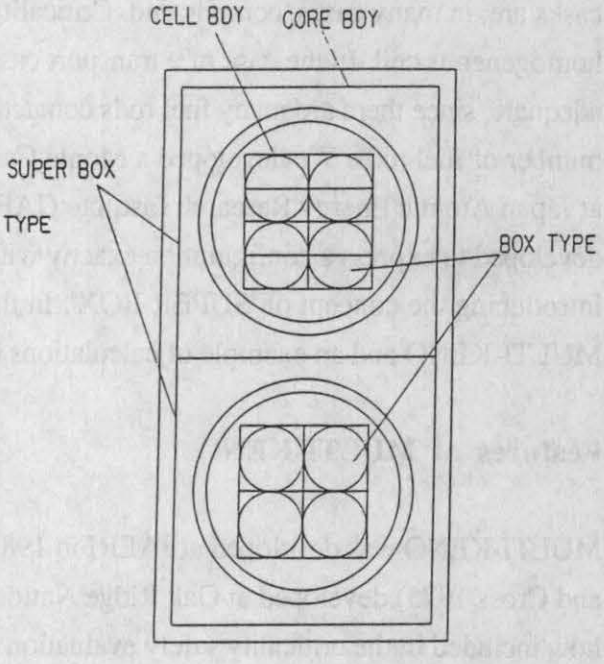

Fig.2 Example of SUPER BOX of MULTI-KENO (2).

Intersecting Geometry

In calculation of transport casks, A CORE BOUNDARY (CORE BDY) frequently intersects other outer boundaries, as shown in Fig.3, and calculations are stopped due to error when a Monte Carlo code is used. With MULTI-KENO, intersecting geometries are permitted, i.e. it allows intersection of the CORE BOUNDARY and other outer boundaries. An example of a transport cask configuration is shown in Fig.3. This model consists of eleven super boxes and the corners of four super boxes intersect outer boundaries. By introducing this function, therefore, the configurations for transport casks can be easily expressed, since it is not necessary to take the approximations of the configurations.

