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M. E. Dunn, N. M. Greene, D. F. Hollenbach and L. M. Petrie

*Oak Ridge National Laboratory,**
P.O. Box 2008, MS-6170,
Oak Ridge, TN 37831-6170, USA
dunnme@ornl.gov
(865) 574-5260

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INTRODUCTION

KENO V.a[1] is a multigroup Monte Carlo code that is used throughout the world to analyze fissionable systems for criticality safety applications. The KENO series of codes are developed and maintained at the Oak Ridge National Laboratory (ORNL) as part of the SCALE[1] (**S**tandardized **C**omputer **A**nalyses for **L**icensing **E**valuation) system. The objective of this work is to develop and test a new version of KENO V.a that utilizes pointwise cross sections to model the radiation transport.

Previous work[2] has focused on the development of a prototypic version of KENO V.a. that performs the random walk using continuous-energy MCNP[3] cross sections. Although the previous work was successful, the prototypic KENO V.a is not available for widespread distribution and is limited to ENDF (**E**valuated **N**uclear **D**ata **F**ile) data prior to Version VI.[4] Because the format and structure of the MCNP cross-section data inherently dictate the transport procedures used in the random walk, the prototypic continuous-energy KENO V.a is not completely independent from the MCNP transport procedures. As a result, this work has focused on the development of a production-level continuous-energy version of KENO V.a that is completely independent from other pointwise Monte Carlo codes.

DESCRIPTION OF ACTUAL WORK

A continuous-energy or pointwise version of KENO V.a has been developed by first designing a new continuous-energy cross-section format and secondly by developing the appropriate Monte Carlo transport procedures to sample the new cross-section format. The nuclear data within the ENDF system are voluminous in nature and cannot be used directly in radiation transport codes. As a result, a cross-section processing system must be used to process the ENDF data and generate nuclear data libraries that can be accessed by radiation transport codes.

In order to facilitate the development of independent transport procedures, a new continuous-energy cross-section library structure has been developed for KENO, and multiple processing modules have been developed for the AMPX-2000 (Ref. 5) cross-section processing system that is developed and maintained at ORNL. The details of the continuous-energy library format are beyond the scope of this summary; however, the essential components of a Point KENO cross-section library include the following: (1) average number of neutrons (delayed and prompt) produced by fission, $\bar{\nu}$ (E); (2) one-dimensional continuous-energy cross sections as a function of temperature, σ (E, T); (3) two-dimensional pointwise joint probability distributions that describe the energy and angle of particles emerging from a collision, $f(E \div E', \mu)$; and (4) probability tables for sampling the cross sections in the unresolved-resonance region.

Following the cross-section methods development, a pointwise version of KENO V.a has been developed to read the AMPX-generated cross sections and perform continuous-energy transport calculations for criticality applications. Additional details about the continuous-energy cross sections and transport procedures will be provided in the full paper. AMPX has been used to generate pointwise cross sections for a test library consisting of 50 ENDF/B-VI Release 7 nuclides (i.e., includes all the uranium and plutonium isotopes in addition to thermal data for H in H_2O and H in CH_2).

RESULTS

In an effort to demonstrate and test the computational capability of Point KENO, 21 critical benchmarks were selected from the work by Bowman *et al.*[6] and modeled with the continuous-energy version of KENO V.a. The calculated results are provided in Table I. Using the case designation specified in Ref. 6, the first 13 benchmarks in Table I involve LWR-type UO_2 fuel pin lattices with various absorber and reflector configurations. The next four benchmarks in Table I (i.e., cas82 through cas85) are the "green block" experiments that involve 2–3 wt % enriched homogenized uranium in paraffin blocks. Cases cas86 through cas88 involve uranyl fluoride and uranyl nitrate solutions at 93.2 wt % enrichment, and cas91 consists of uranyl fluoride solution at 4.89 wt % enrichment.

