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Use of CPXSD for Generation of Effective Fast Multigroup Libraries for Pressure Vessel Fluence Calculations

ABSTRACT: Multigroup (i.e., broad-group) libraries play a significant role in the accuracy of transport calculations. There are several broad-group libraries available for particular applications. For example the 47-neutron (26 fast groups), 20-gamma-group BUGLE libraries are commonly used for light water reactor shielding and pressure vessel dosimetry problems. However, there is no publicly available methodology to construct group structures for a problem and objective of interest. Therefore, we have developed the CPXSD (Contributon and Point-wise Cross-Section Driven) methodology, which constructs effective fine- and broad-group structures. In this paper, we use the CPXSD methodology to construct broad-group structures for fast neutron dosimetry problems. It is demonstrated that the broad-group libraries generated from CPXSD constructed group structures, while only 14 groups (rather than 26 groups) in the fast energy range are in good agreement (~1-2%) with the fine-group library from which they were derived, in reaction rate calculations.

KEYWORDS: multigroup library, pressure vessel fluence, contributon, adjoint, cross section

Introduction

Multigroup (broad-group) libraries are required for deterministic transport calculations. However, one of the major uncertainties in deterministic transport calculations can be attributed to the multigroup cross-sections. Therefore, it is important to analyze the procedure used to generate a standard multigroup cross-section library and identify the processes that may be improved.

A standard broad-group cross-section library generation can be given as a three-step process, as listed below [*1*]:

Step (i) Point-wise evaluated data are processed to obtain fine-group cross sections.

Step (ii) Self-shielding calculations are performed to correct cross sections in resonance regions.

Step (iii) Self-shielded fine-group cross sections are collapsed into broad-group cross sections using scalar fluxes as the weighting function.

Among the three steps listed above, three items are identified that may improve a broad-group library generation: multigroup structure, self-shielding method, and cross-section collapsing approach. In order to obtain an effective multigroup structure, the CPXSD (Contributon and Point-wise Cross-Section Driven) methodology was developed by Alpan and Haghighat [*2,3*]. For fine- to broad-group collapsing, the bi-linear adjoint weighting technique [*4*] was developed by Hanshaw and Haghighat as an alternative to scalar flux weighting [*5*]. Improvements to the self-shielding method are beyond the scope of this paper.

The CPXSD methodology is an iterative method that constructs effective fine- and broad-group structures for a problem and objective of interest. In this paper, the CPXSD is utilized to

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generate broad-group libraries for fast neutron dosimetry calculations at the cavity dosimeter for a light water reactor. Scalar flux and bilinear adjoint weighting techniques are used in generating broad-group libraries.

The Multigroup Transport Equations

The discrete ordinates (S_N) method is commonly used to solve the neutron transport equation. In order to solve the time-independent neutron transport equation by the S_N method, discretization is performed over space ($\vec{r}$), energy (E), and angle ($\hat{\Omega}$). Since our focus is on multigroup calculations, we present the formulations for energy discretization in this section.

The time-independent multigroup neutron transport equation is derived by integrating the linear Boltzmann equation over each energy interval g, as given below.

$$\begin{aligned}&\hat{\Omega}\cdot\nabla\int_g dE\Psi(\vec{r},E,\hat{\Omega})+\int_g dE\sigma_t(\vec{r},E)\Psi(\vec{r},E,\hat{\Omega})\\&=\sum_{g'=1}^{G}\int_g dE\int_{g'} dE'\int_{4\pi} d\Omega'\sigma_s(\vec{r},E'\to E,\mu_o)\Psi(\vec{r},E',\Omega')\\&\qquad+\int_g dE\ q_{ex}(\vec{r},E,\hat{\Omega})\end{aligned}\tag{1}$$

In Eq 1, Ψ is flux; σ_t and σ_s are total and scattering cross sections, respectively; μ_o is the direction cosine between incident and emerging particle; and q_{ex} is the external source. Equation 1 is rewritten by preserving reaction rates as

$$\begin{aligned}&\hat{\Omega}\cdot\nabla\Psi_g(\vec{r},\hat{\Omega})+\sigma_{t,g}(\vec{r},\hat{\Omega})\Psi_g(\vec{r},\hat{\Omega})\\&=\sum_{g'=1}^{G}d\Omega'\ \sigma_{s,g'\to g}(\vec{r},\mu_o,\hat{\Omega}')\Psi_{g'}(\vec{r},\hat{\Omega}')+q_{ex,g}(\vec{r},\hat{\Omega})\end{aligned}\tag{2}$$

