CONF-8410170--3

DE85 002577

THE KENO-V CODE

S. N. Cramer
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831, USA

DISCLAIMER

KENO-V

1. Introduction

The KENO-V[1] code is the current release of the Oak Ridge multigroup Monte Carlo criticality code development. The original KENO,[2] with 16 group Hansen-Roach[3] cross sections and P_1 scattering, was one of the first multigroup Monte Carlo codes and it and its successors have always been a much-used research tool for criticality studies. KENO-V is able to accept large neutron cross section libraries (a 218 group set is distributed with the code) and has a general P_N scattering capability. A supergroup feature allows execution of large problems on small computers, but at the expense of increased calculation time and system input/output operations. This supergroup feature is activated automatically by the code in a manner which utilizes as much computer memory as is available.

The primary purpose of KENO-V is to calculate the system k_{eff}, from small bare critical assemblies to large reflected arrays of differing fissile and moderator elements. In this respect KENO-V neither has nor requires the many options and sophisticated biasing techniques of general Monte Carlo codes. All effort in terms of computer memory, geometry, cross sections, input/output, etc., is directed toward the most efficient calculation of k_{eff}, which will usually be from 5 to 10 times faster than a similar calculation with a general code. Among other quantities calculated (most optional) are lifetime and generation time and spatial and energy dependent fluxes, fission density, absorption, and leakage. In addition to cross sections, the code is distributed with sets of biasing weights and albedo data, either of which may be used to reduce computation time in a reflector.

The KENO-V input data is simple and flexible and has been designed for use with

remote terminals. Much of the data has default values, which are adequate for many problems, and the restart capability allows subsequent calculations to be made with only a minimum of input changes. KENO-V is essentially a completely new code from its immediate predecessor, KENO-IV,[4] but has retained, or improved upon, most of the features of the older code. The input data is different and, even without supergrouping, KENO-V is able to handle much larger problems than KENO-IV. The original KENO body-type array geometry is retained in KENO-V, but the next version (KENO-V.a, soon to be released) contains a more flexible geometry with an "array of arrays" feature similar to that in MARS geometry.[5]

The KENO-V code is distributed by the Radiation Shielding Information Center (RSIC) at Oak Ridge and, through RSIC, by the National Energy Software Center (Argonne Code Center) and the N.E.A. Data Bank at Saclay in France. KENO-V is one of the functional modules of the SCALE-2 code system[6] created at Oak Ridge for the U.S. Nuclear Regulatory Commission commercial reactor licensing program. This code system and its rather formidable volumes of documentation contain many former stand-alone codes and special auxiliary routines for performing a variety of tasks related to cross section treatment and transport calculations. SCALE-2 is distributed only for IBM computers, and it also contains a KENO-IV module. An earlier release, SCALE-0, has a CDC computer version with only KENO-IV. It is within a sub-system of SCALE-2, CSAS4, that the automated "search" feature of KENO-V is retained. Here, through a series of multiple KENO-V and other module calculations, geometric dimensions are adjusted to determine a specific k_{eff}, the final output being the adjusted dimensions. Also, optimal (maximum or minimum) k_{eff}'s may be determined by geometric adjustments within some specified restraints.

II. Cross Sections

The multigroup cross sections to be used in KENO-V must be in the "AMPX working library" format.[7] The user must either have such data available or have the capability to create it. The SCALE-2 code package from RSIC not only provides this capability, but it

also supplies several basic problem-independent cross section libraries. Only the code package sample problem cross section library (HRXSCS, file 78) does not require problem dependent processing, however, it is in BCD card image form. The other libraries (files 82-89) are also in BCD card image form and must first be converted to binary format with AIM (file 33) before beginning any problem dependent processing. The four cross section files applicable to KENO-V are given in Table I. More information concerning these libraries is given in Section C1.A.1 in SCALE-2 (module CSAS1).

Table I. KENO-V Cross Section Libraries

File No.	Name	Description
82	SCALE.X16	16 groups Hansen-Roach data
83	SCALE.X27	27 group collapsed sub-set of the 218 group data
84	SCALE.X123	123 group data from the GAM and THERMOS codes
88	SCALE.X218	218 group data from ENDF/B IV data processed by the XLACS code

KENO-V may be run independently or as part of the SCALE calculation sequence. If it is run independently, the cross section processing may be done with either of the SCALE modules NITAWL or ICE (for the Hansen-Roach data the BONAMI module is needed). Cross section mixing input (elements, isotopes, densities, weight fractions, etc.) must be supplied to KENO-V when NITAWL is used but not with ICE, which requires its own mixing input data.

If KENO-V is run as part of SCALE, the same basic operations are performed, but in a more automated manner. The mixing input is the same as that for the CSAS4 (similar to that of CSAS1 and CSAS2) module in Sections C4. The KENO-V calculation is performed in the CSAS25 module. This is the version 5 extension of CSAS2, which was created for KENO-IV. The CSAS25 operation is essentially the same as that in the CSAS4 (KENO-V criticality search) module with only one pass through the search procedure.

In addition to providing the various data collection, mixing, group collapsing, and

input/output format features, the cross section modules (SCALE or others) also provide essential problem dependent nuclear physics analysis. For example, in creating the problem independent group structures in Table I, the basic data (ENDF/B or other) may have been weighted above 60 keV with a 1.22 MeV peak fission spectrum, from 60 keV to 0.12 eV with a 1/E spectrum, and below 0.12 eV with a 300°K Maxwellian spectrum. For nuclides with resolved resonance energy parameters, NITAWL gives the data a temperature dependent Doppler broadening self-shielded treatment based on Nordheim integral resonance theory. For nuclides with Bondarenko factors (the Hansen-Roach data), the BONAMI code provides a similar treatment based on the narrow-resonance approximation. The XLACS code treats nuclides with unresolved resonance energy parameters on an individual basis, and the output is passed to NITAWL as smooth (non-resonance) data. Geometric lattice effects may be included in the NITAWL and BONAMI calculations by the use of Dancoff correction factors. The SCALE sequence also provides a one-dimensional discrete ordinates (XSDRNPM) calculation for cell-weighting cross sections, if needed.

