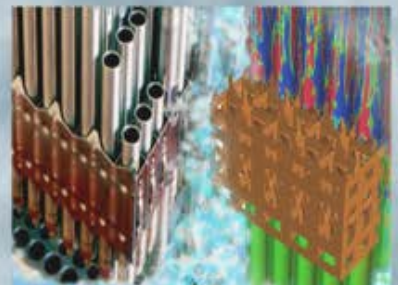
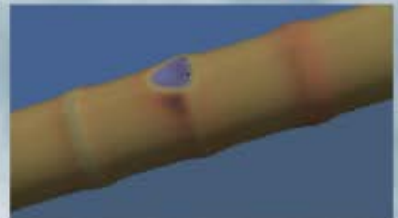
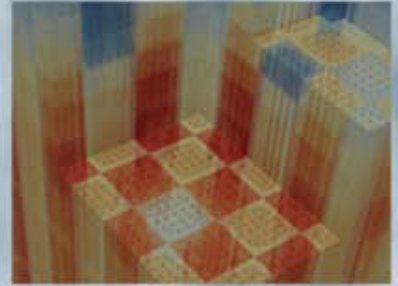


# Procedure to Generate the MPACT Multigroup Library

**Revision 0**  
**December 17, 2015**

**Kang Seog Kim**

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## **EXECUTIVE SUMMARY**

The CASL neutronics simulator MPACT is under development for the neutronics and T-H coupled simulation for the light water reactor. The objective of this document is focused on reviewing the current procedure to generate the MPACT multigroup library. Detailed methodologies and procedures are included in this document for further discussion to improve the MPACT multigroup library.

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## ACRONYMS

|      |                                                            |
|------|------------------------------------------------------------|
| 0D   | Zero-Dimensional (Homogeneous)                             |
| 1D   | One-Dimensional                                            |
| 2D   | Two-Dimensional                                            |
| CASL | Consortium for Advanced Simulation of Light Water Reactors |
| CE   | Continuous Energy (as in cross sections)                   |
| ESSM | Embedded Self-Shielding Method                             |
| IR   | Intermediate Resonance                                     |
| MG   | Multi-group (as in cross sections)                         |
| MOC  | Method of Characteristics                                  |
| NR   | Marrow Resonance                                           |
| ORNL | Oak Ridge National Laboratory                              |
| PW   | Pointwise                                                  |
| PWR  | Pressurized Water Reactor                                  |
| RI   | Resonance Integral                                         |
| URR  | Unresolved resonance                                       |
| WR   | Wide Resonance                                             |





## 1. INTRODUCTION

The main neutronic simulator in the Consortium for Advanced Simulation of Light Water Reactors (CASL) is the MPACT code which is being developed collaboratively by Oak Ridge National Laboratory and University of Michigan. The key characteristics of the MPACT code includes the subgroup method [Hel03] and the embedded self-shielding method (ESSM) [Wil12] for resonance self-shielding calculation, and a whole core solver with a 2D/1D synthesis method on the frame of 3D coarse mesh finite difference (CMFD) method for which axial and radial correction factors are obtained from 1D nodal expansion method (NEM) or Simplified  $P_N$  and 2D method of characteristics (MOC), respectively. The multigroup (MG) cross section library for MPACT has been generated with significant extensions of the AMPX [Wia14] and SCALE [Sca09] code packages developed at Oak Ridge National Laboratory for which the AMPX code package has been significantly improved especially for the Bondarenko approach of resonance self-shielding calculation such as the subgroup method and ESSM used in MPACT.

The objective of this document is focused on reviewing the current procedure to generate the MPACT MG library. Detailed methodologies and procedures for the generation of the MPACT MG library are included in this document for further discussion to improve the library. This document includes the followings.

- MG cross section processing by AMPX (Chapter 2)
- Improvement of resonance data (Chapter 3)
- Resonance self-shielding methods (Chapter 4)
- Generation of miscellaneous data (Chapter 5)
- Generation of the MPACT MG library (Chapter 6)
- Challenging technical issues (Chapter 7)

## 2. MULTIGROUP CROSS SECTION PROCESSING BY AMPX

### 2.1 Conventional AMPX procedure

AMPX-6 [Wia14] is a modular system of FORTRAN computer programs to generate the MG and continuous energy (CE) cross section libraries for modern deterministic and Monte Carlo transport codes by processing the ENDF/B libraries. Since the CASL neutronics simulator MPACT is a deterministic transport code, only the AMPX MG library generation procedure is discussed.

The AMPX MG library includes various neutron reactions of the Bondarenko F-factors as a function of background cross section for all energy groups including resolved and unresolved resonances. The resolved resonance F-factors have been generated by the narrow resonance (NR) approximation and the unresolved ones by the J-function method [Lee11] based on the NR approximation. And MG 1D and 2D cross section data have been obtained by using a weighting function of Maxwellian spectrum +  $1/E$  + fission spectrum. At low energies, the weighting function is a Maxwellian spectrum which has a flux shape that assumes the neutron scatters into a region with a free gas scatterer that has no absorption. The Maxwellian flux spectrum has the following formula.

$$\phi(E) = M(E) = Ee^{-\frac{E}{kT}}, \quad (2.1.1)$$

where  $E$ ,  $k$ ,  $T$  denote neutron energy, Boltzmann constant and temperature of the material in Kelvin, respectively. In the slowing down range,  $0.125 \text{ eV} < E < E_{cut}$ , the weighting spectrum is assumed to be  $\phi(E) = 1/E$ . The cutoff energy  $E_{cut}$  for the slowing down range must be selected and is typically 55 keV by default in the AMPX modules. In the region,  $E_{cut} < E < 10^7 \text{ eV}$ , where fission neutrons are born, the following fission spectrum is used.

$$\phi(E) = \chi(E) = E^{0.5} e^{-\frac{E}{\theta}}, \quad (2.1.2)$$

where  $\theta$  is temperature of the fission spectrum (e.g.,  $1.2 \times 10^6 \text{ eV}$ ). For energies above  $10^7 \text{ eV}$ , the particles are considered to be in another slowing down region; hence, the spectrum is assumed to have a  $1/E$  shape.

MG self-shielded cross sections of reaction type  $i$  (Bondarenko F-factors) as a function of background cross sections ( $\sigma_0$ ) and material temperature (T) are calculated by using the following equation based on narrow resonance approximation.

$$\sigma_{i,g}(T, \sigma_0) = \frac{\int_g \frac{\sigma_i(T, E) \sigma_0 \phi(E)}{\sigma_i(T, E) + \sigma_0} dE}{\int_g \frac{\sigma_0 \phi(E)}{\sigma_i(T, E) + \sigma_0} dE}, \quad (2.1.3)$$

where  $T$  denotes temperature and  $\sigma_t$  is total cross section. The MG scattering matrix can be obtained by

$$\sigma_{s,l,gg'} = \frac{1}{\int_g \phi(E) dE} \int_g y(E) \sigma_s(E) \phi(E) dE \int_{g'} f_l(E, E') dE', \quad (2.1.4)$$

where  $y(E)$  is a multiplicity which is unity for scattering and  $f_l(E, E')$  a normalized double differential distribution. It should be noted that the 2-D scattering data (not for 2-D thermal scattering) are temperature independent in the AMPX MG library.

## 2.2 Practical weighting function

In the standard SCALE sequences (e.g. TRITON), self-shielded MG 1D and 2D data for resolved resonance and thermal energy groups are determined by performing the problem-dependent CENTRM slowing down transport calculation for each pin cell type with eq. (2.2.1), and thus the resolved resonance data in the AMPX MG library are not used.

$$\hat{\Omega} \cdot \nabla \psi + \Sigma_t(\vec{r}, u) \psi(\vec{r}, u, \hat{\Omega}) = \int_{4\pi} d\Omega' \int_0^\infty du' \Sigma_s(\vec{r}, u' \rightarrow u, \hat{\Omega}' \cdot \hat{\Omega}) \psi(\vec{r}, u', \hat{\Omega}') + q(\vec{r}, u, \hat{\Omega}), \quad (2.1.1)$$

where

- $\hat{\Omega}$  : neutron direction
- $\vec{r}$  : space coordinate
- $u$  : neutron lethargy
- $\psi$  : angular flux
- $\Sigma_t$  : macroscopic total cross section
- $\Sigma_s$  : double differential scattering cross section
- $q$  : external source.

However, since Bondarenko approach for resonance self-shielding calculation does not include any problem-dependent pointwise slowing down calculation, weighting function would make a significant impact on the accuracy mostly through 2-D scattering matrix. The weighting function of Maxwellian spectrum + 1/E + fission spectrum is far from a realistic weighting function. Practical weighting functions for various temperatures can be obtained by performing the CENTRM MG/PW calculations for a typical PWR fuel pin. Figure 2.2.1 provides a sample of pointwise neutron spectra obtained by CENTRM. This would be performed using the XSProc sequence with “parm=centrm”. The weight functions need to be generated to include various temperatures.

There are several pending issues associated with the pointwise weighting function. Since resonance self-shielded cross sections are independently estimated, 1D cross sections are independent upon the pointwise weighting function. However, in reality since the current MPACT procedure includes resonance data only for the specified energy groups, 1D cross

sections of non-resonance groups should be dependent on the weighting functions. MG scattering matrices are obtained by simple flux weighting and renormalization is applied only to total scattering, within-group and out-scattering components. Therefore, this procedure may introduce wrong neutron fluxes resulting in some errors in reaction rates. Pointwise neutron spectra are very changeable according to  $^{235}\text{U}$  enrichment, moderator-to-fuel ratio, burnup and void fraction. Therefore, selection of weighting function would be another challenging issue.