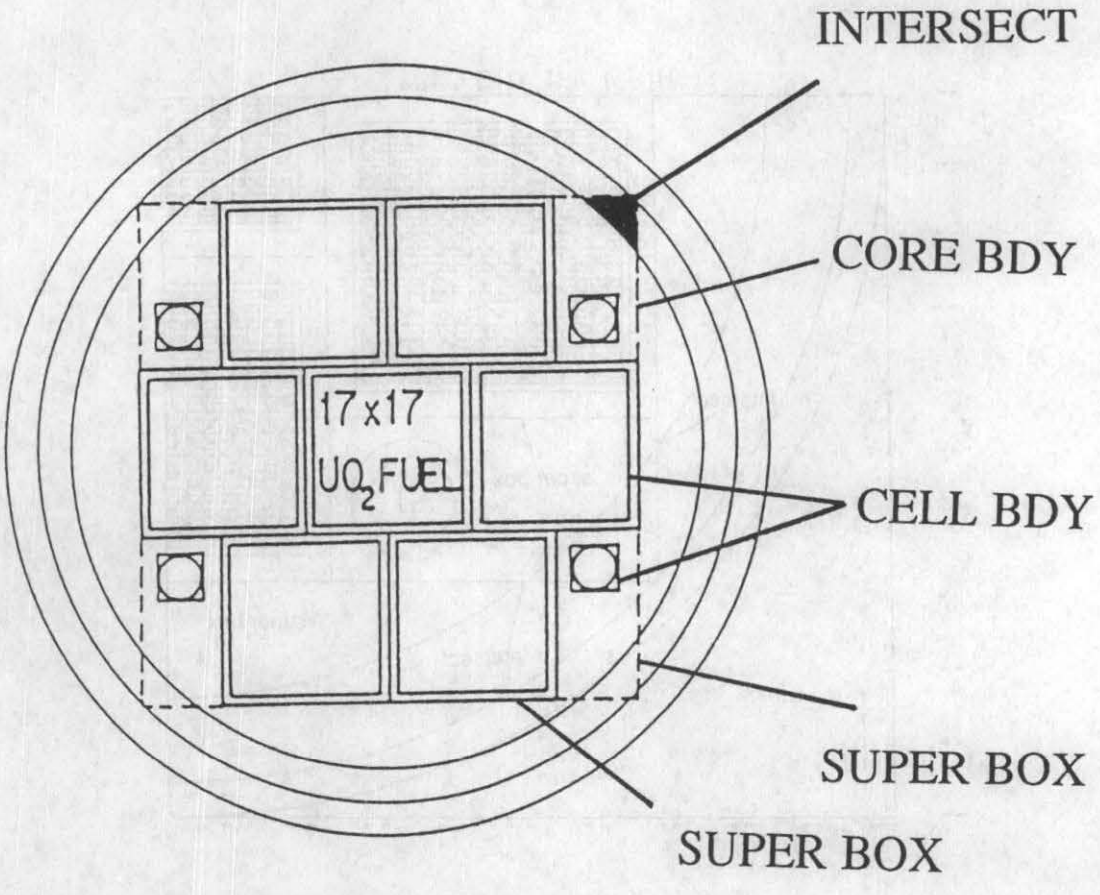

Fig.3 Transport cask configuration.

Graphic Processing

Graphic processing is very useful in checking geometrical input data. It enables us to draw an arbitrary cross sectional figure of a system defined by geometrical input data. An example output is shown in Fig.4.

Hexagonal Array System

In criticality calculation for transport casks, it is sometimes necessary to calculate for the system of a hexagonal array of fuel rods, and we made it possible to express a configuration of a hexagonal array of fuel rods.

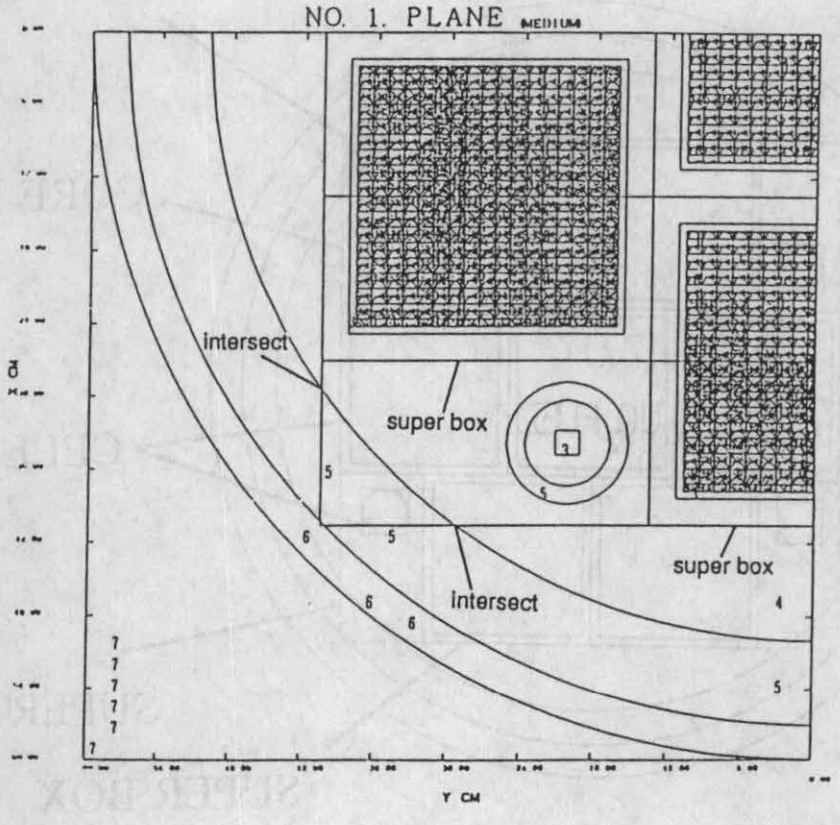

Fig.4 Arbitrary cross sectional figure by graphic processing of MULTI-KENO.

Calculation Method

MULTI-KENO is now included in the criticality safety evaluation code system JACS developed at JAERI. The calculation flow diagram of JACS is shown in Fig.5. JACS has the following library and codes.