The cross section treatments described here are at least as important as any other aspect of a multigroup Monte Carlo calculation. However, a full discussion of them could easily fill a full course such as we have here for Monte Carlo methods alone. The SCALE documentation provides ample discussion, theory, examples, sample problems, and references to satisfy most needs. Further information is provided in reference 8. The traditional use of Monte Carlo codes has been as the "method of last resort" due to geometric complexities inappropriate to one- and two-dimensional deterministic methods. These simpler geometric methods are also multigroup codes requiring the same general cross section treatments as described here, and it is assumed that some expertise in these areas is available.

III. Geometry

The principal element in the KENO-V geometry input is referred to as a UNIT (also BOX or BOX TYPE). An array or lattice is constructed by stacking these elements. The

outer boundary of a UNIT is a rectangular parallelepiped. Different UNITS may have different internal structure as well as different outer dimensions. However, it is necessary to describe a specific UNIT type only once, no matter how often it appears in the system to be calculated. The outer boundary of the array is also a rectangular parallelepiped, and the UNITS are stacked in the array as shown schematically in Fig. 1. This stacking may be done in any three-dimensional arrangement of rows and columns, with the following restrictions: (1) the units must be oriented along the coordinate axes of the array, (2) faces (including tops and bottoms) of adjacent UNITS must match exactly, i.e., one UNIT must touch only one other UNIT along any one face, and (3) the UNITS must fit snugly into the array with no overlap or empty spaces.

Each UNIT has its own internal coordinate system, but the alignment of all coordinate axes must be the same and match that of the array. The geometric bodies available for UNIT specifications are (1) rectangular parallelepiped (called a CUBOID), (2) a special CUBE, (3) sphere, (4) hemisphere, (5) cylinder, and (6) hemicylinder. Combinations of these bodies may be arranged in a general manner inside the UNIT, with the following restrictions: (1) Bodies must not overlap or intersect, i.e., each body must be completely contained within another body. Bodies can touch at points of tangency and share common faces, (2) CUBES, CUBOIDS, the flat side of hemispheres and hemicylinders, and cylindrical axes must be aligned along coordinate axes, i.e., bodies may not be rotated in space. Examples of valid and invalid body arrangements are shown in Fig. 2. It is not necessary for valid body configurations to be concentric about any particular axis, i.e., the interior bodies in Fig. 2 may all be off-centered so long as the above restrictions hold.

Geometry "regions" are defined by the volume of a body excluding any interior bodies. Regions are used to identify cross section mixtures, biasing (Russian roulette and splitting) parameters, and any spatial dependent results. Special body designations are used to identify core-reflector interfaces and reflector geometry. The code calculates and prints out the volume (in cm^3) for each region in the system. Two convenient options are available for locating UNITS inside the array. The KENO-V manual in SCALE has many

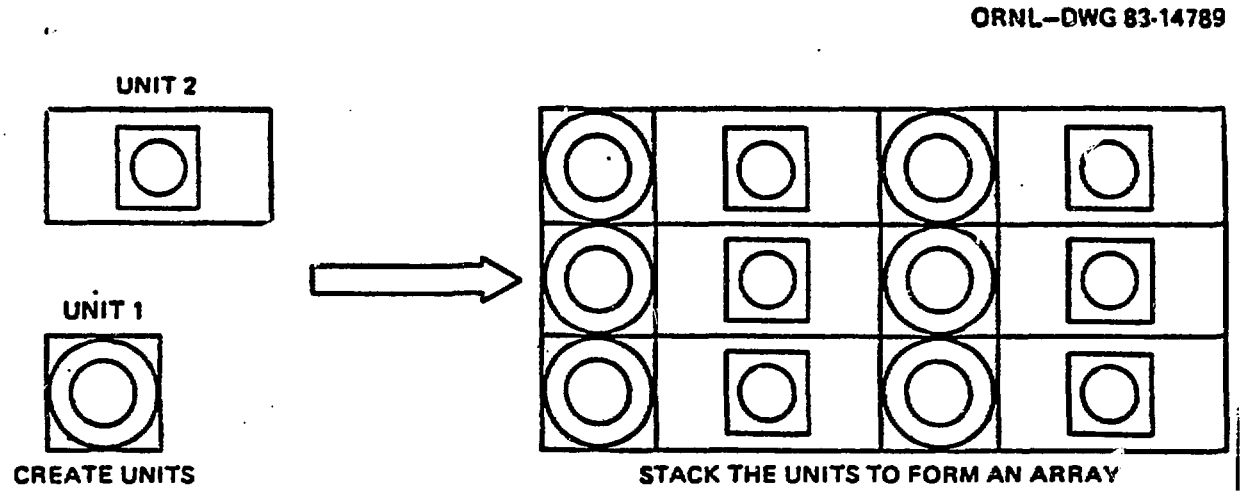
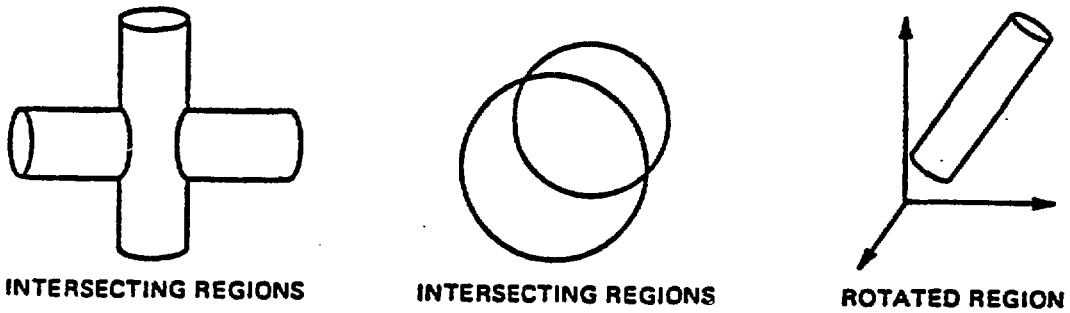
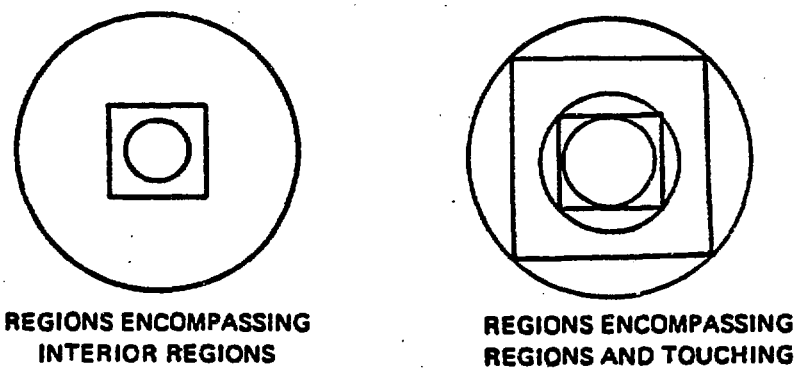


Fig. 1. Example of Array Construction.