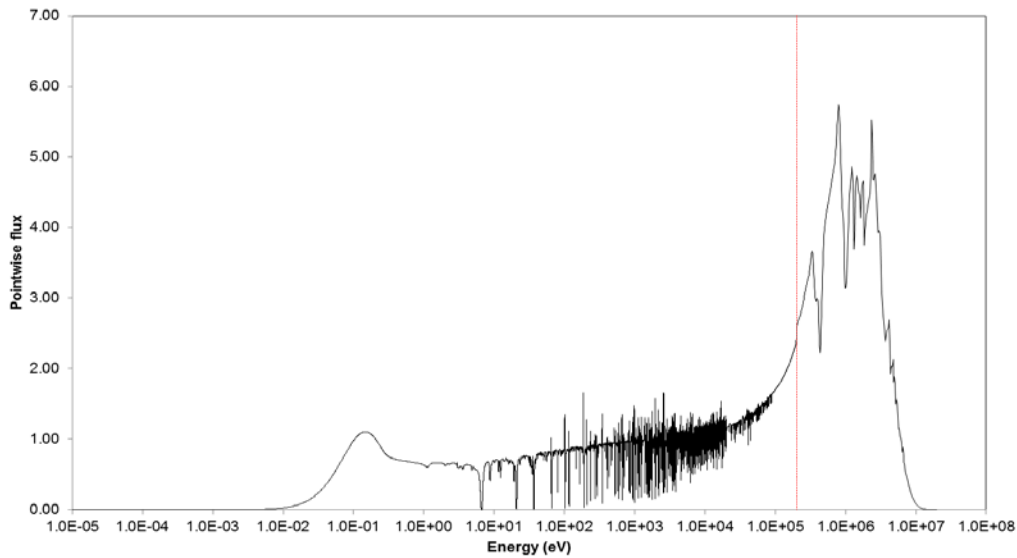


Figure 2.2.1 Pointwise neutron spectra

### 3. GENERATION OF IMPROVED RESONANCE DATA

Although the AMPX MG library includes MG resonance self-shielded cross sections (Bondarenko F-factors) as a function of background cross sections generated by eq. (2.1.3), they cannot guarantee accurate result due to significant approximations such as coarse energy group, weighting functions and resonance interferences. Therefore, the MG resonance self-shielded data need to be improved in accordance with resonance self-shielding method. This chapter describes several methods to improve the resonance self-shielded cross section data.

#### 3.1 Slowing down equation

The time-independent Boltzmann equation is as follows:

$$\hat{\Omega} \cdot \nabla \psi + \Sigma_t(\vec{r}, u) \psi(\vec{r}, u, \hat{\Omega}) = \int_{4\pi} d\Omega' \int_0^\infty du \Sigma_s(\vec{r}, u' \rightarrow u, \hat{\Omega}' \cdot \hat{\Omega}) \psi(\vec{r}, u', \hat{\Omega}') + q(\vec{r}, u, \hat{\Omega}), \quad (3.1.1)$$

where

- $\hat{\Omega}$  : neutron direction
- $\vec{r}$  : space coordinate
- $u$  : neutron lethargy
- $\psi$  : angular flux
- $\Sigma_t$  : macroscopic total cross section
- $\Sigma_s$  : double differential scattering cross section
- $q$  : external source.

The continuous scalar flux can be obtained by solving eq. (3.1.1) using the point-wise cross sections. The effective group self-shielded cross section of the resonance isotope can be obtained using the equation (3.1.2):

$$\sigma_{x,g} = \frac{\int \sigma_x(u) \phi(u) du}{\int_{\Delta u_g} \phi(u) du}. \quad (3.1.2)$$

In the SCALE code package CENTRM and PMC perform the above calculations to obtain the group self-shielded cross sections for various geometrical models including infinite medium, slab, 1D cylinder and sphere and 2D square pin.

A flowchart of the SCALE modules executed for self-shielding is shown in Figure 3.1.1. First, the fast-executing Bondarenko method is applied to all energy groups for the total, elastic, capture, and fission cross sections of all materials by using BONAMI. In this method MG data at problem-specific conditions are interpolated from the tabulated self-shielding factors in the MG library. While the analytical flux expression used to process Bondarenko shielding factors is reasonable for higher energy groups, it tends to break down at lower energies. SCALE 6

provides other approaches to generate problem-dependent MG data directly from PW data within the energy range in which the Bondarenko method is not desirable - typically the resolved resonance and thermal ranges for criticality safety and reactor physics applications. As shown in Figure 3.1.2, the energy range of interest is divided into the upper multigroup (UMR), PW, and lower multigroup (LMR) ranges, as defined by the input energy boundaries  $E_{max}$  and  $E_{min}$ , which have default values of 20 keV and  $10^{-3}$  eV, respectively, in SCALE 6. Generally it is recommended that the PW energy range should encompass the resolved resonance ranges of all nuclides that impact the spectral fine-structure; however MG data in the UMR and LMR are always self-shielded with the Bondarenko method. Thus it is not necessary to perform a PW calculation within the energy range where the Bondarenko approximation is adequate. This is done by the CENTRM code, which computes neutron spectra on a fine energy mesh within the PW energy range using various approximations to the Boltzmann equation. In the energy region outside the specified PW range, CENTRM performs an MG calculation so that a full energy range solution is obtained.

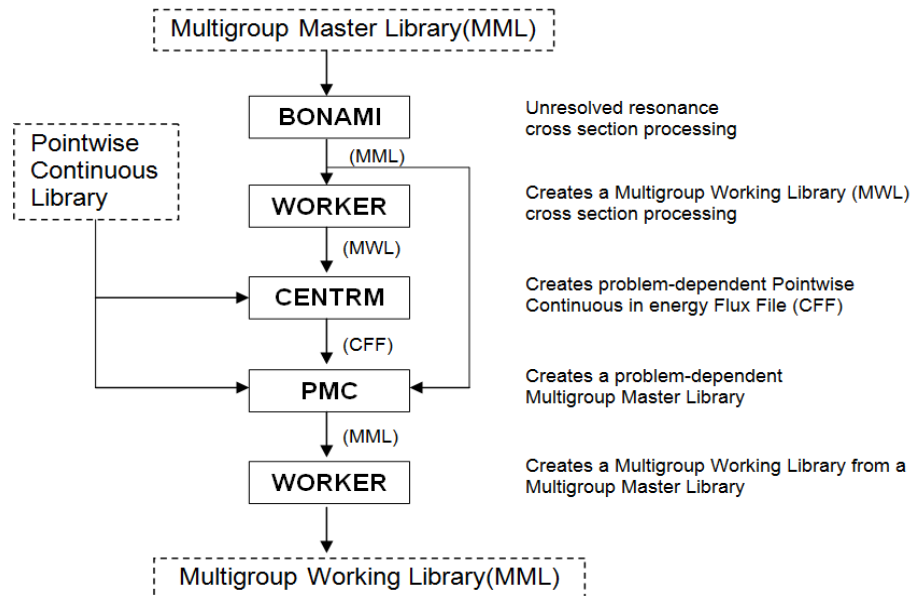


Figure 3.1.1 Flowchart of modules used to generate problem-dependent MG data

The PMC module computes problem-specific MG data for groups within the PW energy range of the CENTRM calculation and replaces the more approximate values from BONAMI. PMC uses the CENTRM spectra for each spatial region as a problem-dependent weight function for averaging energy-dependent data in the PW libraries into MG data. The resulting problem-specific cross-section library is passed to higher-dimensional MG transport calculations performed with other SCALE modules.

Exact same procedure can be utilized in improving MG resonance data in the AMPX MG library. The only difference is to additionally introduce the following assumptions to exclude resonance interference effect according to a user option.

- a. Only one resonant nuclide is assumed in the mixture with a constant potential scattering cross section ( $\sigma_p$ ), a resonance scattering cross section ( $\sigma_{rs}(u)$ ) and a resonance absorption cross section ( $\sigma_{ra}(u)$ ).
- b. The non-resonant nuclides are considered to have negligible absorption and constant potential scattering cross sections.

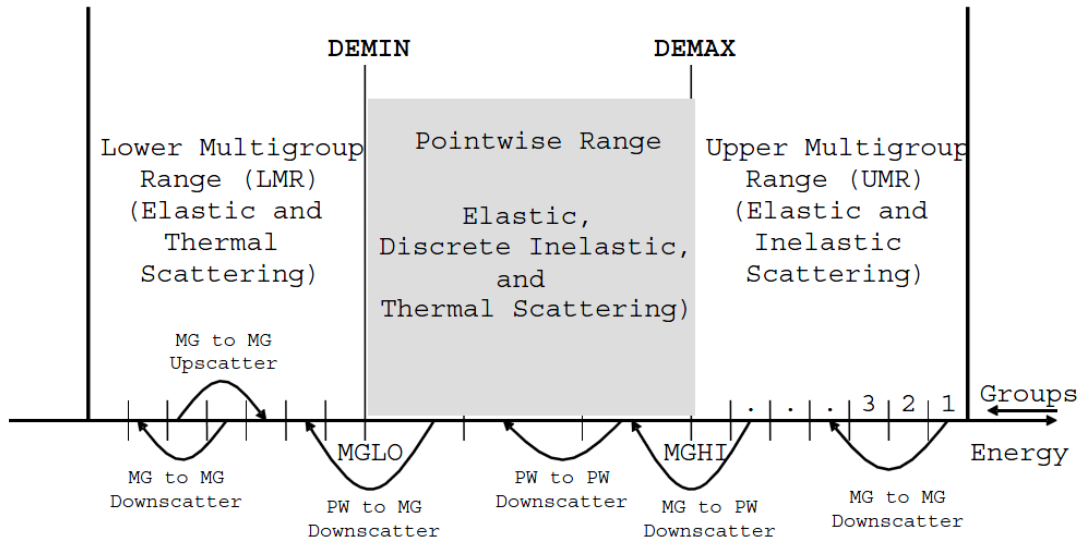


Figure 3.1.2 Definition of UMR, PW, and LMR energy ranges

Exact same procedure by using CENTRM can be utilized in improving resonance data. Several methods are available to improve resonance data which include homogeneous or heterogeneous models, and multiple absorber model to pre-consider resonance interferences or single absorber model to exclude resonance interference. A new procedure using I-CRAWDIUS has been developed to provide potential cross sections or pointwise cross sections to CENTRM for the background nuclides to exclude or include resonance interference between resonance nuclides according to user option.

### 3.2 Intermediate resonance (IR) approximation

To cast the scattering source term in eq. (3.1.6) into a more tractable form, the intermediate resonance (IR) approximation is introduced.