MGCL library
MAIL code
ANISN-JR code
REMAIL code
KENO-IV code
MULTI-KENO code

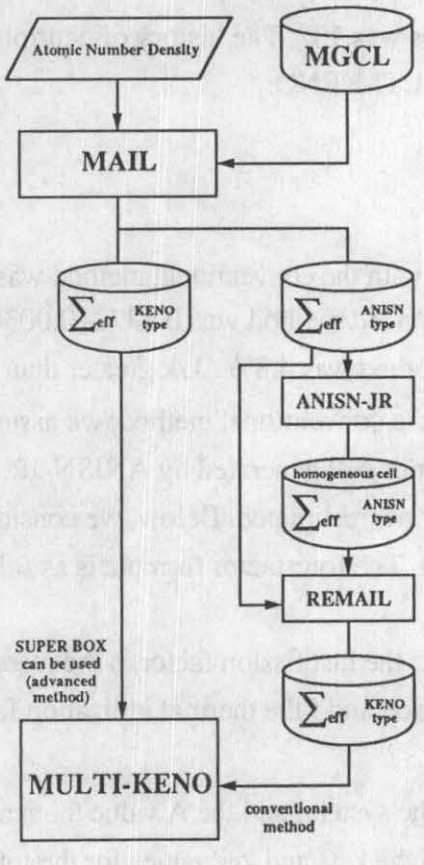

Fig.5 Calculation flow diagram of JACS.

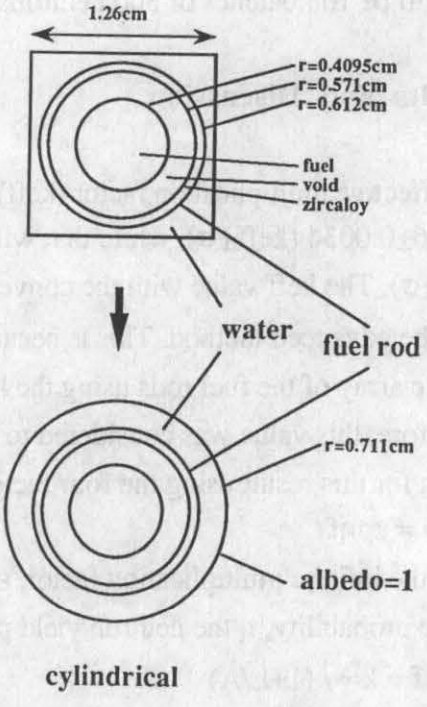

Fig.6 Calculation model for ANISN-JR to generate a homogeneoue cell.

MGCL (Naito et al. 1981) is a multigroup constant library generated from an evaluated nuclear data file ENDF/B-IV (ENDF-201 1975) or JENDL (Shibata et al. 1990). MAIL (Komuro et al. 1990) is a code which generates effective macroscopic cross sections from MGCL and atomic number densities. The Sn code ANISN-JR (Koyama et al. 1977) is the JAERI version of ANISN (Engle 1967). REMAIL (Naito et al. 1981) is used to change ANISN-type macroscopic cross sections to KENO-type ones, and vice versa.
In this report, we show an example of criticality calculations for the transport cask applying two uses of MULTI-KENO. The calculation configuration is shown in Fig.4, and the calculation flow diagram is shown in Fig.5. The first calculation was performed with the conventional method, using the MAIL, ANISN-JR, REMAIL and MULTI-KENO codes. Figure 6 shows the calculation model for ANISN-JR. The regions of the fuel assemblies were homogenized by this code. REMAIL was used to change the ANISN-type effective macroscopic cross sections to KENO-type ones. The second calculation was performed by the SUPER BOX function of MULTI-KENO. This is a more advanced method. In these calculations, the number of neutron energy groups was 137. The history of neutrons were taken to be 103 batches of 300 neutrons with MULTI-KENO.

Results and Discussion

The effective multiplication factor (keff) obtained with the conventional method was 0.8776 ± 0.0034 ($keff\pm\sigma$), while that with the advanced method was 0.8378 ± 0.0038 ($keff\pm\sigma$). The keff value with the conventional method was 4.8% $\Delta k/k$ greater than that with the advanced method. This is because, with the conventional method, we assumed the infinite array of the fuel rods using the homogeneous cell generated by ANISN-JR. Therefore, this value was considered to be greatly overestimated. Below, we consider the reason for this result using the four factor formula. The four factor formula is as follows:

$$k_\infty = \varepsilon p \eta f$$

k_∞ is the infinite multiplication factor; ε represents the fast fission factor, p the resonance escape probability, η the neutron yield per absorption and f the thermal utilization factor.