INVALID BODY CONFIGURATIONS



VALID BODY CONFIGURATIONS

Fig. 2. KENO Geometry.

geometry input examples and sample problems.

The code will detect many geometry input errors either with warning messages and/or termination. However, it is possible the user will make errors acceptable to the code, resulting in a geometry different from that desired. There exists a code (JOMCHK from KENO-IV) which produces printer plots of two-dimensional cuts through the system (such as the array in Fig. 1), enabling the user to verify his geometry input data. This code is not a part of the SCALE-2 package, but in the next code version, KENO-V.a, this geometry plotting capability will be a standard feature. The newer code is essentially the same as KENO-V except for the geometry options, which will be greatly expanded. In addition to greater flexibility in UNIT definition, an "array of arrays" feature (similar to MARS in SCALE) will allow arrays to be treated as UNITS for further stacking in larger arrays. The RSIC Newsletter will announce when KENO-V.a is ready for distribution.

The KENO-V geometry capability is essentially the same as that for KENO-IV, but with a slightly different input description. However, KENO-V has no general or combinational geometry features found in some non-SCALE versions of KENO-IV.

IV. Random Walk

IV.A. Source

KENO-V is a static (time-independent) Monte Carlo criticality code, and the normal batch feature produces true neutron generations — the fissions of one generation being used as the source for the following generation. There are several options available for creating the spatial distribution of the first generation source. Among these are (1) point source, (2) cosine distribution, (3) uniform in volume, (4) specific coordinates for each neutron, and (5) an output file from a previous calculation (restart option). Neutrons are started isotropically in direction, and an energy group is determined from the fission spectrum of the mixture of the starting location. Locations in non-fissile mixtures are discarded (unless a fission spectrum has been specified) until the desired number of starting

neutrons is reached. The procedure is the same for subsequent generations, the starting location being that where the fission neutron was created. Fission neutrons are created using a collision density estimator and the fission production probability, $\nu\Sigma_f/\Sigma_t$, in a manner which keeps both the number and weight (unity) of neutrons starting each generation constant, regardless of the value of k_{eff}. During each generation, the parameters of the fission neutrons produced are stored in an area of computer memory called the "fission bank," which becomes the source file for the next generation.

Since the first generation spatial distribution will not, in general, be correct, several generations must be skipped for the purpose of estimating results until the correct distributions of fission neutron production has been achieved. Otherwise, the results will be in error. Observation of the generation-by-generation k_{eff} in the output will indicate if enough generations have been skipped. It has been found that this fission source convergence takes fewer generations if the initial distribution produces too high a reactivity rather than too low.

IV.B. Transport and Collision

Neutrons are transported in the standard method of mean-free-path exhaustion across boundaries between different mixtures. The outgoing energy and direction from scattering are selected using a polynominal P_N expansion technique similar to that in the MORSE code.[9] All mathematical moments from the scattering distribution formulation are preserved through N + 1 moments, producing energy and angle scattering probabilities for m discrete angles where $m = (N + 1)/2$. The code will produce angles and probabilities only for a valid distribution, a condition sometimes violated by a request for large m for which data does not exist, and the code automatically inserts zeros for the missing data. In this case the code reverts to a valid distribution resulting from the available data and prints out various warning messages.

The m discrete directions are not fixed in the system geometry but are determined for each collision as the possible angles between the incoming and outgoing directions, i.e., the

scattering angle. Thus, any multiple scattering coupled with the isotropic production of fission neutrons precludes any directional bias in the calculation. For an even N, the entire procedure is performed as for the next lower (odd) integer, N - 1. It has been found that a P_3 calculation is often superior to that for P_1, but extension to P_5 and beyond rarely gives improvement but can increase computation time and memory requirements. If a P_1 expansion is requested, the code does not pick the one discrete angle, but uniformly from some interval on either side of the angle. Even if a P_0 treatment is not requested, any isotropic distribution from the data is automatically recognized and selected from accordingly.

IV.C. Super Groups

If all the energy dependent information necessary for a calculation cannot be contained in the available computer memory at one time, the code automatically moves this data into and out of memory via direct access devices until the calculation is complete. The collection of groups in memory at one time is called a super group, and the data affected includes all cross sections and related distributions, biasing parameters, albedo data, and any energy dependent calculated quantities such as fluxes. The procedure begins during data processing when the code determines the minimum number of super groups necessary. The grouping starts with the lowest energy groups (highest group numbers) in an effort to minimize the number of super groups containing thermal up-scatter data, thus minimizing the number of super group exchanges. If the data from any one group is too large, more computer memory must be made available or the amount of information decreased in some manner (simplify the input). Use of this super group feature allows KENO-V to be executed on small computers, but at the expense of longer computation times and more input/output operations due to use of direct access devices and various extra bookkeeping operations in the code.

In the calculation, when a neutron emerges from a collision (or source) event in an energy group not contained in the current supergroup, the neutron parameters are stored in a "neutron bank" corresponding to the appropriate supergroup, its processing continuing

when that supergroup data is read into memory. This process continues until all neutrons in a generation are processed through all appropriate groups. The entire process is then repeated for the source neutrons of the next generation, starting with the supergroup containing the most source neutrons.

IV.D. Biasing and Weighting

The only biasing in KENO-V is non-absorption weighting accompanied by Russian roulette and splitting for neutron population and weight control. The weight control parameters have input default values which have been shown to be near optimal in terms of overall calculational efficiency for a large number of problem types.[10] The default values may be overridden by input of energy group and region dependent values of WTA, the weight assigned a neutron which survives the Russian roulette game. Splitting and Russian roulette occur when the neutron weight is above WTH*WTA or below WTL*WTA, respectively, where WTH and WTL are input constants identical for all regions and energy groups.