where the group flux is defined as

$$\Psi_g(\vec{r},\hat{\Omega})=\int_g dE\Psi(\vec{r},E,\hat{\Omega})\tag{3}$$

The total and scattering group constants are defined by Eqs 4 and 5, respectively, as

$$\sigma_{t,g}(\vec{r},\hat{\Omega})=\frac{\int_g dE\sigma_t(\vec{r},E)\Psi(\vec{r},E,\hat{\Omega})}{\int_g dE\Psi(\vec{r},E,\hat{\Omega})}\tag{4}$$

and

$$\sigma_{s,g'\to g}(\vec{r},\mu_o,\hat{\Omega}') = \frac{\int_g dE \int_{g'} dE'\ \sigma_s(\vec{r},E'\to E,\mu_o)\Psi(\vec{r},E',\hat{\Omega}')}{\int_{g'} dE'\ \Psi(\vec{r},E',\Omega')} \tag{5}$$

Finally, the external group source is defined as

$$q_{ex,g}(\vec{r},\hat{\Omega}) = \int_g dE\ q_{ex}(\vec{r},E,\hat{\Omega}) \tag{6}$$

The total cross section formulated in Eq 4 has an additional angular dependency after energy discretization is introduced. Furthermore, if the scattering multigroup cross section is examined, it is seen that there is dependency on both the angle $\hat{\Omega}'$ and the scattering cosine μ_o. Note that the angular flux introduces additional angular dependency on the scattering cross section. The two techniques used to eliminate these additional dependencies in the multigroup cross sections are the scalar flux collapsing and bi-linear adjoint weighting approaches.

Scalar Flux Collapsing

The common procedure to collapse cross sections is to use scalar flux as the weighting function. The scalar flux is given as

$$\Phi(E) = \int_{D_s} dr \int_{4\pi} d\Omega\Psi(\vec{r},E,\hat{\Omega}) \tag{7}$$

where D_s is the sub-domain. The total and scattering group constants weighted with the scalar flux are given as

$$\sigma_{t,g} = \frac{\int_g dE\sigma_t(E)\int_{D_s} dr \int_{4\pi} d\Omega\Psi(\vec{r},E,\hat{\Omega})}{\int_g dE \int_{D_s} dr \int_{4\pi} d\Omega\Psi(\vec{r},E,\hat{\Omega})} \tag{8}$$

and

$$\sigma_{s,g'\to g}(\mu_o) = \frac{\int_g dE \int_{g'} dE'\sigma_s(E'\to E,\mu_o)\int_{D_s} dr \int_{4\pi} d\Omega\Psi(\vec{r},E',\hat{\Omega}')}{\int_{g'} dE' \int_{D_s} dr \int_{4\pi} d\Omega\ \Psi(\vec{r},E',\Omega')} \tag{9}$$

Since the scattering group constant is dependent on angle, it is common to expand the differential scattering cross section in terms of Legendre polynomials.

$$\sigma_s(E' \to E, \mu_o) = \sum_{l=0}^{\infty} \frac{2l+1}{4\pi} \sigma_{s,l}(E' \to E)\, P_l(\mu_o) \tag{10}$$

where

$$\sigma_{s,l}(E' \to E) = 2\pi \int_{-1}^{1} d\mu_o\ \sigma_s(E' \to E, \mu_o)\, P_l(\mu_o) \tag{11}$$

Therefore, the multigroup scattering cross section becomes

$$\sigma_{s,g' \to g}(\mu_o) = \sum_{l=0}^{\infty} \frac{2l+1}{4\pi}\ P_l(\mu_o)\, \sigma_{s,l,g' \to g} \tag{12}$$

and

$$\sigma_{s,l,g' \to g}(\vec{r}) = \frac{\int_g dE \int_{g'} dE' \sigma_{s,l}(\vec{r}, E' \to E)\Phi(E')}{\int_{g'} dE' \Phi(E')} \tag{13}$$

The scalar flux approximation is equal to the exact expressions given in Eqs 4 and 5 if the angular flux is isotropic. Although this method is commonly used, it has some limitations. Scalar flux weighting does not account for angular variations, and all scattering moments are weighted with the scalar flux. The bi-linear adjoint weighting technique overcomes this deficiency by using the adjoint function.