In this approximation, the fraction  $\lambda_i$  of the scattering for each isotope  $i$  is assumed to be so effective that the maximum lethargy gain per collision is significantly greater than the practical resonance width (i.e., the narrow resonance scattering)[Hel03, Sta83]. A resonance affects only a small interval of the integration range from  $u-\Delta_i$  to  $u$ , and its contribution to the scattering source  $Q(u)$  is negligible. Since outside the resonances  $\Sigma_{si}(u)=\Sigma_{pi}$  and  $\phi(u)=constant$ , the contribution

of isotope  $i$  to  $Q(u)$  is  $\lambda_i \Sigma_{pi}$ . This is the unperturbed slowing down source. Conversely, the remaining fraction  $(1-\lambda_i)$  is assumed to be so ineffective that neutrons gain a negligible amount of lethargy compared with  $\Delta u_g$  (i.e., the wide resonance scattering). The resonances are so wide that the integrand can be replaced by its average value, which leads to the  $(1-\lambda_i) \Sigma_{si}(u) \phi(u)$  contribution to the source  $Q(u)$ . Thus, this fraction of scattering does not provide source neutrons from outside the resonance widths but it should be considered as due to self-scattering which neither adds nor removes neutrons. With these approximations, eq. (3.1.6) can be rewritten for the coarse energy groups:

$$\hat{\Omega} \cdot \nabla \psi_{g,k} + \sum_i \Sigma_{i,g,t}^k \psi_{g,k}(\hat{\Omega}) = \sum_i \lambda_{i,g} \Sigma_{i,p}^k + \sum_i (1-\lambda_{i,g}) \Sigma_{i,g,s}^k \phi_{g,k}. \quad (3.2.1)$$

Eq. (3.2.1) can be rewritten by assuming isotropic angular flux as

$$\hat{\Omega} \cdot \nabla \psi_{g,k} + \sum_i (\Sigma_{i,g,a}^k + \lambda_{i,g} \Sigma_{i,p}^k) \psi_{g,k}(\hat{\Omega}) = \sum_i \lambda_{i,g} \Sigma_{i,p}^k. \quad (3.2.2)$$

For the homogeneous mixture, the self-shielded flux can be obtained from eq. (3.2.2) as

$$\phi_g = \frac{\sum_i \lambda_{i,g} \Sigma_{i,p}}{\sum_{i=\text{resonance}} \Sigma_{i,g,a} + \sum_i \lambda_{i,g} \Sigma_{i,g,s}} = \frac{\lambda_g \Sigma_p}{\Sigma_{g,a} + \lambda_g \Sigma_{g,s}}. \quad (3.2.3)$$

### 3.3 Intermediate resonance parameters

Let us assume there is only one resonance isotope having an atomic number density  $N_R$ . In a homogeneous system, the background cross section of this mixture is defined by

$$\sigma_b = \frac{\sum_{i=\text{all}} N_i \lambda_i \sigma_{i,p}}{N_R}. \quad (3.3.1)$$

Eq. (3.2.1) can be rewritten by using eq. (3.3.1) as

$$\phi_g = \frac{\sigma_{g,b}}{\sigma_{g,a} + \sigma_{g,b} + \lambda_g \sigma_{g,rs}}. \quad (3.3.2)$$

Since the atomic mass of hydrogen ( $^1H$ ) is very close to unity, lethargy gain of any neutron colliding with hydrogen is very big and the neutron can scatter beyond a resonance without any collision. This is essentially the same as the narrow resonance approximation. Therefore, for hydrogen, the intermediate resonance parameter is defined as unity. For other isotopes, the intermediate resonance parameter is obtained by comparing results of various  $^{238}U/^1H$  mixtures



where the hydrogen is partly replaced by the other isotopes. This is often referred to as a hydrogen-equivalence parameter.

For homogeneous mixtures including  $^{238}\text{U}$  and  $^1\text{H}$ , eq. (3.2.1) can be written as follows:

$$(N^{238}\sigma_{g,a}^{238} + N^{238}\sigma_{g,rs}^{238} + N^{238}\lambda_g^{238}\sigma_p^{238} + N^1\lambda_g^1\sigma_p^1)\phi_g = N^{238}\lambda_g^{238}\sigma_p^{238} + N^1\lambda_g^1\sigma_p^1. \quad (3.3.3)$$

In order to obtain the hydrogen-equivalence parameter for a nuclide, a part of hydrogen ( $N^1$ ) is replaced with a certain nuclide ( $N^x$ ). The amount of the hydrogen atomic number density replaced by another isotopic atomic number density can be adjusted to obtain the same  $\sigma_{g,a}^{238}$  from the slowing down calculation by using eq. (3.1.6). The balance equation will be

$$(N^{238}\sigma_{g,a}^{238} + N^{238}\sigma_{g,rs}^{238} + N^{238}\lambda_g^{238}\sigma_p^{238} + (N^1 - N^x)\lambda_g^1\sigma_p^1 + N^x\lambda_g^x\sigma_p^x)\phi_g = N^{238}\lambda_g^{238}\sigma_p^{238} + (N^1 - N^x)\lambda_g^1\sigma_p^1 + N^x\lambda_g^x\sigma_p^x. \quad (3.3.4)$$

By comparing these two equations, the unknown intermediate resonance parameter ( $\lambda_g^x$ ) can be obtained as follows:

$$\lambda_g^x = \frac{\Sigma_{g,b} - N^{238}\lambda_g^{238}\sigma_p^{238} - (N^1 - N^x)\lambda_g^1\sigma_p^1}{N^x\sigma_p^x}, \quad (3.3.5)$$

where

$$\Sigma_{g,b} = N^{238}\lambda_g^{238}\sigma_p^{238} + N^1\lambda_g^1\sigma_p^1. \quad (3.3.6)$$

In the actual application, it is too time consuming to determine the fraction of the hydrogen atomic number density that needs to be replaced with a target nuclide. Therefore, the following procedure can be used instead using data derived from  $^{238}\text{U}$  data.

- Compute a  $\sigma_{g,a}^{238}$  table as a function of  $\sigma_{g,b}^{238}$  by changing the particle number density of hydrogen for a homogeneous mixture at the fixed  $^{238}\text{U}$  ( $N^{238}$ ) particle number density.
- Compute the slowing down calculation for a mixture of  $^{238}\text{U}$  ( $N^{238}$ ),  $^1\text{H}$  ( $N^1$ ) and a target nuclide  $x$  ( $N^x$ ), and obtain a new  $\sigma_{g,a}^{238}$ .
- Read the corresponding  $\sigma_{g,b}^{238}$  from the prepared  $\sigma_{g,a}^{238}$  table.
- Calculate the hydrogen equivalent parameter using the following equation.

$$\lambda_g^x = \frac{\sigma_{g,b}^{238}N^{238} - N^{238}\lambda_g^{238}\sigma_p^{238} - N^1\lambda_g^1\sigma_p^1}{N^x\sigma_p^x}. \quad (3.3.7)$$

Intermediate resonance parameters can be also calculated for the other resonance nuclide such as  $^{235}\text{U}$ ,  $^{239}\text{Pu}$  and others.

### 3.4 Resonance self-shielded data by homogeneous models

The energy dependence of the cross sections in the library has been discretized by dividing the energy range of interest,  $10^{-5}$ eV~10MeV, into a number of broad groups. These cross sections have been obtained by flux-averaging point XS – some-times more than  $10^5$  points with typical reactor spectra. However, it is unfortunate that this procedure is impractical for the resonance isotopes in the range from 100KeV~1eV[4,5]. In this range, the cross sections exhibit many resonances and hence so that thousands of energy groups would be required for a satisfactory discretization. In general, the number of resonance groups is limited to 5~30 for very coarse group structures including 50~60 energy groups, though SCALE generally uses well over 150. The objective of the resonance treatment is to evaluate the effective cross section for the resonance isotopes in the all resonance energy groups:

$$\sigma_{x,g} = \frac{\int_{\Delta u_g} \sigma_x(u)\phi(u)du}{\int_{\Delta u_g} \phi(u)du} . \quad (3.4.1)$$

In eq. (3.4.1), the lethargy ( $u=\ln(E_0/E)$ ,  $E_0=10$ MeV) is used instead of the neutron energy and  $x$  represents a reaction type. However, the library data available for this purpose are tables of group-dependent resonance integrals (RI) (i.e., the numerator of eq. (3.4.1)) versus temperatures and background cross sections. In CENTRM, the flux in eq. (3.4.1) is calculated by solving the neutron slowing down equation for homogeneous or heterogeneous system.

To describe the slowing down equation in homogeneous infinite system containing a mixture of isotopes, indexed  $i$ , of which one is a resonance absorber, the following three assumptions are used:

- 1) Non-resonance isotopes have negligible absorption and a constant potential scattering cross section for the resonance energy range.
- 2) Resonance isotopes have resonance absorption ( $\sigma_{ai}(u)$ ) and scattering ( $\sigma_{rs,i}(u)$ ) cross sections in addition to the potential scattering cross section ( $\sigma_{pi}$ ). However, apart from the resonance energy region, these resonance absorption and scattering cross sections are negligible.
- 3) The resonances are so well-separated that the flux between them has its constant asymptotic value (i.e., set to 1).

With these assumptions, the slowing down equation at resonance energy region, and away from fission sources is given by

$$\begin{aligned} \Sigma(u)\phi(u) &= Q(u), \\ Q(u) &= \sum_i \int_{u-\Delta_i}^u \Sigma_{si}(u')\phi(u') \frac{\exp(u'-u)}{1-\alpha_i} du', \end{aligned} \quad (3.4.2)$$

where

$$\begin{aligned}
 \Sigma_{xi} &= N_i \sigma_{xi}, \\
 \Sigma_{si}(u) &= \Sigma_{pi} + \Sigma_{rs,i}(u), \\
 \Sigma_i(u) &= \Sigma_{si}(u) + \Sigma_{ai}(u), \\
 \Sigma(u) &= \sum_i \Sigma_i(u), \\
 \alpha_i &= (A_i - 1)^2 / (A_i + 1)^2, \\
 \Delta_i &= -\ln(\alpha_i).
 \end{aligned} \tag{3.4.3}$$

In eq. (3.4.3),  $\Sigma(u)$  and  $N$  are the total macroscopic cross section and atomic number density, respectively and  $A_i$  and  $\Delta_i$  are the atomic mass and the maximum lethargy gain per collision with isotope  $i$ , respectively.  $1 - \alpha_i$  is the maximum fractional energy loss per collision with isotope  $i$ .

In a homogeneous system, for a given composition the self-shielded cross section can be calculated using equations (3.4.1) through (3.4.3) and the corresponding background cross section can be calculated using eq. (3.4.5).

$$\sigma_b = \frac{\sum_{i=all} N_i \lambda_i \sigma_{i,p}}{N_R}. \tag{3.4.5}$$

Different background cross sections can be obtained by using various composition mixtures (typically a mixture of  $^1\text{H}$  and target resonance nuclide), creating a self-shielded cross-section table as a function background cross section. Therefore, for a given composition, the corresponding background cross section can be easily calculated using eq. (3.4.5) and the self-shielded cross section can be read from the table directly.