A special weighting option is available for a reflector in order to reduce the calculation time spent as neutrons leave and re-enter the core. The input is the same form as for the general Russian roulette and splitting input, except that the data is already contained in a "weighting library" (file 80 of the RSIC code package) and is accessed by an identification similar to cross section data. This reflector weighting data has been generated from adjoint one-dimensional discrete ordinate k_{eff} calculations. A representative core was used in conjunction with several reflector materials. The reflector WTA values are proportional to the inverse of the adjoint flux for each energy group and several spatial increments, and they are normalized by the corresponding data in the core-reflector interface region. Data is available for the following materials and energy group structures.

Material	Available Group Structure
Concrete	16, 27, 123
Paraffin	16, 27 123
Water	16, 27, 123, 218
Graphite	16, 27, 123

This data can be used in combination for multi-material reflectors. For materials other than the four given, the data deemed to most closely represent it should be used. Inappropriate use of this weighting data will still, in theory, give correct results, but at a cost in efficiency in terms of computing time and statistical error. Even correct use of the data can lead to inefficient calculation of quantities other than k_{eff}, such as leakage.

The desired scheme in the reflector is to decrease the number of neutrons as they move into regions of smaller importance away from the core. This is done by playing Russian roulette, and the few surviving neutrons have higher weights. The total weight is conserved, but less computation time is spent on unimportant neutrons (computation time per neutron is independent of the weight). As neutrons migrate back toward the core into regions of higher importance, they are split into more neutrons of smaller weight. Thus, neutrons should re-enter the core with the same distributions in all variables as if this special reflector biasing had not been done. It has been found that reflector biasing will reduce most KENO-V problem calculation times by approximately a factor of two.

IV.E. Reflector Albedo Data

Another method for treating a reflector in KENO-V is to simulate the tracking process by use of albedo data. When a neutron encounters a reflector surface designated as an albedo surface, the neutron is directed back into the core immediately with a distribution as determined by random selection from albedo data for that surface. Thus, no time at all is spent in the reflector. This data is on file 79 of the RSIC code package for the following materials:

Material	Thickness (cm)
Water	30.48
Paraffin	30.48
Graphite	200.0
Polyethylene	30.48
Concrete	10.6, 20.32, 30.48 40.64, 60.96

The data was compiled from forward one-dimensional discrete ordinates calculations for each material and thickness. The 16 group Hansen-Roach group structure was used with 4 incident and 4 exit polar angles. Azimuthal symmetry is assumed. This 16 group albedo data may be used with the other distributed group structures because the code automatically matches the cross section data with albedo data group boundaries. For group structures where group boundaries do not match, the code makes certain interpolation approximations which could be rather crude in some cases.

For the maximum thicknesses given, the albedo data corresponds to an effectively infinite medium, and thicker reflectors may be correctly simulated by it. Unlike the reflector weighting data, use of albedo data for materials not closely related to the scattering characteristics of those given will result in incorrect answers. It is assumed that there is no spatial dependence of the albedo, i.e., the neutron returns to the core at the same position where it struck the albedo surface. This could introduce some error in small reflectors or where corners are important, because the data was calculated for infinite slabs of material.

It has been found that use of albedo data can reduce the computation time of many reflected systems by as much as a factor of 10. Of course, no quantities such as leakage or reflector fluxes can be calculated. Also, the system generation time and lifetime calculations will be in error. The albedo option, without albedo data, can also be used to create several system boundary conditions. These are (1) vacuum, (2) specular (mirror image) reflection, and (3) periodic conditions for simulating an infinite array.

IV.F. Adjoint Calculations

Because of the multigroup cross section treatment in KENO-V, adjoint calculations are a simple task. The entire cross section processing sequence must be done in the adjoint

mode, and the transposed scattering matrix is transferred to the code. The code automatically adjoints all energy related quantities such as cross section group structure, weighting and albedo data, fission spectra, etc. Any energy dependent output, such as adjoint fluxes, has no meaning in relationship to corresponding forward calculated quantities. There is no great advantage in adjoint capability for criticality calculations as there is for certain fixed source Monte Carlo calculations. The system k_{eff} is determined with approximately the same effort in either the forward or adjoint mode. There may be some improvement in some cases for other quantities, such as leakage, when the adjoint mode is used.

V. Estimation of Results

The accumulation of results in a KENO-V calculation is done simultaneously with the random walk. There is no post-processing of results or history file creation, although the restart file capability could be extended for this use. All quantities in KENO-V, except leakage (boundary crossing) and fluxes (track length), are calculated from a collision density estimator using WT, the neutron weight before collision. As a result, differential results in very thin spatial regions may have high statistical errors. All results, except k_{eff}, are compiled on a per source neutron basis only for those generations following the initial generations skipped for source convergence.

V.A. Generation k_{eff}

The k_{eff} for each generation is defined as the sum of the fission neutron weight produced in a generation divided by the sum of the weight of the source neutrons for that generation. The k_{eff} for each generation is output along with the average k_{eff} through that generation. Another edit gives the average k_{eff} after a given number of generations has been skipped. Included here are confidence limits corresponding to one, two, and three standard deviations, (σ, 2σ, 3σ).

V.B. Matrix k_{eff}

An optional k_{eff} calculation is made by calculating a matrix giving the probability that a neutron created in one geometry UNIT will produce a fission neutron in each of the other UNITS. The procedure is similar to that given in reference 11. The largest eigenvalue of this matrix gives k_{eff}. The calculation may be done for UNIT type or UNIT position. Additional information given is the probability matrix, source fraction, and cofactor k_{eff} by UNIT position or type. The cofactor k_{eff} is that value calculated as if that UNIT

or position fission source had been excluded from the system.

V.C. Fission, Absorptions, and Leakage

These compilations give at each collision, fission, $\upsilon\Sigma_f/\Sigma_t$*WT, and absorption, Σ_a/Σ_t*WT, fractions by region and energy group and leakage, WT_{ESC} (weight at escape), fraction from the system outer boundary by energy group. The sum of the leakage and absorption, in the absence of fluctuations produced by Russian roulette, should add to unity. The sum of the fission fractions is k_{eff}. The fission density is the fission fraction divided by the region volume.

V.D. Fluences

Fluxes, WT/Σ_t, in units of neutrons/cm^2/source neutron are printed for each energy group for each UNIT and each region in each UNIT. Calculations of fluxes are optional and could result in a large amount of printout for some problems.

V.E. Lifetime and Generation Time

The system lifetime, ℓ, and generation time, Λ, are calculated by KENO-V for the entire system. The generation time is compiled at each collision from $\upsilon\Sigma_f/\Sigma_t$*WT*T, where T is elapsed time since the source event. This time is determined from the flight distance divided by the velocity. The velocities can be input, or if not, they are calculated by the code from the energy corresponding to the mid-point lethargy of the energy group.