The Bi-linear Adjoint Weighting Technique

The bi-linear adjoint weighting technique was developed by Haghighat and Hanshaw. This method includes angular dependencies of a problem in multigroup constants by using the forward flux and adjoint function moments to weight cross sections. Weighting with angular dependencies is performed according to important particles that contribute to an objective. The bi-linear adjoint weighting formulation is derived by multiplying the numerator and denominator of Eqs 4 and 5 with the adjoint function and integrating over angle and space. The total collision term is given by

$$\sigma_{t,g} = \frac{\int_g dE \sigma_t(E) \int_{D_s} dr \int_{4\pi} d\Omega \Psi(\vec{r}, E, \hat{\Omega}) \Psi_g^+(\vec{r}, \hat{\Omega})}{\int_g dE \int_{D_s} dr \int_{4\pi} d\Omega \Psi(\vec{r}, E, \hat{\Omega}) \Psi_g^+(\vec{r}, \hat{\Omega})} \tag{14}$$

where

$$\Psi_g^+(\vec{r},\hat{\Omega}) = \int_g dE \Psi^+(\vec{r},E,\hat{\Omega}) \tag{15}$$

If flux expansions are inserted and the orthogonality property is used, Eq 14 reduces to

$$\sigma_{t,g} = \frac{\int_g dE \sigma_t(E) \int_{D_s} dr \left(\sum_{l=0}^{\infty} \frac{2l+1}{4\pi} \sum_{m=-l}^{l} \Psi_l^m(\vec{r},E) \Psi_{l,g}^{m,+}(\vec{r}) \right)}{\int_g dE \int_{D_s} dr \left(\sum_{l=0}^{\infty} \frac{2l+1}{4\pi} \sum_{m=-l}^{l} \Psi_l^m(\vec{r},E) \Psi_{l,g}^{m,+}(r) \right)} \tag{16}$$

Accordingly, the scattering cross section is formulated by

$$\sigma_{s,l,g^{/}\to g} = \frac{\int_g dE \int_{g^{/}} dE^{/} \sigma_{s,l}(E^{/} \to E) \int_{D_s} dr \left(\sum_{m=-l}^{l} \Psi_l^m(\vec{r},E) \Psi_{l,g}^{m,+}(\vec{r}) \right)}{\int_j dE^{/} \int_{D_s} dr \left(\sum_{m=-l}^{l} \Psi_l^m(\vec{r},E) \Psi_{l,g}^{m,+}(\vec{r}) \right)} \tag{17}$$

The bi-linear adjoint weighting technique is superior compared with the scalar flux weighting approach, especially for problems that have high angular dependencies (e.g., fast neutron dosimetry).

Energy Group Structure Construction Method of CPXSD

The CPXSD methodology constructs energy group structures by refining an arbitrary group structure. Refinement of the arbitrary group structure is performed according to the importance of groups in the group structure, and point-wise cross sections of an important isotope/material in the problem.

The importance of groups can be determined using the "response flux" (i.e., the contributon formulation) [*6*]:

$$C(E) = \int_V dr \int_{4\pi} d\Omega \Psi(\vec{r},E,\hat{\Omega}) \Psi^+(\vec{r},E,\hat{\Omega}) \tag{18}$$

Considering spherical harmonics expansion of the flux and adjoint function, and using orthogonality, the group-dependent response flux is written as

$$C_g = \sum_{s\in D} V_s \sum_{l=0}^{L} \sum_{m=0}^{l} \frac{2l+1}{4\pi} \Psi_{l,g,s}^m \Psi_{l,g,s}^{m,+} \tag{19}$$

where g refers to energy group, s refers to a materially uniform sub-domain in domain D; and l and m refer to polar and azimuthal indices, respectively. The group-dependent response flux formulation given in Eq 19 represents the importance of particles in group g for an objective.

There is similarity between contributon and reaction rate formulations. The response at the cavity dosimeter can be written as

$$response = \langle \sigma_d \Psi \rangle = \langle q_{ex} \Psi^+ \rangle \tag{20}$$

If the angular flux is substituted for the external source in Eq 20, it is possible to obtain an expression similar to that for the response flux.

The second item to determine in CPXSD is an important isotope in the problem of interest. For this study ^{56}Fe is chosen since this isotope plays an important role in the pressure vessel. The ^{56}Fe point-wise group structure is used to identify group boundaries in constructing new group structures.