### 3.5 Resonance self-shielded data by heterogeneous models

In a heterogeneous system, for a given composition and geometry the self-shielded cross section can also be calculated using equations (3.4.1) through (3.4.3). However, since there is a leakage effect in equation (3.1.1); equation (3.3.1) cannot be used to obtain the corresponding background cross section. Thus, the equivalence theory between the infinite homogeneous and heterogeneous problems was devised by introducing an equivalence cross section ( $\Sigma_e$ ) as follows:

$$\Sigma_b = \lambda \Sigma_p \quad \rightarrow \quad \Sigma_b = \lambda \Sigma_p + \Sigma_e. \tag{3.5.1}$$

The balance equation with an equivalence theory and an elimination of the scattering resonance ( $\lambda \Sigma_{rs}$ ) thus becomes:

$$(\Sigma_a + \lambda \Sigma_p + \Sigma_e) \phi = \lambda \Sigma_p + \Sigma_e. \tag{3.5.2}$$

The background cross section is not a physical quantity but an artificial one used to retrieve the correct self-shielded cross section at a given composition and geometry. It is only important to be consistent in the procedures between its generation and its use. Therefore, the elimination of the scattering resonance is possible as long as consistency is maintained. In a heterogeneous system, the self-shielded scalar flux can be obtained by solving eq. (3.2.1) with a self-shielded absorption cross section obtained by solving eq. (3.1.1). The corresponding background cross section is obtained as follows:

$$\sigma_{g,b} = \frac{\Sigma_{g,b}}{N_R} = \frac{\sum_i N_i \lambda_i \sigma_{i,g,p} + \Sigma_{g,e}}{N_R} = \frac{\sigma_{g,a} \phi_g}{1 - \phi_g}. \quad (3.5.3)$$

The following is a procedure to obtain the self-shielded cross section table.

- (a) Compute the pointwise slowing down calculations, eq. (3.1.1), by using CENTRM/PMC for the heterogeneous models.
- (b) Edit various MG self-shielded cross sections including absorption ( $\sigma_{g,a}$ ), fission ( $\sigma_{g,f}$ ) and scattering ( $\sigma_{g,s}$ ).
- (c) Solve the MG fixed source eq. (3.5.4) with  $\sigma_{g,a}$  from step (b) to obtain the scalar flux ( $\phi_g$ ).

$$\hat{\Omega} \cdot \nabla \psi_{g,k} + \sum_i (\Sigma_{i,g,a}^k + \lambda_{i,g} \Sigma_{i,p}^k) \psi_{g,k}(\hat{\Omega}) = \sum_i \lambda_{i,g} \Sigma_{i,p}^k \quad (3.5.4)$$

- (d) Obtain the corresponding background cross section ( $\sigma_{g,b}$ ) using eq. (3.5.3) for the heterogeneous model.
- (e) Repeat the procedures (a) through (d) by changing the geometry and composition configurations to obtain different background cross sections.
- (f) Add infinite dilution cross sections and complete the self-shielded cross section tables as a function of background cross section for  $\sigma_{g,a}$ ,  $\sigma_{g,f}$  and  $\sigma_{g,s}$ .
- (g) Repeat the above procedures for various temperatures.

In eq. (3.4.1) the numerator is defined as a resonance integral (RI) with the flux and cross section a function of lethargy instead of Energy. In reference [Sta83], the self-shield cross section is approximated using the background cross section ( $\sigma_b$ ) and the resonance integral:

$$\sigma_{g,a}(\sigma_b) = \frac{R_{g,a}(\sigma_b)}{1 - R_{g,a}(\sigma_b)/\sigma_b}, \quad (3.5.5)$$

and

$$\nu \sigma_{g,f}(\sigma_b) = \frac{R_{g,f}(\sigma_b)}{1 - R_{g,a}(\sigma_b)/\sigma_b}, \quad (3.5.5)$$

where  $\sigma_{g,a}$  is an absorption cross section,  $\sigma_{g,f}$  a fission cross section,  $\nu$  the number of neutrons released from a fission, and  $R_{g,a}$  and  $R_{g,\nu f}$  are absorption and  $\nu$ \*fission resonance integrals, respectively. Using equations (3.5.5) and (3.5.6), the self-shield cross section table can be converted into the resonance integral table.

Various background cross sections can be achieved by changing geometrical and compositional configurations as shown in Table 3.5.1.

Table 3.5.1 Example of variations for  $^{238}\text{U}$

| Case             | 1                | 2      | 3   | 4   | 5    | 6   | 7   | 8   | 9   | 10   | 11    | 12     | 13      | 14   | 15    | 16     |
|------------------|------------------|--------|-----|-----|------|-----|-----|-----|-----|------|-------|--------|---------|------|-------|--------|
| Volume           | Fuel             | 1.0    | 1.0 | 1.0 | 1.0  | 1.0 | 1.0 | 1.0 | 1.0 | 1.0  | 1.0   | 1.0    | 1.0     | 1.0  | 1.0   | 1.0    |
|                  | Clad             | 1.0    | 1.0 | 1.0 | 1.0  | 1.0 | 1.0 | 1.0 | 1.0 | 1.0  | 1.0   | 1.0    | 1.0     | 1.0  | 1.0   | 1.0    |
|                  | Mod              | 1.0    | 1.0 | 1.0 | 1.0  | 1.0 | 2.0 | 5.0 | 5.0 | 5.0  | 5.0   | 5.0    | 5.0     | 5.0  | 5.0   | 5.0    |
| Fuel             | $^{235}\text{U}$ | 1.0    | 1.0 | 1.0 | 1.0  | 1.0 | 1.0 | 1.0 | 0.5 | 0.25 | 0.125 | 0.0625 | 0.03125 | 0.01 | 0.01  | 0.01   |
|                  | $^{238}\text{U}$ | 1.0    | 1.0 | 1.0 | 1.0  | 1.0 | 1.0 | 1.0 | 0.5 | 0.25 | 0.125 | 0.0625 | 0.03125 | 0.01 | 0.001 | 1.0E-6 |
|                  | $^{16}\text{O}$  | 1.0    | 1.0 | 1.0 | 1.0  | 1.0 | 1.0 | 1.0 | 0.5 | 0.25 | 0.125 | 0.0625 | 0.03125 | 0.01 | 0.01  | 0.01   |
| H <sub>2</sub> O | $^1\text{H}$     | 2.5E-3 | 0.2 | 0.5 | 0.75 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0  | 1.0   | 1.0    | 1.0     | 1.0  | 1.0   | 1.0    |
|                  | $^{16}\text{O}$  | 2.5E-3 | 0.2 | 0.5 | 0.75 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0  | 1.0   | 1.0    | 1.0     | 1.0  | 1.0   | 1.0    |

### 3.6 Resonance self-shielded data by Monte Carlo code

Conceptually this procedure is almost same with the deterministic procedure introduced in Sections 3.4 and 3.5. The only difference is to perform Monte Carlo CE calculations to edit MG self-shielded cross sections instead of performing pointwise slowing down calculation with a deterministic transport code. Either eigenvalue or fixed source Monte Carlo calculations can be performed.

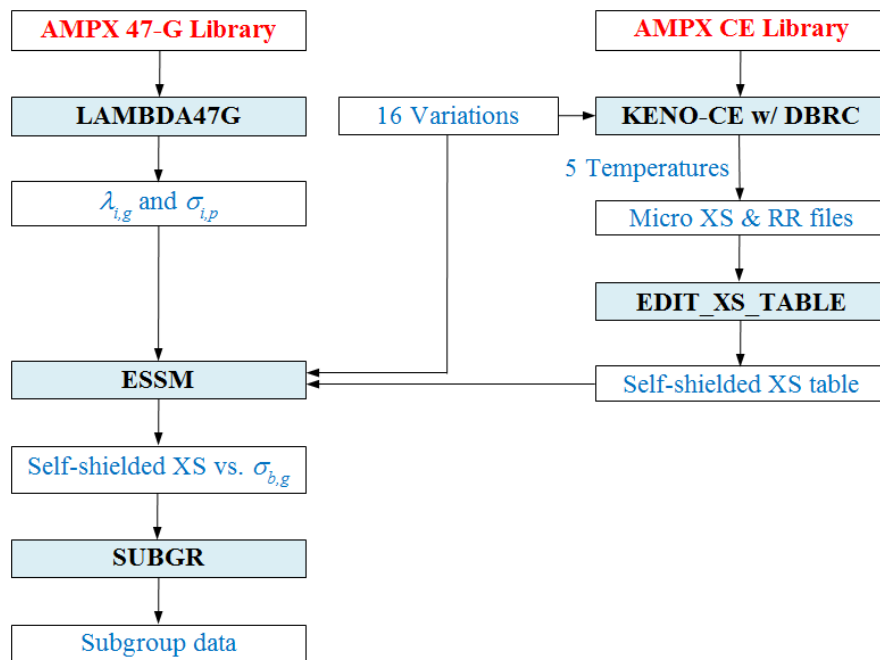


Figure 3.6.1 Procedure to generate resonance data with epithermal upscattering

No deterministic slowing down program is available to consider epithermal upscattering (>5.0 eV). Therefore the Bondarenko f-factor (self-shielded cross section) tables must be generated by performing the continuous energy Monte Carlo calculations and evaluating the corresponding background cross sections [Kim03]. The F-factor tables can be generated by using either homogeneous or heterogeneous models. The key feature of this procedure is to perform the Monte Carlo calculations to obtain MG self-shielded cross sections instead of solving deterministic PW slowing down calculations. In this study, the KENO CE calculations were performed to edit effective microscopic cross sections rather than the CENTRM slowing down calculations. The corresponding background cross sections can be obtained by the same method described in Sections 3.4 and 3.5. The Bondarenko F-factors have been obtained by performing the KENO CE calculations without and with the Doppler Broadening Rejection Correction (DBRC) [Bec10] option to consider the resonance upscattering effect. Figure 3.6.1 provides the procedure to generate resonance data with epithermal upscattering by using KENO-CE.

### 3.7 Unresolved resonance self-shielded data

In the conventional MG procedure, the Bondarenko f-factor tables at the URR energy groups are generated by using a narrow resonance approximation with a J-function method [Lee11] for which arbitrary background cross-sections are provided. The background cross-section is estimated for the specified composition and geometry, and the self-shielded 1-D cross-sections are read from the f-factor table. This procedure introduces some bias in the self-shielded cross-sections at the URR groups. We are going to solve this problem by improving the overall procedure including a new generation of the f-factor tables for the URR groups.