The generation time is the average time between successive neutron generations and can be thought of as the mean time to birth (fission). Likewise, the lifetime is the average life span of a neutron and can be thought of as the mean time to death (absorption or leakage). The ℓ is compiled at each collision and leakage occurrence as Σ_a/Σ_t)*WT*T + WT_{ESC}*T_{ESC} where T_{ESC} is the time from source to escape.

V.F. Extra Cross Section Data

A provision has been made in the cross section input to provide extra data not actually needed to calculate the system. This data must be present throughout the entire cross section processing procedure. It is made available in the estimation process in the form of response functions for possible calculations of reaction rates, central reactivity worths, and other such quantities. The implementation of the automated use of this extra data has not been completed in the current version of KENO-V. The only present use of this option is to calculate, $\bar{\upsilon}$, the average number of neutrons per fission from Σ_f(the code uses $\upsilon\Sigma_f$ as one quantity for normal operation).

V.G. Statistical Uncertainty

Most output quantities in KENO-V are accompanied by an indication of the statistical error associated with one standard deviation of the mean (the final k_{eff} edit also gives two and three deviations). This error is given in one of three ways: (1) absolute (plus or minus the mean result), (2) fractional (absolute error divided by the mean), and (3) per cent (fractional error x 100). The uncertainty calculation in KENO-V, as in other criticality codes, assumes a normal distribution of results from independent generations. An optional output feature gives a computer plot of the distribution of the generation k_{eff} as an aid to determine the applicability of this assumption for each calculation. The tabulated values of k_{eff} should also be scrutinized in order to make judgements on the effect of the initial source distribution and later fluctuations in k_{eff}. All Monte Carlo criticality uncertainty calculations are plagued with a theoretical bias due to the fact that the generations are not independent (as are the batches in a fixed source calculation), since the source for one generation is derived fron the previous generation. In practice, there is usually no problem unless the initial soure distribution never converges to the true distribution. In this case, the calculated uncertainties may give unrealistically high or low values.

VI. Input/Output and Sample Problems

The documentation of the KENO-V module in SCALE contains many examples of input card images and output listings. There is also an extensive list of code generated error message and their explanation and/or rectification. There are sixteen sample problem input listings given, accompanied by various photographs and sketches of the problem geometries. Complete output listings are given for all problems either printed in the report or on microfiche included in the SCALE document. These sample problems are also included in the RSIC code package (file 52, input; file 127, output; files 78-80 with files 34-36, cross section, weighting and albedo data). All sample problems contain Hansen-Roach nuclide identification which are listed in the sample problem section of the KENO-V manual in SCALE.

A photograph of the assembly[12] for sample problem 12 is shown in Fig. 3. In Fig. 4 are listed the input data for this composite array of four highly enriched uranium metal cylinders and four cylindrical plexiglass containers filled with uranyl nitrate solution. The next three figures show selected portions of the output. The average generation k_{eff} in Fig. 5 always begins with the third generation regardless of the number of generations to be skipped (three in this case) before compilation of other results begins. The matrix k_{eff} begins with the first generation. Fig. 6 gives the average k_{eff} as if other numbers of generations had been skipped. The first value here, 1.00979, differs from the final value in Fig. 5 only because one more generation has been skipped. A frequency distribution of k_{eff} values for various generation combinations is shown in Fig. 7. One asterisk equals one occurrence of k_{eff} in an interval.

From the data available in these last three figures, the system k_{eff} is selected. This value is usually the average k_{eff} from Fig. 6 with the lowest deviation. If the initial generation source is slow in converging, this value may not necessarily be the first listed. The data in the other two figures will indicate if there is any trouble in source convergence.