The generation of a fine-group structure with CPXSD has been previously published [*3*,*4*], and this paper focuses on broad-group structure construction. Starting from a fine-group library, the construction of broad-group libraries for the purpose of calculating effective reaction rates at the cavity dosimeter of a reactor model is described as follows.

(i) An initial broad-group structure is selected. For fast neutron dosimetry calculations, the initial broad-group structure may be selected by considering energy thresholds of the dosimetry isotopes.

(ii) Fine-group cross sections are collapsed to the initial broad group, and a transport calculation is performed with the broad-group library to calculate the objectives.

(iii) The group that has the highest importance in the problem is identified, and the number of sub-groups to be added in this group is specified. Resonance and non-resonance parts of the point-wise cross sections of the important isotope are determined within this group, and their areas are calculated. The resonance that has the largest area is selected, and one sub-group is placed that encloses the resonance. Remaining resonance and non-resonance parts are combined such that they approach the size of the largest resonance area. Based on a user-input tolerance, the area of the largest resonance is modified until the number of sub-groups formed matches the number of sub-groups requested. If a resonance structure does not exist, the group is simply partitioned into even energy intervals.

(iv) The fine-group library is collapsed into the new broad-group library, and steps (ii) and (iii) are repeated until a user-specified convergence criterion is achieved on the calculated objectives.

Construction of Broad-Group Structures for Shielding Applications using the Three Mile Island-1 (TMI-1) Reactor

In this section, broad-group structures are constructed for pressure vessel dosimetry problems using the one-dimensional TMI-1 reactor. The TMI-1 reactor consists of 27 nuclides including, ^{27}Al, ^{10}B, C, Ca, ^{50}Cr, ^{52}Cr, ^{53}Cr, ^{54}Cr, ^{54}Fe, ^{56}Fe, ^{57}Fe, ^{58}Fe, ^{1}H, K, Mg, ^{55}Mn, ^{23}Na, ^{58}Ni, ^{60}Ni, ^{61}Ni, ^{62}Ni, ^{64}Ni, ^{16}O, Si, Zr, ^{235}U, and ^{238}U.

In generating an effective fine-group library for the TMI-1 reactor, the NJOY code [*7*] was used for cross-section processing. NJOY version 94.15 was used for ^{27}Al and ^{235}U, and NJOY version 91.94m was used for the remaining isotopes. ENDF/B-VI release 3 was used for ^{235}U, and ENDF/B-VI release 2 was used for the remaining isotopes. The choice of the NJOY versions and ENDF data for the 27 isotopes was made considering the NJOY versions and ENDF data

that were used in generating the 47-neutron-group BUGLE-96 library [*5*]. Note that the BUGLE-96 library uses ENDF/B-VI release 3 data for ^{27}Al; however, we used ENDF/B-VI release 2 for this isotope because of the difficulty we encountered in processing the release 3 data file.

The AMPX modules [*8*] were used for the post-processing of cross sections. The modules used from the AMPX code package and their functions in the cross-section library generation procedure for this study is listed in Table 1. In the BONAMI calculations, water was not included in the self-shielding treatment of iron in the pressure vessel. The one-dimensional option of the two-dimensional DORT S_N code [*9*] was used for neutron transport calculations. For these calculations, a P_3 scattering order and a S_8 quadrature set were used.

TABLE 1— Modules that are used in the generation of fine-group libraries.

Module Name	Function
SMILER	Converts NJOY Group-wise Evaluated Nuclear Data Files (GENDF) to AMPX master library format
AJAX	Gathers AMPX master libraries
BONAMI	Performs self-shielding calculations
NITAWL	Converts AMPX master libraries to AMPX working library format (the code may also be utilized to use the Nordheim Integral method for resonance self-shielding)

For fast neutron dosimetry calculations, six reactions that are sensitive to energies above 0.1 MeV were selected. These are ^{63}Cu(n,α), ^{54}Fe(n,p), ^{58}Ni(n,p), ^{46}Ti(n,p), ^{237}Np(n,f), and ^{238}U(n,f). The normalized cross sections of these dosimetry isotopes and their sum are shown in Fig. 1.