Typically narrow resonance approximation could be a good solution for the URR energy range in which PW neutron spectrum could be approximated in terms of total and background cross-sections. Since the probability tables are provided at the URR energy range which include cross-section levels and weights, multi-group self-shielded cross-section tables (f-factor table) could be obtained by using a narrow resonance approximation with probability table as follows:

$$\sigma_{x,g}(\sigma_0) = \frac{\int_g \sum_m p^m \sigma_x^m(E) \phi^m(E, \sigma_0) dE}{\int_g \sum_m p^m \phi^m(E, \sigma_0) dE}, \quad (3.7.1)$$

$$\phi^m(E, \sigma_0) = \frac{\sigma_0}{\sigma_t^m(E) + \sigma_0} W(E), \quad (3.7.2)$$

where

- $\sigma_0$  = background cross-section,
- $\sigma_x^m$  = a cross-section level of the level  $m$  and reaction  $x$ ,
- $\sigma_t^m$  = a total cross-section level of the level  $m$ ,

$p^m$  = a probability of the level  $m$ ,  
 $\sigma_{x,g}$  = a self-shielded cross-section of reaction  $x$ .

### 3.8 Self-shielded data for within-group elastic scattering

The AMPX MG library does include temperature independent elastic scattering matrix which results in temperature bias when performing self-shielding calculation based on Bondarenko approach. Recently new MT=2022 has been added to consider removal correction to generate temperature dependent elastic scattering matrix. DECLIB has been modified to generate temperature dependent temperature dependent elastic scattering matrix by using pre-obtained temperature dependent background cross sections.

The new MT=2022 is for within-group cross section ( $\sigma_g^{within}$ ) for elastic scattering which includes Bondarenko F-factors and 1D data. If background cross section is determined by ESSM or subgroup method, total elastic and within-group elastic cross sections can be obtained through interpolation. Temperature dependent elastic scattering matrix at temperature  $T$  can be obtained by using the following equations.

$$\tilde{\sigma}_{gg}^{elastic}(T) = \sigma_g^{within}(T). \quad (3.8.1)$$

$$\tilde{\sigma}_{gg'}^{elastic}(T) = \sigma_{gg'}^{elastic}(T_0) \cdot \frac{\sigma_g^{elastic}(T) - \sigma_g^{within}(T)}{\sum_{g'} \sigma_{gg'}^{elastic}(T_0) - \sigma_{gg}^{elastic}(T_0)}. \quad (3.8.2)$$

The DECLIB program has been modified to estimate temperature dependent elastic scattering matrix by using eqs. (3.8.1) and (3.8.2) with the nuclide-wise background cross sections.

## 4. RESONANCE SELF-SHIELDING METHODS

### 4.1 Subgroup method

Figure 4.1.1 shows the coarse energy group including resonances. Effective self-shielded cross section for this group can be obtained by the following flux weighting:

$$\sigma_{x,g} = \frac{\int_{\Delta u_g} \sigma_x(u) \phi(u) du}{\int_{\Delta u_g} \phi(u) du}, \quad (4.1.1)$$

where  $\sigma_{x,g}$  denotes the effective cross section of reaction  $x$  at the energy group  $g$ ,  $u$  is lethargy, and  $\phi(u)$  the continuous or pointwise scalar flux. Continuous scalar flux can be obtained by solving the slowing down equation with the point-wise cross sections.

In the subgroup method, the resonances are divided by the subgroup levels and the corresponding probability for each subgroup level ( $\sigma_{xn}$ ) is defined as the subgroup weight ( $w_{xn}$ ) as shown in Figure 4.1.1. [Hel03] Therefore, eq. (4.1.1) can be approximated with removing the coarse group index as follows:

$$\sigma_x = \frac{\sum_n w_{xn} \sigma_{xn} \phi_n}{\sum_n w_{an} \phi_n}, \quad (4.1.2)$$

where the summation of the subgroup weights is unity.

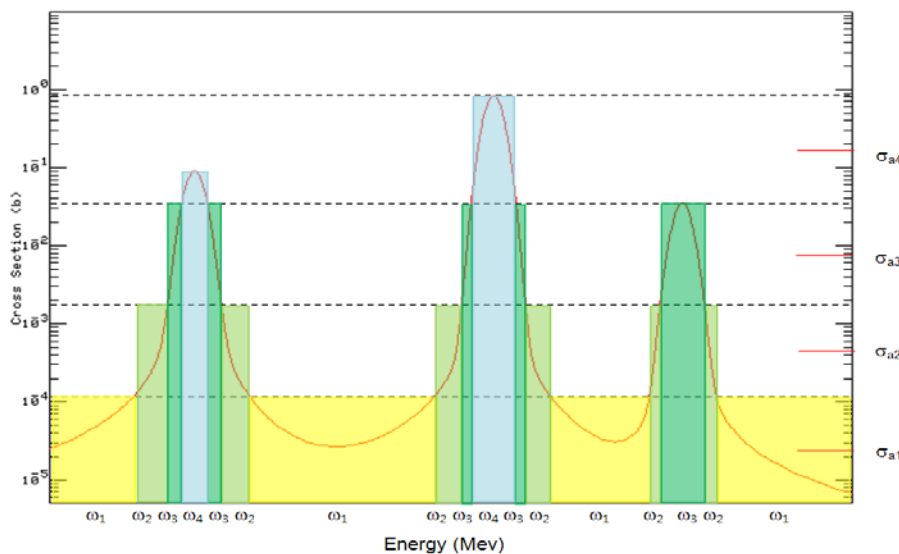


Figure 4.1.1 Resonances and subgroups



In the transport lattice calculation, the final goal of the resonance treatment is to construct a procedure such that the self-shielded cross section estimated by eq. (4.1.2) will be identical to the self-shielded cross section by eq. (4.1.1). The self-shielded scalar flux  $\phi_n$  in eq. (4.1.2) should be estimated from the fixed source transport calculation. If there is only one resonant nuclide, the fixed source transport equation will be

$$\hat{\Omega} \cdot \nabla \psi_n + (N_r \sigma_{an} + \sum_i \lambda_i \Sigma_p^i) \psi_n = \sum_i \lambda_i \Sigma_p^i, \quad (4.1.3)$$

and

$$\phi_n = \int \psi_n d\hat{\Omega}, \quad (4.1.4)$$

where  $N_r$  denotes the particle number density for the resonant nuclide,  $\Sigma_p$  a potential cross section, and  $\psi_n$  an angular flux. Equivalence theory enforces that the self-shielded scalar flux is expressed with the absorption ( $\sigma_{an}$ ) and the background cross sections ( $\sigma_{bn}$ ).

$$\phi_n = \frac{\sigma_{bn}}{\sigma_{an} + \sigma_{bn}}. \quad (4.1.5)$$

Typically the background cross section in the heterogeneous model is divided into two parts as follows:

$$\sigma_{bn} = \lambda \sigma_p + \sigma_{en}(\sigma_{an}), \quad (4.1.6)$$

where  $\lambda$  denotes an intermediate resonance parameter,  $\sigma_p$  a potential cross section, and  $\sigma_{en}$  an equivalence cross section. By using eq. (4.1.5), eq. (4.1.2) can be written as follows:

$$\sigma_x = \frac{\sum_n w_{xn} \sigma_{xn} \frac{\sigma_{bn}}{\sigma_{an} + \sigma_{bn}}}{1 - \sum_n w_{an} \frac{\sigma_{an}}{\sigma_{an} + \sigma_{bn}}}. \quad (4.1.7)$$

Therefore, if the subgroup levels and weights are given for a certain nuclide, the effective self-shielded cross section can be obtained by estimating the corresponding background cross sections ( $\sigma_{bn}$ ) in the lattice calculation. Since the equivalence cross section is not sensitive to the absorption cross section, but to the geometrical configuration,  $\sigma_{bn}$  is often approximated by a single background cross section  $\sigma_b$ . Eq. (4.1.7) can be written as follows:

$$\sigma_x = \frac{\sum_n w_{xn} \sigma_{xn} \frac{\sigma_b}{\sigma_{an} + \sigma_b}}{1 - \sum_n w_{an} \frac{\sigma_{an}}{\sigma_{an} + \sigma_b}}. \quad (4.1.8)$$

If there are several resonant nuclides, there will be a resonant interference between the resonant nuclides. Since this interference will affect on the self-shielded scalar flux, eq. (4.1.8) can be written as follows:

$$\sigma_x = \frac{\sum_n w_{xn} \sigma_{xn} \frac{\sigma_b}{\sigma_{an} + \hat{\sigma}_a + \sigma_b}}{1 - \sum_n w_{an} \frac{\sigma_{an}}{\sigma_{an} + \hat{\sigma}_a + \sigma_b}}, \quad (4.1.9)$$

where  $\hat{\sigma}_a$  is the total absorption cross section of the other resonant nuclides. The effective self-shielded cross sections are estimated iteratively.

#### 4.2 Subgroup data generation

The resonance integral ( $RI_x$ ) is defined as the numerator of eq. (4.1.1), and the resonance integral divided by lethargy width is also called resonance integral ( $R_x$ ). The resonance integral can be written by using the subgroup weights and levels from eq. (4.1.7) as follows:

$$R_x = \sum_n w_n \sigma_{xn} \frac{\sigma_{bn}}{\sigma_{an} + \sigma_{bn}} \approx \sum_n w_n \sigma_{xn} \frac{\sigma_b}{\sigma_{an} + \sigma_b}. \quad (4.2.1)$$

Since the denominator of eq. (4.1.1) can be understood as a scalar flux ( $\phi$ ) for the coarse energy group, eq. (4.1.1) can be rewritten as follows:

$$\sigma_x = \frac{R_x}{\phi}. \quad (4.2.2)$$

As described in Section 3.5, the constituent compositions and the geometrical configurations are varied to obtain various background cross sections to complete the resonance integral table. Index 'k' for the variation cases can be added to eq. (4.2.2).

$$\sigma_{x,k} = \frac{R_{x,k}}{\phi_k} \quad (k = 1, 2, \dots, K). \quad (4.2.3)$$

The resonance integral ( $R_{x,k}$ ) can be estimated directly from the IRFFACTOR program for the variation cases. The subgroup levels and weights can be obtained from eqs. (4.2.1) by the least

square fitting to minimize the difference between the original resonance integral and the reconstructed resonance integral using eq. (4.2.1).