ORNL NO. 70801

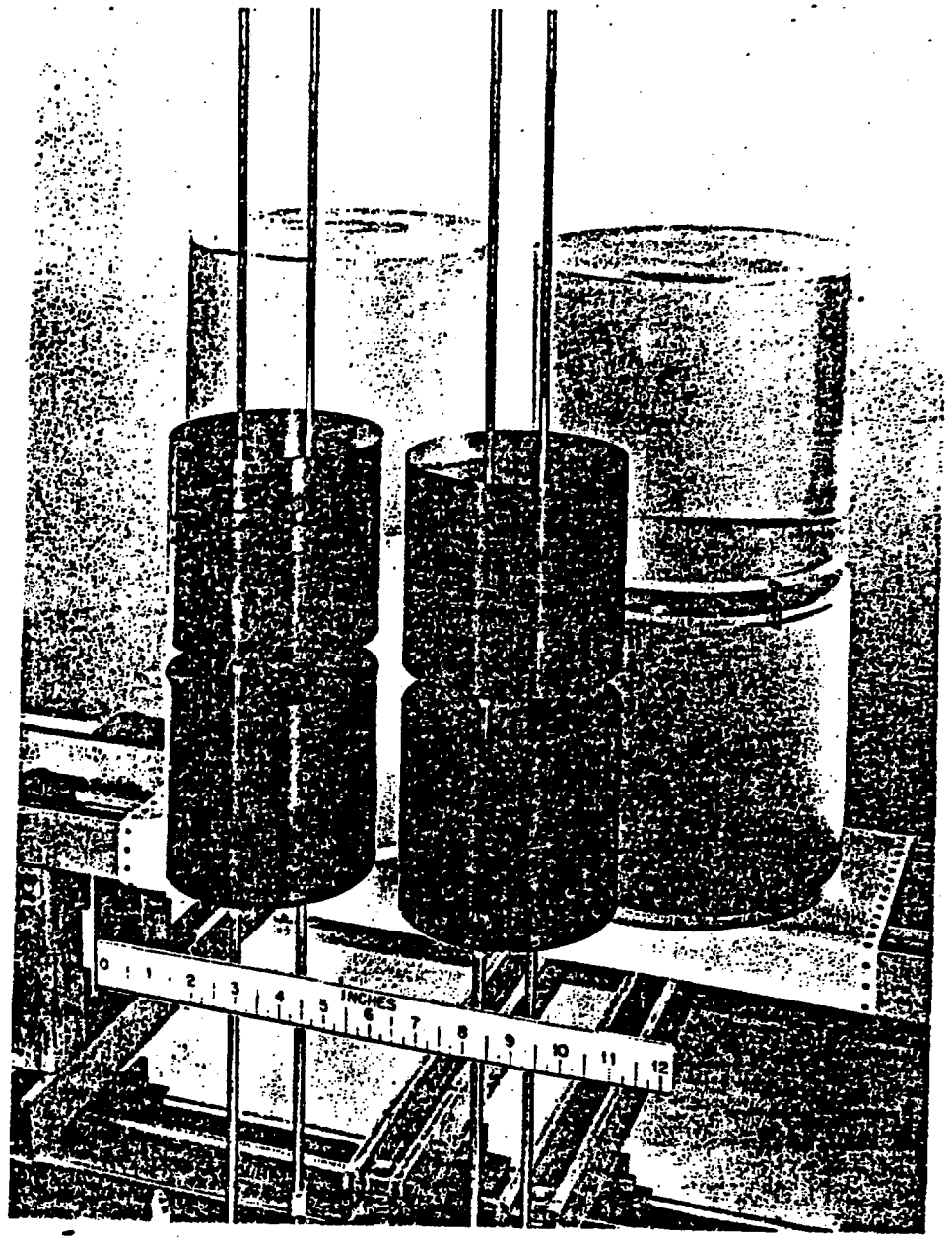


Fig. 3. Critical Assembly of 4 Solution Units and 4 Metal Units.

```
SAMPLE PROBLEM 12 4 AQUEOUS 4 METAL MIXED UNITS
READ PARAM  LIB=41  FLX=YES FDN=YES  THE=5.0 NUB=YES SMU=YES MKP=YES
MKU=YES FMP=YES END PARAM
READ MIXT SCT=1  MIX=1  92860 3.2275-3  92501 4.4802-2  MIX=2  1102 5.81-2
7100 1.9753-3  8100 3.6927-2  92501 9.8471-4  92860 7.7697-5
MIX=3  6100 3.5552-2  1102 5.6884-2  8100 1.4221-2  END MIXT
READ GEOM
BOX TYPE 1
CYLINDER  2 1 9.525 8.89 -8.89
CYLINDER  3 1 10.16 9.525 -9.525
CUBOID  0 1 10.875 -10.875 10.875 -10.875 10.24 -10.24
BOX TYPE  2
CYLINDER  1 1 5.748 5.3825 -5.3825
CUBOID  0 1 6.59 -15.16 6.59 -15.16 6.225 -14.255
BOX TYPE  3
CYLINDER  1 1 5.748 5.3825 -5.3825
CUBOID  0 1 6.59 -15.16 15.16 -6.59 6.225 -14.255
BOX TYPE  4
CYLINDER  1 1 5.748 5.3825 -5.3825
CUBOID  0 1 6.59 -15.16 6.59 -15.16 14.255 -6.225
BOX TYPE  5
CYLINDER  1 1 5.748 5.3825 -5.3825
CUBOID  0 1 6.59 -15.16 15.16 -6.59 14.255 -6.225
END GEOM  READ ARRAY  NUX=2 NUY=2 NUZ=2  LOOP
1 3R2 1 2 1 1 2 1  2 9R1  3 3R1 2 2 1 3R1  4 6R1 2 2 1  5 3R1 2 2 1 2 2 1
END ARRAY
END DATA
```

Fig. 4. Card Input Images for KENO V Sample Problem #12.