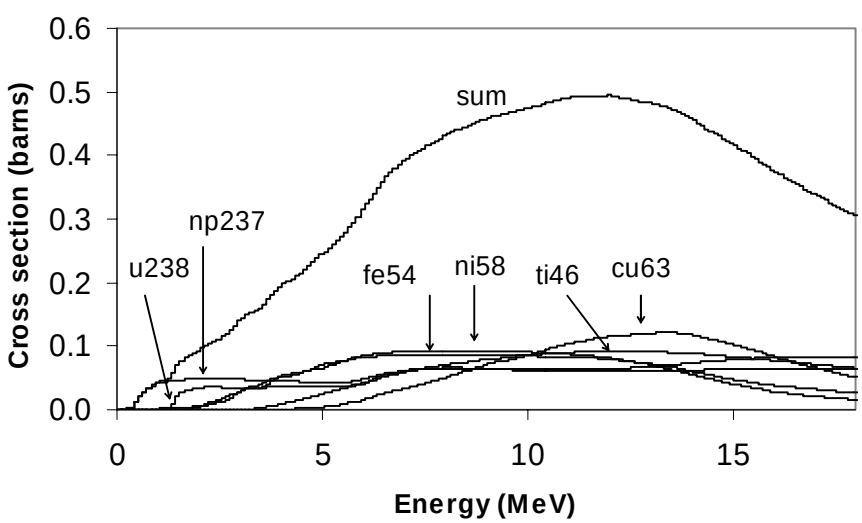

FIG. 1— *Normalized dosimetry cross sections and their sum.*

Broad-group structures were constructed from the 450-group LIB450 library [*2,3*]. LIB450 was generated from a fine-group structure constructed by CPXSD for fast neutron dosimetry problems. For broad-group structure construction, initial broad-group boundaries were selected based on the threshold energies of the six dosimeter isotopes. The initial broad-group structure boundaries were selected as follows: 18.032, 3.6815, 2.9737, 1.0196, 0.51487, and 0.11109MeV. Except for the highest energy value, the remaining energies correspond to the threshold energies of the following interactions: ^{63}Cu(n,α), ^{46}Ti(n,p), ^{54}Fe(n,p), ^{58}Ni(n,p) and ^{238}U(n,f), and ^{237}Np(n,f), respectively.

A broad-group structure was generated for the isotope that has the highest threshold first [i.e., for ^{63}Cu(n,α)], which is sensitive to energies above 3.6815 MeV. After a group structure was obtained above 3.6815 MeV, the isotope with the next highest threshold energy [i.e., ^{46}Ti(n,p)] was considered. We proceeded with this procedure for the remaining dosimetry isotopes. When refining the initial broad-group structure, the number of groups was increased one at a time, until a relative difference of 1% was achieved between fine- and broad-group reaction rates. By performing this procedure, 16- and 15-group libraries were generated considering the scalar flux and bi-linear adjoint weighting techniques, respectively. The 16-group LIB16 and the 15-group LIB15 libraries contain only one group below 0.1 MeV. Table 2 gives the group boundaries for the LIB16 and LIB15 libraries. Comparing the group structures of the two libraries, it is evident that for the bi-linear adjoint weighting technique, fewer groups are placed in very high energies where scattering is highly directional. For example, for the ^{63}Cu(n,α) reaction, placing seven groups using the scalar flux collapsing and two groups using the bi-linear adjoint weighting above 3.69 MeV results in comparable reaction rates. This means, as expected, the bi-linear weighting effectively includes the angular behavior of the differential scattering cross section.

TABLE 2—Group structures generated with the scalar flux and bi-linear adjoint technique.

Energy Group Number	Upper Energy (MeV)	
	Scalar Flux Weighting, LIB16	Bi-linear Adjoint Weighting, LIB15
1	18.032	18.032
2	14.832	8.1320
3	11.632	3.6815
4	8.6320	2.9737
5	7.1320	2.6515
6	5.9553	2.3320
7	4.8320	2.0000
8	3.6815	1.7016
9	2.9737	1.4053
10	2.4726	1.0196
11	1.9734	0.82706
12	1.5320	0.70218
13	1.0196	0.51487
14	0.77260	0.32168
15	0.51487	0.11109
	0.11109	

Table 3 gives the broad- (LIB16 and LIB15) to fine-group (LIB450) reaction rate ratios. The reaction rate ratios of the broad- to fine-group library calculations are ~1% different. The results indicate that the broad-group libraries compare closely with their respective fine-group libraries.

TABLE—3 Broad- (LIB16 and LIB15) to fine-group (LIB450) reaction rate ratios

Reaction	LIB16[a]/LIB450	LIB15[b]/LIB450
^{63}Cu(n,α)	0.9882	1.006
^{54}Fe(n,p)	0.9884	0.9924
^{58}Ni(n,p)	0.9888	0.9946
^{46}Ti(n,p)	0.9896	1.005
^{237}Np(n,f)	0.9996	1.012
^{238}U(n,f)	0.9992	1.003

[a] Scalar flux weighting.
[b] Bi-linear adjoint weighting.