Subgroup levels are arbitrary given and the corresponding subgroup weights are obtained by minimizing the following function  $f$ .

$$f(w_1, \dots, w_K) = \sum_k \frac{1}{R_{x,k}^2} \left( R_{x,k} - \sum_n w_{xn} \sigma_{xn,k} \frac{\sigma_{bn,k}}{\sigma_{an,k} + \sigma_{bn,k}} \right)^2. \quad (4.2.4)$$

$$f(w_1, \dots, w_K) = \sum_k \frac{1}{R_{x,k}^2} \left( R_{x,k} - \sum_n w_{xn} \sigma_{xn,k} \frac{\sigma_{b,k}}{\sigma_{an,k} + \sigma_{b,k}} \right)^2. \quad (4.2.5)$$

Subgroup levels are to be automatically adjusted to minimize the difference between the original resonance integral and the reconstructed resonance integral using the subgroup levels and weights. The resonance integral table from the IRFFACTOR calculation includes the background cross sections as a function of absorption cross section at each variation  $k$  to cover all the subgroup levels. Although the subgroup levels are varied at iterations, the corresponding background cross section ( $\sigma_{bn,k}$ ) can be obtained through the interpolation of the given table. SUBGR includes two options to generate the subgroup weights and levels by using eqs. (4.2.4) and (4.2.5) which use the level dependent and constant background cross sections, respectively.

When the resonance interference is neglected, errors in estimating the effective self-shielded cross section come mainly from the subgroup levels and weights themselves, and the scalar flux estimation. In the real application the self-shielded scalar flux is estimated by the following equation.

$$\hat{\phi}_k = 1 - \sum_n w_{an} \frac{\sigma_{an,k}}{\sigma_{an,k} + \sigma_{bn,k}}, \quad (4.2.6)$$

where the subgroup weights and levels are given and the corresponding background cross sections ( $\sigma_{bn,k}$ ) are obtained by the fixed source transport calculations of the transport lattice code using eq. (4.1.3). The difference between the scalar flux ( $\phi_k$ ) in eq. (4.2.3) and the scalar flux ( $\hat{\phi}_k$ ) in eq. (4.2.6) causes a difference in the effective self-shielded cross section.

In order to remove this error a new method [Joo09] has been proposed recently to generate the subgroup weights, in which the subgroup weights are to be estimated to conserve the self-shielded cross sections as follows:

$$f(w_1, \dots, w_K) = \sum_k \frac{1}{R_{x,k}^2} \left( \hat{R}_{x,k} - \sum_n w_{xn} \sigma_{xn,k} \frac{\sigma_{bn,k}}{\sigma_{an,k} + \sigma_{bn,k}} \right)^2, \quad (4.2.7)$$

where

$$\hat{R}_{x,k} = \sigma_{x,k} \hat{\phi}_k = \sigma_{x,k} \left( 1 - \sum_n w_{an} \frac{\sigma_{an,k}}{\sigma_{an,k} + \sigma_{bn,k}} \right). \quad (4.2.8)$$

### 4.3 Subgroup method for non-uniform temperature distribution

In a heterogeneous system, the self-shielded resonance cross sections are estimated from the self-shielded scalar fluxes obtained by the following fixed source transport equation.

$$\hat{\Omega} \cdot \nabla \psi_{g,m} + \sum_i (\Sigma_{i,a,g}^m + \lambda_{i,g} \Sigma_{i,p}) \psi_{g,m}(\hat{\Omega}) = \sum_i \lambda_{i,g} \Sigma_{i,p}, \quad (4.3.1)$$

where subscript  $m$  denotes a problem case with different absorption cross section levels at energy group  $g$ . In eq. (4.3.1),  $\Sigma_{i,a,g}$  and  $\Sigma_{i,p}$  denote macroscopic absorption and potential cross sections of nuclide  $i$ , respectively, and  $\lambda_{i,g}$  intermediate resonance parameter.

Eq. (4.3.1) should be modified for the resonance transport calculations involving non-uniform temperature distribution in which the macroscopic absorption cross sections should include the temperature distribution as follows:

$$\Sigma_{i,a,g}^m = N_i \sigma_{i,a,g}^m(T_{ave.}) \frac{\sigma_{i,a,g}^m(T)}{\sigma_{i,a,g}^m(T_{ave.})} \approx N_i \sigma_{i,a,g}^m(T_{ave.}) f_{i,a,g}(T), \quad (4.3.2)$$

where  $T$  and  $T_{ave.}$  are local and volume-averaged temperatures, respectively. While the function  $f(T)$  was approximated by the following equation [Wem07].

$$f_{i,a,g}(T) = \frac{R_{i,a,g}(T, \sigma_p)}{R_{i,a,g}(T_{ave.}, \sigma_p)} \cdot \frac{\sigma_p - R_{i,a,g}(T_{ave.}, \sigma_p)}{\sigma_p - R_{i,a,g}(T, \sigma_p)}, \quad (4.3.3)$$

where

$$\sigma_p \approx \frac{\sum_{j=all} N_j \lambda_{j,g} \sigma_{j,p}}{N_i}, \quad (4.3.4)$$

$N_i$  is the particle number density of nuclide  $i$ , and  $R_{i,a,g}$  resonance integral. Eq. (4.3.4) can be rewritten without any approximation as follows:

$$\sigma_{b,g}^m = \frac{\sum_{j=all} N_j \lambda_{j,g} \sigma_{j,p} + \Sigma_{e,g}^m}{N_i}. \quad (4.3.5)$$

Therefore the explicit equation will be

$$f_{i,a,g}^m(T) = \frac{\sigma_{i,a,g}^m(T)}{\sigma_{i,a,g}^m(T_{ave.})} = \frac{R_{i,a,g}(T, \sigma_{b,g}^m)}{R_{i,a,g}(T_{ave.}, \sigma_{b,g}^m)} \cdot \frac{\sigma_{b,g}^m - R_{i,a,g}(T_{ave.}, \sigma_{b,g}^m)}{\sigma_{b,g}^m - R_{i,a,g}(T, \sigma_{b,g}^m)}. \quad (4.3.6)$$

When performing a MPACT calculation with T-H feedback,  $\Sigma_{e,g}^m$  can be obtained from the previous outer iteration.

Another better approximation is to get the correction factor from the previous step as follows:

$$\Sigma_{i,a,g}^{m(\ell+1)} = N_i \sigma_{i,a,g}^m(T_{ave.}) \frac{\sigma_{i,a,g}^m(T)}{\sigma_{i,a,g}^m(T_{ave.})} \approx N_i \sigma_{i,a,g}^m(T_{ave.}) \frac{\sigma_{i,a,g}^{(\ell)}(T)}{\sigma_{i,a,g}^{(\ell)}(T_{ave.})}. \quad (4.3.7)$$

#### 4.4 Embedded self-shielding method (ESSM)

ESSM [Hon11, Wil12] has been developed for the resonance self-shielding calculation which requires the fixed source transport calculations in estimating background cross sections. When estimating resonance self-shielded cross section by ESSM, absorption cross sections ( $\Sigma_{i,a,g}$ ) at fuel and cladding are iteratively determined by solving the fixed source transport eq. (4.4.1).

$$\hat{\Omega} \cdot \nabla \varphi_g(\hat{\Omega}_m) + \sum_j (\Sigma_{j,a,g} + \lambda_{j,g} \Sigma_{j,p}) \varphi_g(\hat{\Omega}_m) = \sum_j \lambda_{j,g} \Sigma_{j,p}, \quad (4.4.1)$$

$$\hat{\Omega} \cdot \nabla \varphi_g(\hat{\Omega}_m) + \Sigma_{t,g} \varphi_g(\hat{\Omega}_m) = S_g, \quad (4.4.2)$$

where  $\varphi_g$  is an angular flux of group  $g$ ,  $\Sigma_{j,a,g}$  a macroscopic absorption cross section of nuclide  $j$ ,  $\Sigma_{j,p}$  a macroscopic potential cross section and  $\lambda_{j,g}$  an intermediate resonance parameter. The corresponding macroscopic and microscopic background cross sections can be obtained by using the following eq. (4.4.3).

$$\Sigma_{i,b,g} = \frac{\Sigma_{i,a,g} \phi_{i,g}}{1 - \phi_{i,g}} \rightarrow \sigma_{i,b,g}^j = \frac{\Sigma_{i,b,g}}{N_{i,j}}. \quad (4.4.3)$$

Self-shielded absorption cross sections for each nuclide can be read from the Bondarenko F-factor tables in the AMPX MG library [Dun02], and the absorption cross sections are utilized in updating  $\Sigma_{j,a,g}$ . And then the same ESSM calculation will be performed. This iteration procedure will be continued until the macroscopic background cross sections ( $\Sigma_{i,b,g}$ ) are converged.

There are several issues associated with ESSM as follows:

- Counter influence between cladding and fuel in ESSM
- ESSM with correction factors for non-uniform temperature distribution

The above issues are discussed in detail and probable solutions are proposed for some issues.

## 5. GENERATION OF MISCELLANEOUS DATA

### 5.1 Transport cross sections for $^1\text{H}$

The MOC eigenvalue calculation with high order ( $\geq P_2$ ) scattering requires 2 times longer computing time compared to the  $P_0$  calculation. Typically most of transport lattice codes are utilizing out-scattering based transport corrected  $P_0$  (TCP $_0$ ) scattering matrix in which diagonal terms of  $P_0$  scattering matrix are subtracted by total  $P_1$  scattering cross sections. The  $P_1$  corrected  $P_0$  scattering matrix for  $^1\text{H}$  may include negative value in the diagonal components which causes negative flux resulting in convergence error. Another issue of out-scattering based transport correction does show significant global power tilt for whole core problems including reflector. Therefore a proper transport correction method is needed to have reasonable neutron leakage as high order scattering is considered and to guarantee no negative flux.