SAMPLE PROBLEM 12 4 AQUEOUS 4 METAL MIXED UNITS

GENERATION	GENERATION K-EFFECTIVE	ELAPSED TIME MINUTES	AVERAGE K-EFFECTIVE	AVG K-EFF DEVIATION	MATRIX K-EFFECTIVE	MATRIX K-EFF DEVIATION
1	8.66527E-01	4.00000E-02	1.00000E+00	0.0	9.79134E-01	0.0
2	1.09597E+00	5.88333E-02	1.00000E+00	0.0	1.10340E+00	5.19832E-02
3	9.20249E-01	7.45000E-02	9.20249E-01	0.0	1.05033E+00	4.46176E-02
4	1.00423E+00	9.21667E-02	9.62241E-01	4.19929E-02	1.05201E+00	3.76604E-02
5	1.03887E+00	1.10167E-01	9.87783E-01	3.52164E-02	1.04205E+00	3.92122E-02
6	9.35916E-01	1.25833E-01	9.74816E-01	2.80751E-02	1.01091E+00	3.52512E-02
7	9.29246E-01	1.41667E-01	9.65702E-01	2.35801E-02	9.92464E-01	3.18704E-02
8	9.51662E-01	1.57500E-01	9.63395E-01	1.93913E-02	9.89424E-01	2.93697E-02
9	1.06618E+00	1.75167E-01	9.78079E-01	2.20038E-02	1.00267E+00	2.78655E-02
10	1.03545E+00	1.92000E-01	9.85251E-01	2.03609E-02	1.00591E+00	2.70376E-02
11	1.11733E+00	2.10333E-01	9.99927E-01	2.31895E-02	1.01277E+00	2.67038E-02
12	9.74897E-01	2.26167E-01	9.97424E-01	2.08932E-02	1.01130E+00	2.74552E-02
13	1.07753E+00	2.44167E-01	1.00471E+00	2.02575E-02	1.01591E+00	2.58240E-02
14	1.01685E+00	2.60833E-01	1.00573E+00	1.85219E-02	1.01504E+00	2.50836E-02
15	9.91040E-01	2.78167E-01	1.00460E+00	1.70712E-02	1.01284E+00	2.40948E-02
16	9.59711E-01	2.94500E-01	1.00139E+00	1.61286E-02	1.00973E+00	2.31418E-02
17	1.02644E+00	3.11667E-01	1.00309E+00	1.51108E-02	1.00979E+00	2.22241E-02
18	1.00110E+00	3.29167E-01	1.00296E+00	1.41360E-02	1.00766E+00	2.19264E-02
19	9.68668E-01	3.45667E-01	1.00094E+00	1.34344E-02	1.00560E+00	2.13461E-02
20	1.09780E+00	3.63833E-01	1.00632E+00	1.37598E-02	1.01047E+00	2.08720E-02
21	9.54717E-01	3.79667E-01	1.00361E+00	1.32986E-02	1.00778E+00	2.03766E-02
22	9.58608E-01	3.95833E-01	1.00136E+00	1.28136E-02	1.00558E+00	2.03086E-02
23	9.57813E-01	4.12000E-01	9.99284E-01	1.23633E-02	1.00421E+00	2.01733E-02
24	9.99677E-01	4.29167E-01	9.99302E-01	1.17888E-02	1.00411E+00	1.98192E-02
25	1.02595E+00	4.46167E-01	1.00046E+00	1.13256E-02	1.00558E+00	1.92374E-02
26	8.96247E-01	4.61667E-01	9.96117E-01	1.16815E-02	1.00133E+00	1.88969E-02
27	1.06863E+00	4.80167E-01	9.99017E-01	1.15752E-02	1.00396E+00	1.85242E-02
28	1.01752E+00	4.97500E-01	9.99732E-01	1.11451E-02	1.00432E+00	1.80812E-02
29	9.47758E-01	5.13667E-01	9.97807E-01	1.08967E-02	1.00251E+00	1.77485E-02
30	9.79810E-01	5.29833E-01	9.97164E-01	1.05212E-02	1.00232E+00	1.72949E-02
•						
•						
•						
•						
85	1.01315E+00	1.45733E+00	1.00514E+00	5.98311E-03	1.00731E+00	1.06790E-02
86	8.99401E-01	1.47200E+00	1.00388E+00	6.04375E-03	1.00599E+00	1.05952E-02
87	1.11237E+00	1.48967E+00	1.00516E+00	6.10607E-03	1.00716E+00	1.05362E-02
88	1.01922E+00	1.50600E+00	1.00532E+00	6.03762E-03	1.00721E+00	1.04883E-02
89	9.75695E-01	1.52200E+00	1.00498E+00	5.97742E-03	1.00680E+00	1.04487E-02
90	1.03313E+00	1.53883E+00	1.00530E+00	5.91801E-03	1.00705E+00	1.03757E-02
91	1.04500E+00	1.55617E+00	1.00574E+00	5.86861E-03	1.00742E+00	1.03091E-02
92	1.05817E+00	1.57383E+00	1.00633E+00	5.83188E-03	1.00794E+00	1.02124E-02
93	1.00156E+00	1.59067E+00	1.00628E+00	5.76733E-03	1.00784E+00	1.01330E-02
94	1.06903E+00	1.60867E+00	1.00696E+00	5.74559E-03	1.00844E+00	1.00575E-02
95	9.48173E-01	1.62467E+00	1.00633E+00	5.71972E-03	1.00771E+00	9.95576E-03
96	1.01861E+00	1.64133E+00	1.00646E+00	5.65906E-03	1.00778E+00	9.90270E-03
97	9.74680E-01	1.65717E+00	1.00612E+00	5.60902E-03	1.00742E+00	9.84926E-03
98	1.06813E+00	1.67483E+00	1.00677E+00	5.58661E-03	1.00805E+00	9.82353E-03
99	1.02618E+00	1.69183E+00	1.00697E+00	5.53415E-03	1.00819E+00	9.76077E-03
100	1.12873E+00	1.71017E+00	1.00821E+00	5.61605E-03	1.00932E+00	9.71434E-03
101	9.97908E-01	1.72633F+00	1.00811E+00	5.56027E-03	1.00910E+00	9.66513E-03
102	9.94445E-01	1.74250F+00	1.00787E+00	5.51118E-03	1.00876E+00	9.60668E-03
103	1.11203E+00	1.76067F+00	1.00890E+00	5.55097E-03	1.00973E+00	9.55387E-03

KENO MESSAGE NUMBER K5-123 EXECUTION TERMINATED DUE TO COMPLETION OF THE SPECIFIED NUMBER OF GENERATIONS.

THE MATRIX K-EFFECTIVE IS THE LARGEST EIGENVALUE OF THE MATRIX OF FISSION PROBABILITIES BY ARRAY POSITION..
THERE ARE NRXMAX * NRYMAX * NRZMAX POSITIONS IN AN ARRAY.

Fig. 5. KENO V Generation k_{eff} Printout.

SAMPLE PROBLEM 12 4 AQUEOUS 4 METAL MIXED UNITS

LIFETIME = 3.27663E-06 + OR - 4.18352E-08
NU BAR = 2.50379E+00 + OR - 8.37891D-04

GENERATION TIME = 3.94345E-06 + OR - 5.35632E-08
AVERAGE FISSION GROUP = 9.57986E+00 + OR - 5.75226E-02
SELF MULTIPLICATION = 6.52516E-01 + OR - 4.73999E-03