$$keff = k_\infty / (1+L/A)$$

The L value represents the neutron leakage from the system, and the A value the neutron absorption in the system. This time, we calculated the keff and k_∞ values for the total system. The results of the calculation are as follows:

keff = $k\infty$ / (1+L/A) = 0.8378 (the advanced method)

$k\infty = \varepsilon p \eta f = 0.8637$

1+L/A = 1.0309

keff = $k\infty$ / (1+L/A) = 0.8776 (the conventional method)

$k\infty = \varepsilon p \eta f = 0.9002$

1+L/A = 1.0258

The ratios of the keff and $k\infty$ values between the conventional method and the advanced method are as follows:

$keff_{(conventional)} / keff_{(advanced)} = 0.8776 / 0.8378 = 1.048$

$k\infty_{(conventional)} / k\infty_{(advanced)} = 0.9002 / 0.8637 = 1.042$

$1+L/A_{(conventional)} = 1+L/A_{(advanced)} = 1.0258 / 1.0309 = 0.995$

With the conventional method, the keff and $k\infty$ values are higher than those with the advanced method. We calculated the following ratios to examine this result.

$\varepsilon_{(conventional)} / \varepsilon_{(advanced)} = 0.995$

$p_{(conventional)} / p_{(advanced)} = 1.008$

$\eta f_{(conventional)} / \eta f_{(advanced)} = 1.039$

The values of ε, p and ηf were obtained by the following equations.

$\varepsilon = F_{total} / F_{thermal}$; $p = A_{thermal} / A_{total}$; $\eta f = F_{thermal} / A_{thermal}$

The F_{total} value represents the fission in the total neutron energy, the $F_{thermal}$ value the fission in the thermal neutron energy; the A_{total} value represents the neutron absorption in the total neutron energy, the $A_{thermal}$ value the neutron absorption in the thermal neutron energy.

As a result, the ηf value is the strong factor attributable to the increments of the keff and $k\infty$ values with the conventional method. This is because, with the conventional method, we assumed the infinite array of fuel rods with the homogeneous cell. The neutron spectrum of the homogeneous cell was different to that of fuel rods contacting reflectors. In the case of a transport cask, there are not many fuel rods, so it is not adequate to perform the cell calculation. On the other hand, using the super box function of MULTI-KENO, we can express each fuel rod even though the calculation configuration is complicated.

References

W.W.Engle, Jr., "A Users Manual for ANISN; A One Dimensional Discrete Ordinates Transport Code with Anisotoropic Scattering", K-1693 (1967).

J.Katakura, Y.Naito and Y.Komuro, "Development of the Computer Code System JACS for Criticality Safety", Transaction of ANS41 (1982).

Y.Komuro et al., "MAIL3.0: A Computer Program Calculating Cross Section Sets for SIMCRI, ANISN, KENO-IV, MULTI-KENO and MULTI-KENO-II", JAERI-M90-126 (1990).

K.Koyama et al., "ANISN-JR, A One-Dimensional Discrete Ordinates Code for Neutron and Gamma-Ray Transport Calculations", JAERI-M6954 (1977).

Y.Naito et al., "MGCL-PROCESSOR: A Computer Code System for Processing Multi-Group Constant Library MGCL", JAERI-M9396 (1981).

Y.Naito et al., "MULTI-KENO: A Monte Carlo Code for Criticality Safety Analysis", JAERI-M83-049 (1983).

L.M.Petrie and N.F.Cross, "KENO-IV, An Improved Monte Carlo Criticality Program", ORNL-4938 (1975).

K.Shibata et al., "Japanese Evaluated Nuclear Data Library, Version3-JENDL3", JAERI 1319 (1990).

ENDF/B Summary Documentation, BNL-NCS-17541 (ENDF-201), 2nd Edition (ENDF/B-IV) (1975).

REGULATIONS, SYSTEM ANALYSIS,
ENVIRONMENTAL IMPACTS, AND RISK ASSESSMENT

Session 6:

RADIATION PROTECTION-II

Chairman : K. Ueki
Co-Chairman : G. E. Whitesides
Coordinator : N. Watabe