The Neutron Leakage Conservation method (NLCP $_0$ ) [Her13] used in the CASMO series for long time has been used to generate transport cross sections for  $^1\text{H}$ . The goal of NLCP $_0$  is to obtain diffusion coefficients to have same neutron leakage as obtained in the high order scattering transport calculation as follows:

$$\hat{\Omega} \cdot \nabla \phi_g(\hat{\Omega}) = -D_g \nabla^2 \phi_g. \quad (5.1.1)$$

When we have a 1D model, neutron leakages at the specified surfaces can be obtained from the high order scattering transport calculation.

$$Leakage = J_{right} - J_{left}. \quad (5.1.2)$$

We can determine diffusion coefficients to have same leakage in the diffusion calculation by using the following derivation.

$$-D \nabla^2 \phi + \Sigma_a \phi = \frac{1}{k_{eff}} \nu \Sigma_f \phi. \quad (5.1.3)$$

$$\nabla^2 \phi + B^2 \phi = 0 \rightarrow \frac{\partial^2}{\partial x^2} \phi(x) + B^2 \phi(x) = 0. \quad (5.1.4)$$

$$\phi(x) = \cos(Bx). \quad (5.1.5)$$

$$J_{right} - J_{left} = \int_0^W dx \left( -D \frac{\partial^2}{\partial x^2} \phi(x) \right) = \int_0^W dx D B^2 \phi(x) = D B^2 \int_0^W dx \phi(x) = D B^2 \hat{\phi}. \quad (5.1.6)$$

$$D = \frac{J_{right} - J_{left}}{B^2 \hat{\phi}}. \quad (5.1.7)$$

$$B_{geom}^2 = \left( \frac{\pi}{W} \right)^2. \quad (5.1.8)$$

When performing the multigroup transport calculation with high order scattering, it is required to check if the group-wise bucklings are constant independently upon energy group. This work can be completed by drawing normalized neutron spectrum shape for each group.

$$B_{geom}^2 = B^2 = B_g^2. \quad (5.1.9)$$

We can obtain multi-group diffusion coefficients and transport cross sections.

$$D_g = \frac{J_{g,right} - J_{g,left}}{B^2 \hat{\phi}_g}. \quad (5.1.10)$$

$$\Sigma_{tr,g} = \frac{1}{3D_g}. \quad (5.1.11)$$

The computational model was from the reference paper [Her13] as follows:

- 1-D slab 100 cm w/ vacuum boundary, 0.005 cm mesh size
- All  $^1\text{H}$  with  $4.780\text{E}+23$  atom/cm<sup>3</sup>
- 9 temperatures : 293.6, 350.0, 400.0, 450.0, 500.0, 550.0, 600.0, 650.0, 800.0 K
- Source :  $^{235}\text{U}$  fission spectrum with buckled Cosine spatial distribution

This work can be done by using either continuous energy Monte Carlo or multigroup deterministic transport codes. A new 1D MOC code including high order scattering capability has been developed to generate transport corrected  $^1\text{H}$  cross sections. Figure 5.1.1 provides the 47-group transport correction factors for  $^1\text{H}$  with various temperatures.

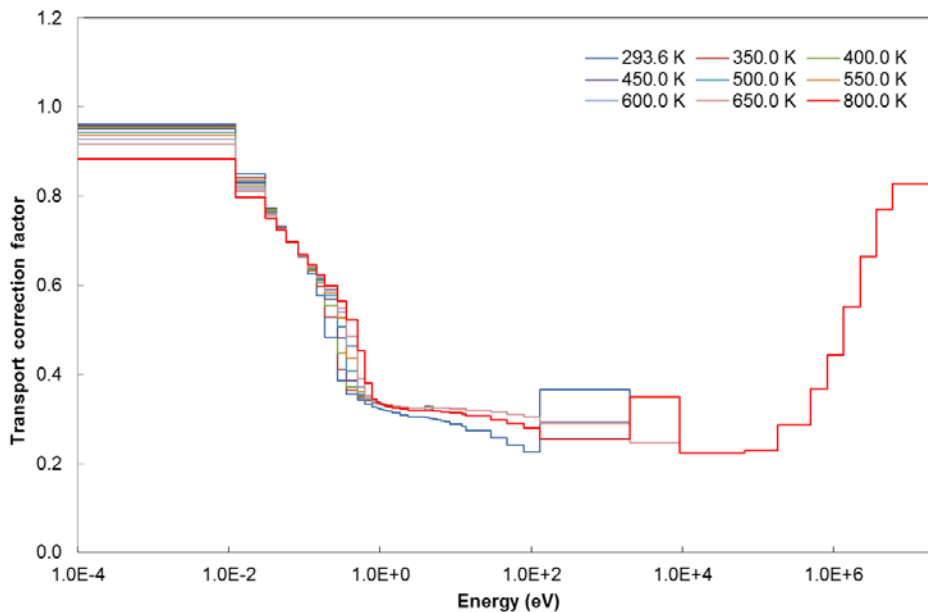


Figure 5.1.1 47-group transport correction factors for  $^1\text{H}$

## 5.2 Transient data

Since the ORNL AMPX code package [Wia14] does not support to generate transient data such as decay constants and delayed neutron fractions, the NJOY program [Mac94] developed at LANL has been utilized to generate the transient data. Since there was no information for the ENDF/B nuclides including transient data, the NJOY input files were generated for all 37 heavy nuclides in the MPACT library. Tape 24 from the GROUPT module includes 47-group neutron data including transient data. A simple editing program has been developed to edit transient data such as decay constants and delayed neutron fractions. In order to obtain total number of neutrons released per fission to be used in obtaining delayed neutron fraction, typical 47-group PWR spectrum has been obtained from the VERA progression problem 1b as shown Figure 5.2.1.

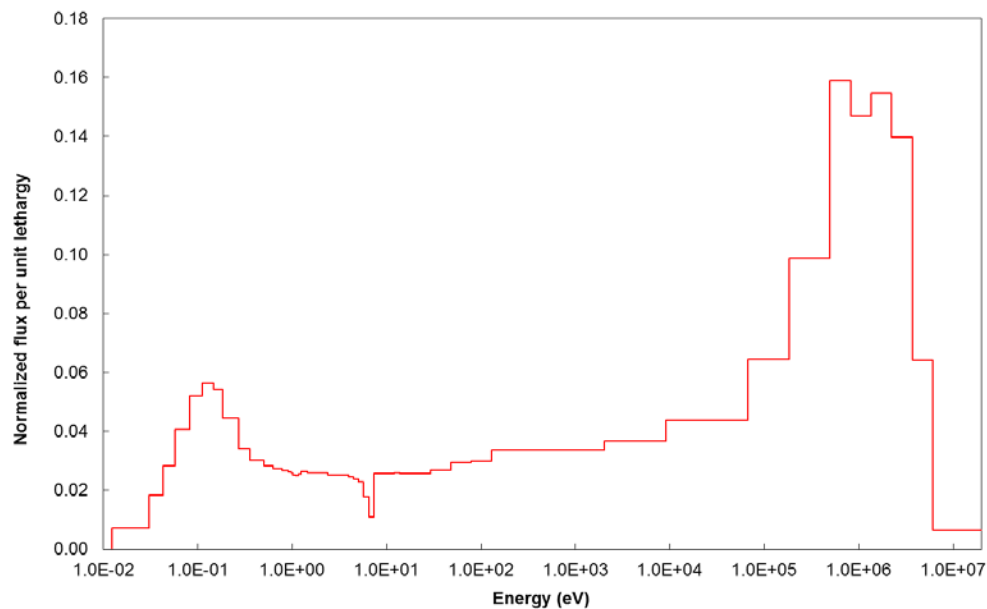


Figure 5.2.1 47-group weighting function

The ENDF/B-7.0 and 7.1 neutron data include transient data only for 21 nuclides. Figure 5.2.2 provides transient data including nuclide identification, total number of neutrons released per fission, decay constants and delayed neutron fractions for each nuclide.

|              |              |              |              |              |              |
|--------------|--------------|--------------|--------------|--------------|--------------|
| 90232        | 2.417105E+00 |              |              |              |              |
| 1.240000E-02 | 3.340000E-02 | 1.210000E-01 | 3.210000E-01 | 1.210000E+00 | 3.290000E+00 |
| 8.237550E-04 | 2.839532E-03 | 3.391569E-03 | 9.961879E-03 | 3.760367E-03 | 1.827437E-03 |
| 91231        | 2.429466E+00 |              |              |              |              |
| 1.240000E-02 | 3.340000E-02 | 1.210000E-01 | 3.210000E-01 | 1.210000E+00 | 3.290000E+00 |
| 3.772028E-04 | 1.018876E-03 | 7.348728E-04 | 1.774840E-03 | 4.798565E-04 | 1.831329E-04 |
| 91233        | 2.429815E+00 |              |              |              |              |
| 1.240000E-02 | 3.340000E-02 | 1.210000E-01 | 3.210000E-01 | 1.210000E+00 | 3.290000E+00 |
| 7.746643E-04 | 2.092295E-03 | 1.509164E-03 | 3.645136E-03 | 9.855432E-04 | 3.761569E-04 |
| 92233        | 2.500074E+00 |              |              |              |              |