Broad-Group and Continuous-Energy Comparisons

In this section, LIB15, LIB16, and BUGLE-96 libraries are compared with continuous-energy calculations that are performed using A^{3}MCNP (Automated Adjoint Accelerated MCNP) [*10,11*]. The A^{3}MCNP code is a version of MCNP with automated variance-reduction capability. In the A^{3}MCNP calculations, fluxes were tallied within energy bins of the broad-group libraries. A^{3}MCNP flux tallies were calculated to have statistical errors below ~1%.

Table 4 gives the reaction rate ratios calculated by DORT (broad group) to A^{3}MCNP (continuous energy). The CPXSD libraries are about the same as, or in better agreement with, continuous-energy predictions as compared with BUGLE-96. The differences in LIB15 and LIB16 as compared with continuous energy are attributed to the 1% convergence criteria used on reaction rates, when constructing group structures in CPXSD.

TABLE—4 Broad-group to continuous-energy (CE) reaction rate ratios.

Reaction	LIB16/CE	LIB15/CE	BUGLE-96/CE
^{63}Cu(n,α)	0.9301	0.9056	0.9111
^{54}Fe(n,p)	0.9196	0.9152	0.9006
^{58}Ni(n,p)	0.9167	0.9168	0.9000
^{46}Ti(n,p)	0.9255	0.9059	0.9074
^{237}Np(n,f)	0.9052	0.9380	0.8984
^{238}U(n,f)	0.9063	0.9219	0.8990

Summary and Conclusions

This paper presents the broad-group structure construction procedure used in CPXSD and compares the new broad-group libraries with their fine-group libraries as well as with continuous-energy cross sections. For fast neutron dosimetry calculations, broad- to fine-group reaction rate ratios are ~1% for CPXSD libraries. Compared to BUGLE-96, calculated reaction rates by CPXSD libraries are about the same as or in better agreement with the continuous-energy predictions. Furthermore, CPXSD-constructed group structures have almost half the number of groups of the BUGLE group structure above 0.1 MeV that results in significant computational time.

References

[1] *ANSI/ANS-6.1.2*-1999, "Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations for Nuclear Power Plants," 1999.

[2] Alpan, F. A., "An Advanced Methodology for Generating Multigroup Cross-Sections for Shielding Calculations," Ph. D. Thesis, Nuclear Engineering, The Pennsylvania State University, May 2003.

[3] Alpan, F. A. and Haghighat, A., "Development of the CPXSD Methodology for Generation of Fine-Group Libraries for Shielding Applications," *Nuclear Science and Engineering*, 149, January 2005, pp. 38-50.

[4] Hanshaw, H. L., "Multigroup Cross Section Generation with Spatial and Angular Adjoint Weighting," M.S. Thesis, Nuclear Engineering, The Pennsylvania State University, August 1995.

[5] "BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," DLC-76, Oak Ridge National Laboratory, Oak Ridge, Tennessee, March 1996.

[6] Williams, M. L., "Generalized Contributon Response Theory," *Nuclear Science and Engineering*, 108, 1991, pp. 355-383.

[7] "NJOY94.61: Code System for Producing Pointwise and Multigroup Neutron and Photon Cross Sections from ENDF/B Data," PSR-355, Los Alamos National Laboratory, Los Alamos, New Mexico, December 1996.

[8] "AMPX-77: Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," PSR-315, Oak Ridge National Laboratory, Oak Ridge, Tennessee, October 1992.

[9] "TORT-DORT-PC, Two and Three Dimensional Discrete Ordinates Transport Version 2.7.3," CCC-543, Oak Ridge National Laboratory, Oak Ridge, Tennessee, June 1996.

[10] Wagner, J. C., "Acceleration of Monte Carlo Shielding Calculations with an Automated Variance Reduction Technique and Parallel Processing," Ph.D. Thesis, Nuclear Engineering, The Pennsylvania State University, 1997.

[11] Wagner, J. C. and Haghighat, A., "Automatic Variance Reduction of Monte Carlo Shielding Calculations Using the Discrete Ordinates Adjoint Function," *Nuclear Science and Engineering*, 128, 1998, pp. 186-208.