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All of the calculations were performed on a DEC Alpha XP1000 workstation. For comparison purposes, the 21 benchmarks were modeled with the multigroup version of KENO V.a using the 238-group ENDF/B-V (Ref. 1) and 199-group ENDF/B-VI Release 3 (Ref. 7) cross-section libraries. The 199-group and 238-group libraries are general-purpose data libraries with adequate group structures for most fast and thermal criticality safety applications. One limitation of the 199-group library is the method used for self-shielding the resonances in the resolved region. The 199-group library was generated using NJOY 91 (Ref. 8), and full energy range Bondarenko factors are provided in the library to perform resonance self shielding. As a result, the narrow resonance approximation is used throughout the resonance region with the 199-group library. The failure of the narrow resonance approximation in the resolved region is observed for the 199-group library with the green block cases in Table I (i.e., cas82 through cas85), and these results are consistent with the results provided in Ref. 6. As noted in Ref. 6, the narrow resonance approximation in the 199-group library provides too much self shielding of the ^{238}U capture cross section, and the capture cross section is too low thereby causing an overprediction in k_{eff}. In contrast, the 238-group results and Point KENO results are closer to critical relative to the 199-group values because of improved modeling of the resonance self-shielding of the ^{238}U capture cross section. The Point KENO results for the remaining cases in Table I are consistent with the multigroup KENO V.a results and demonstrate the computational capability of the new continuous-energy version of KENO V.a.

TABLE I. Calculated k_{eff} for Selected Critical Benchmark Cases

Case	Enrichment (wt %)	Lattice water/fuel volume ratio*	Point KENO ENDF/B-VI Rel. 7	KENO V.a 199-group ENDF/B-VI Rel. 3	KENO V.a 238-group ENDF/B-V
cas01	2.35	2.92	0.9950 " 0.0016	0.9947 " 0.0015	0.9937 " 0.0015
cas07	2.35	2.92	0.9965 " 0.0018	0.9934 " 0.0016	0.9956 " 0.0015
cas09	2.35	2.92	0.9900 " 0.0018	0.9908 " 0.0015	0.9956 " 0.0014
cas19	2.35	1.6	1.0008 " 0.0016	0.9937 " 0.0022	0.9903 " 0.0014
cas21	2.35	1.6	0.9932 " 0.0016	0.9890 " 0.0017	0.9941 " 0.0016
cas34	2.46	0.999	0.9997 " 0.0016	0.9973 " 0.0011	0.9925 " 0.0015
cas36	2.46	1.84	0.9885 " 0.0017	0.9896 " 0.0014	0.9934 " 0.0014
cas37	2.46	1.84	0.9978 " 0.0017	0.9979 " 0.0020	0.9931 " 0.0015
cas38	2.46	1.84	0.9916 " 0.0017	0.9859 " 0.0015	0.9901 " 0.0016
cas46	2.35	1.196	1.0012 " 0.0019	0.9948 " 0.0017	0.9942 " 0.0014
cas47	2.35	2.408	0.9941 " 0.0013	0.9924 " 0.0010	0.9952 " 0.0012
cas48	2.35	2.408	0.9991 " 0.0011	0.9972 " 0.0011	0.9981 " 0.0010
cas50	2.35	3.687	0.9966 " 0.0016	0.9973 " 0.0011	1.0033 " 0.0014
cas82	2.0	$H/^{235}U$ = 293.9	1.0003 " 0.0014	1.0241 " 0.0015	1.0021 " 0.0016
cas83	2.0	$H/^{235}U$ = 406.3	1.0011 " 0.0019	1.0194 " 0.0017	0.9992 " 0.0017
cas84	3.0	$H/^{235}U$ = 133.4	1.0129 " 0.0021	1.0357 " 0.0017	1.0118 " 0.0017
cas85	3.0	$H/^{235}U$ = 133.4	1.0177 " 0.0016	1.0407 " 0.0020	1.0123 " 0.0020
cas86	93.2	$H/^{235}U$ = 1112	1.0068 " 0.0015	1.0026 " 0.0022	1.0088 " 0.0016
cas87	93.2	$H/^{235}U$ = 1270	0.9999 " 0.0020	1.0015 " 0.0014	1.0039 " 0.0013
cas88	93.2	$H/^{235}U$ = 186	1.0026 " 0.0023	1.0029 " 0.0025	1.0090 " 0.0021
cas91	4.89	$H/^{235}U$ = 524	1.0016 " 0.0014	1.0120 " 0.0012	1.0064 " 0.0013

* Unless designated as $H/^{235}U$ ratio

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