NO. OF INITIAL GENERATIONS SKIPPED	AVERAGE K-EFFECTIVE	DEVIATION	67 PER CENT CONFIDENCE INTERVAL	95 PER CENT CONFIDENCE INTERVAL	99 PER CENT CONFIDENCE INTERVAL	NUMBER OF HISTORIES
3	1.00979	+ OR - 0.00553	1.00426 TO 1.01532	0.99873 TO 1.02084	0.99320 TO 1.02637	30000
4	1.00984	+ OR - 0.00558	1.00426 TO 1.01543	0.99868 TO 1.02101	0.99309 TO 1.02660	29700
5	1.00955	+ OR - 0.00563	1.00391 TO 1.01518	0.99828 TO 1.02081	0.99265 TO 1.02645	29400
6	1.01031	+ OR - 0.00564	1.00467 TO 1.01595	0.99903 TO 1.02159	0.99339 TO 1.02723	29100
7	1.01115	+ OR - 0.00563	1.00552 TO 1.01679	0.99988 TO 1.02242	0.99425 TO 1.02806	28800
8	1.01178	+ OR - 0.00566	1.00612 TO 1.01744	1.00046 TO 1.02309	0.99480 TO 1.02875	28500
9	1.01120	+ OR - 0.00569	1.00551 TO 1.01689	0.99982 TO 1.02258	0.99413 TO 1.02827	28200
10	1.01094	+ OR - 0.00575	1.00519 TO 1.01668	0.99945 TO 1.02243	0.99370 TO 1.02817	27900
11	1.00978	+ OR - 0.00569	1.00409 TO 1.01547	0.99840 TO 1.02116	0.99271 TO 1.02685	27600
12	1.01016	+ OR - 0.00574	1.00442 TO 1.01590	0.99869 TO 1.02164	0.99295 TO 1.02738	27300
17	1.00992	+ OR - 0.00599	1.00393 TO 1.01590	0.99794 TO 1.02189	0.99196 TO 1.02787	25800
22	1.01077	+ OR - 0.00617	1.00460 TO 1.01694	0.99843 TO 1.02311	0.99226 TO 1.02928	24300
27	1.01215	+ OR - 0.00631	1.00584 TO 1.01846	0.99953 TO 1.02477	0.99323 TO 1.03108	22800
32	1.01312	+ OR - 0.00667	1.00645 TO 1.01978	0.99978 TO 1.02645	0.99312 TO 1.03312	21300
37	1.01217	+ OR - 0.00702	1.00516 TO 1.01919	0.99814 TO 1.02620	0.99113 TO 1.03322	19800
42	1.01284	+ OR - 0.00721	1.00563 TO 1.02004	0.99842 TO 1.02725	0.99122 TO 1.03446	18300
47	1.01252	+ OR - 0.00765	1.00487 TO 1.02017	0.99722 TO 1.02781	0.98957 TO 1.03546	16800
52	1.01195	+ OR - 0.00780	1.00414 TO 1.01975	0.99634 TO 1.02756	0.98853 TO 1.03536	15300
57	1.01377	+ OR - 0.00859	1.00518 TO 1.02236	0.99660 TO 1.03094	0.98801 TO 1.03953	13800
62	1.01471	+ OR - 0.00922	1.00749 TO 1.02594	0.99827 TO 1.03516	0.98904 TO 1.04438	12300
67	1.01530	+ OR - 0.01011	1.00519 TO 1.02541	0.99509 TO 1.03552	0.98498 TO 1.04562	10800
72	1.01841	+ OR - 0.01128	1.00712 TO 1.02969	0.99584 TO 1.04098	0.98455 TO 1.05226	9300
77	1.02515	+ OR - 0.01258	1.01258 TO 1.03772	1.00000 TO 1.05030	0.99742 TO 1.06287	7800
82	1.02325	+ OR - 0.01270	1.01055 TO 1.03595	0.99785 TO 1.0486[illegible]	0.98514 TO 1.06135	6300
87	1.02880	+ OR - 0.01247	1.01633 TO 1.04128	1.00385 TO 1.05375	0.99138 TO 1.06623	4800
92	1.02995	+ OR - 0.01739	1.01256 TO 1.04734	0.99517 TO 1.06473	0.97778 TO 1.08212	3300

Fig. 6. KENO V k_{eff} for Shipped Generations.

```
SAMPLE PROBLEM 12 A AQUEOUS & METAL MIXED UNITS

                                FREQUENCY FOR GENERATIONS    4 TO  103
0.8750 TO 0.8981      *
0.8981 TO 0.9212      **
0.9212 TO 0.9443      *****
0.9443 TO 0.9674      ****************
0.9674 TO 0.9905      **************
0.9905 TO 1.0136      **********************
1.0136 TO 1.0367      ************
1.0367 TO 1.0598      **********
1.0598 TO 1.0829      **********
1.0829 TO 1.1060      **
1.1060 TO 1.1291      *****
1.1291 TO 1.1522      *
1.1522 TO 1.1752
1.1752 TO 1.1983      *

                                FREQUENCY FOR GENERATIONS   29 TO  103
0.8750 TO 0.8981
0.8981 TO 0.9212      **
0.9212 TO 0.9443      ***
0.9443 TO 0.9674      ***********
0.9674 TO 0.9905      ***********
0.9905 TO 1.0136      ******************
1.0136 TO 1.0367      ********
1.0367 TO 1.0598      *********
1.0598 TO 1.0829      *******
1.0829 TO 1.1060      *
1.1060 TO 1.1291      ****
1.1291 TO 1.1522      *
1.1522 TO 1.1752
1.1752 TO 1.1983      *

                                FREQUENCY FOR GENERATIONS   54 TO  103
0.8750 TO 0.8981
0.8981 TO 0.9212      **
0.9212 TO 0.9443      **
0.9443 TO 0.9674      ******
0.9674 TO 0.9905      ******
0.9905 TO 1.0136      **************
1.0136 TO 1.0367      *****
1.0367 TO 1.0598      ******
1.0598 TO 1.0829      *****
1.0829 TO 1.1060
1.1060 TO 1.1291      ***
1.1291 TO 1.1522
1.1522 TO 1.1752
1.1752 TO 1.1983      *
```

Fig. 7. KENO V k_{eff} Frequency Distribution Plot.

VII. Future Developments

The extension of KENO-V, KENO-V.a, and its documentation are essentially complete, but the new code has not yet been released. The principle improvements are in the geometry capability as explained in Section III. A planned extension is to have triangular array shapes as well as rectangular. The new code also gives computer printer plots of the information contained in the last three figures of the last section to aid in the k_{eff} determination.

REFERENCES

1. KENO V module in SCALE (reference 6).
2. G. E. Whitesides and N. F. Cross, "KENO — A Multigroup Monte Carlo Criticality Program," CTC-5, Oak Ridge Computing Technology Center (1969).
3. G. E. Hansen and W. H. Roach, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS 254[illegible], Los Alamos Scientific Laboratory (1961).
4. C. M. Petrie and N. F. Cross, "KENO IV — An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Laboratory (1975).
5. MARS module in SCALE (reference 6).
6. "SCALE, A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation," ORNL/NUREG/CSO-2/R1, Oak Ridge National Laboratory.
7. N. M. Greene, et al., "AMPX, A Modular System for Multigroup Cross Section Generation and Manipulation," ORNL/TM-3706, Oak Ridge National Laboratory (1976).
8. "A Review of Multigroup Nuclear Cross Section Processing," ORNL/RSIC-41, Oak Ridge National Laboratory (1978).
9. M. B. Emmett, "The MORSE Monte Carlo Radiation Transport Code System," ORNL-4972, Oak Ridge National Laboratory (1975).
10. T. J. Hoffman, "The Optimization of Russian Roulette Parameters for the KENO Computer Code," ORNL/CSD/TM-192, Oak Ridge National Laboratory (1982).
11. G. W. Morrison, et al., "REACT and CONVRG-FORTRAN Subroutines for Determining Source Convergence for the OSR Monte Carlo Neutron Transport Code," ORNL/TM-1325, Oak Ridge National Laboratory (1966).
12. J. T. Thomas, "Critical Three-Dimensional Arrays of U(93.2)-Metal Cylinders," *Nucl. Sci. Eng.* 52, 350 (1973).