|              |              |              |              |              |              |
|--------------|--------------|--------------|--------------|--------------|--------------|
| 1.247938E-02 | 3.227449E-02 | 1.046448E-01 | 2.941518E-01 | 1.242443E+00 | 1.022323E+01 |
| 2.567975E-04 | 7.701037E-04 | 5.537597E-04 | 1.142208E-03 | 2.146458E-04 | 2.239787E-05 |
| 92234        | 2.633240E+00 |              |              |              |              |
| 1.248240E-02 | 3.129530E-02 | 1.074630E-01 | 3.032230E-01 | 1.288730E+00 | 1.044220E+01 |
| 2.754770E-04 | 1.165225E-03 | 8.999964E-04 | 2.034102E-03 | 4.624642E-04 | 6.164186E-05 |
| 92235        | 2.438879E+00 |              |              |              |              |
| 1.249056E-02 | 3.182406E-02 | 1.093753E-01 | 3.169898E-01 | 1.353983E+00 | 8.636377E+00 |
| 2.183402E-04 | 1.136937E-03 | 1.102438E-03 | 3.142815E-03 | 9.126013E-04 | 3.225451E-04 |
| 92236        | 2.585205E+00 |              |              |              |              |
| 1.248999E-02 | 3.068560E-02 | 1.091980E-01 | 3.224705E-01 | 1.378583E+00 | 1.021014E+01 |
| 2.294104E-04 | 1.726554E-03 | 1.473179E-03 | 3.893188E-03 | 1.382420E-03 | 2.693897E-04 |
| 92237        | 2.556079E+00 |              |              |              |              |
| 1.249577E-02 | 3.037978E-02 | 1.068975E-01 | 3.240317E-01 | 1.334253E+00 | 9.544175E+00 |
| 2.081865E-04 | 2.419830E-03 | 2.019126E-03 | 6.103966E-03 | 2.421639E-03 | 5.043196E-04 |
| 92238        | 2.788314E+00 |              |              |              |              |
| 1.249423E-02 | 3.025520E-02 | 1.159376E-01 | 3.414764E-01 | 1.318630E+00 | 9.979027E+00 |
| 1.631869E-04 | 1.811873E-03 | 2.016816E-03 | 7.130047E-03 | 3.684767E-03 | 9.734561E-04 |
| 93237        | 2.943373E+00 |              |              |              |              |
| 1.248610E-02 | 3.077587E-02 | 1.065464E-01 | 3.139911E-01 | 1.334211E+00 | 1.050869E+01 |
| 1.169523E-04 | 7.177156E-04 | 6.106869E-04 | 1.677948E-03 | 4.688961E-04 | 8.045746E-05 |
| 94238        | 2.992942E+00 |              |              |              |              |
| 1.332500E-02 | 3.117100E-02 | 1.161500E-01 | 2.888400E-01 | 8.561400E-01 | 2.713800E+00 |
| 5.261947E-05 | 3.337788E-04 | 2.202819E-04 | 4.974348E-04 | 2.221220E-04 | 7.038188E-05 |
| 94239        | 2.867711E+00 |              |              |              |              |
| 1.248110E-02 | 2.994667E-02 | 1.071553E-01 | 3.176193E-01 | 1.352380E+00 | 1.069116E+01 |
| 7.441243E-05 | 6.029267E-04 | 4.208394E-04 | 8.788662E-04 | 2.338068E-04 | 3.832867E-05 |
| 94240        | 3.081800E+00 |              |              |              |              |
| 1.332900E-02 | 3.051000E-02 | 1.151600E-01 | 2.974000E-01 | 8.476600E-01 | 2.879600E+00 |
| 9.336029E-05 | 7.385726E-04 | 4.404519E-04 | 9.638943E-04 | 5.243305E-04 | 1.597601E-04 |
| 94241        | 2.946479E+00 |              |              |              |              |
| 1.359900E-02 | 2.996600E-02 | 1.167300E-01 | 3.069100E-01 | 8.701000E-01 | 3.002800E+00 |
| 9.923741E-05 | 1.233243E-03 | 7.840723E-04 | 1.920227E-03 | 1.086294E-03 | 3.750161E-04 |
| 94242        | 3.146448E+00 |              |              |              |              |
| 1.360300E-02 | 3.023800E-02 | 1.154300E-01 | 3.041900E-01 | 8.272100E-01 | 3.137200E+00 |
| 1.229591E-04 | 1.449077E-03 | 7.864419E-04 | 2.042159E-03 | 1.412084E-03 | 4.483058E-04 |
| 95241        | 3.236158E+00 |              |              |              |              |
| 1.333800E-02 | 3.079800E-02 | 1.130500E-01 | 2.867700E-01 | 8.653600E-01 | 2.643000E+00 |
| 4.687282E-05 | 3.351367E-04 | 2.062589E-04 | 4.438472E-04 | 2.274633E-04 | 5.988720E-05 |
| 95243        | 3.564782E+00 |              |              |              |              |
| 1.349900E-02 | 2.975900E-02 | 1.137700E-01 | 2.985900E-01 | 8.820200E-01 | 2.811100E+00 |
| 5.213651E-05 | 6.568510E-04 | 3.428585E-04 | 7.020686E-04 | 3.694160E-04 | 1.071179E-04 |
| 96242        | 3.753388E+00 |              |              |              |              |
| 1.295600E-02 | 3.124500E-02 | 1.129100E-01 | 2.783400E-01 | 8.710400E-01 | 2.196900E+00 |
| 2.760101E-05 | 1.027275E-04 | 5.117072E-05 | 1.024804E-04 | 6.385763E-05 | 1.359061E-05 |
| 96245        | 3.597515E+00 |              |              |              |              |
| 1.340000E-02 | 3.070000E-02 | 1.130000E-01 | 3.001000E-01 | 8.340000E-01 | 2.768600E+00 |
| 3.942668E-05 | 3.180151E-04 | 2.974124E-04 | 6.592391E-04 | 3.654310E-04 | 9.948328E-05 |
| 98249        | 3.888388E+00 |              |              |              |              |
| 1.351700E-02 | 2.945000E-02 | 1.053200E-01 | 2.929800E-01 | 8.474900E-01 | 2.469800E+00 |
| 1.709062E-05 | 2.721421E-04 | 9.365231E-05 | 1.804290E-04 | 1.120952E-04 | 1.896568E-05 |
| 98251        | 4.140665E+00 |              |              |              |              |
| 1.569700E-02 | 2.883000E-02 | 1.076900E-01 | 3.245800E-01 | 8.837100E-01 | 2.631400E+00 |
| 9.953225E-06 | 6.522987E-04 | 3.156976E-04 | 4.898075E-04 | 3.069656E-04 | 4.397731E-05 |

Figure 5.2.2 Transient data from ENDF/B-7.0

### 5.3 Effective recoverable energy per fission

The energy release per fission is required for burnup calculations, and is usually defined without the kinetic energy of the incident neutrons and the energy carried away by neutrinos. However, the energy release per fission includes the contributions from the kinetic energy of incident neutrons and from the decay of the capture products:

$$W_{fiss}^i = E_r^i + Q_c^i + W_n^i, \quad (5.3.1)$$

where  $W_{fiss}^i$  is the effective energy release in fission,  $E_r^i$  is the energy release in fission excluding the energy carried away by neutrinos and the kinetic energy of the incident neutrons,  $Q_c^i$  is the contribution from the decay of the capture products, and  $W_n^i$  is the kinetic energy of the incident neutrons. Index  $i$  refers to a fissionable nuclide in all cases.

$E_r^i$  values are extracted from the MF=1/MT=458 section of an evaluated nuclear data file, while the kinetic energy of the incident neutron is obtained by approximately averaging the energy with the fission reaction rate. The averaging process involved the multi-group approximation and the cross sections obtained from the MCNP calculations based on ENDF/B cross section data:

$$W_n^i = \frac{\sum_g \bar{E}_g \sigma_{f,g}^i \phi_g}{\sum_g \sigma_{f,g}^i \phi_g}, \quad (5.3.2)$$

where  $\sigma_{f,g}^i$  is the microscopic fission cross section of the fissionable nuclide  $i$  in the energy group  $g$ ,  $\phi_g$  is the neutron spectrum, and  $E_g$  is the average group energy. A typical PWR fuel pin was used in this calculation.

The average group energy was a simple mean of the group boundary values, except for the highest energy groups for which more accurate values were adopted to give the average energy in a Maxwellian fission spectrum at a temperature corresponding to 1.4 MeV:

$$E_g = \frac{\int E \sqrt{E} \cdot \exp(-E/T) dE}{\int \sqrt{E} \cdot \exp(-E/T) dE}. \quad (5.3.3)$$

The additional energy released due to gamma activation is calculated from

$$Q_c = (\bar{\nu} - 1)Q, \quad (5.3.4)$$

where  $(\bar{\nu} - 1)$  represents the average number of neutrons that are captured, and  $Q=6.1$  MeV [Jam99].

#### 5.4 Effective background cross sections

While the AMPX MG library includes resonance data for all nuclides and energy groups, the MPACT library includes resonance data for the specified nuclides and energy groups. For example, 49 nuclides include subgroup data and resonance integral tables and 191 nuclides out of 295 include IR parameters only for groups 10~26. If infinitely dilute cross sections are provided in the MPACT MG library, the MPACT results would underestimate the multiplication

factors at very high burnup points due to higher capture cross sections. Therefore, pre-self-shielded cross sections need to be included in the MPACT MG library.

At low burnup points heavy and fission product nuclides produced by decay and neutron reactions do not significantly contribute on the multiplication factor due to tiny amount of atomic number densities. It is reasonable to determine high burnup point at which background cross sections are estimated to be used in obtaining 1D self-shielded cross sections. This work can be done by the MPACT single pin depletion calculation to obtain atomic number density at high burnup point (50 MWD/kgU) and then effective microscopic background cross sections can be obtained from the MPACT subgroup or ESSM calculations.

## 6. GENERATION OF THE MPACT MG LIBRARY

### 6.1 Generation of the initial AMPX MG library

The AMPX MG library can be generated by using the program EXSITE by which all the AMPX module input files for all nuclides can be automatically prepared through expanding a template shown in Figure 6.1.1. Figure 6.1.2 provides the AMPX procedure to generate the AMPX MG library where the left side of flow chart is for the conventional procedure and the right side for new procedure to improve the Bondarenko resonance data. The following data are required in generating the AMPX MG library.

- ENDF/B-VII neutron cross section data
- ENDF/B-VII listing file
- ENDF/B-VII Doppler broadening data files
- ENDF/B-VII probability table files
- Pointwise weighting function

```

=neutron_mg
master=result/neut_
temperature=293
broaden=/home/c31/exsite/result/broaden_
neutgroups=47
thermalgroups=26 neutuserdef=yes
neutbounds=<1.000000E-6 1.239596E-2 3.061288E-2 4.275520E-2 5.692194E-2 8.196816E-2 1.115699E-1
1.457206E-1 1.844302E-1 2.705213E-1 3.576701E-1 5.032318E-1 6.250621E-1 7.820830E-1 9.099967E-1
9.710043E-1 1.013699E+0 1.072203E+0 1.125397E+0 1.166404E+0 1.235105E+0 1.457402E+0 1.855391E+0
2.382393E+0 3.927903E+0 4.450897E+0 5.043477E+0 5.715008E+0 6.476017E+0 7.338215E+0 8.315287E+0
1.209903E+1 1.371000E+1 2.902291E+1 4.785117E+1 7.889325E+1 1.300704E+2 2.034700E+3 9.118801E+3
6.737900E+4 1.831601E+5 4.978702E+5 8.208500E+5 1.353400E+6 2.231299E+6 3.678800E+6 6.065300E+6
2.000000E+7>
weightuser=yes
weighttbl=/home/dw8/libraries/endl7.1/mg/252/casl_lib_flux
makeyield=no
thermcut=5.05 thermsplice=5 thintermal=yes
makethermal=yes
thermaltemp=<293.0 600.0 900.0 1200.0 2400.0>
input=input/neut_
evals=/neptune/home02/ykk/libraries/endl7.0/endl7.0.xml
end

=bondarenko_prob
master=result/bond_
broaden=/home/c31/exsite/result/broaden_
nld=result/neut_ temperature=293
prob=/home/c31/exsite/result/ptable_
weightuser=yes
weighttbl=/home/dw8/libraries/endl7.1/mg/252/casl_lib_flux
sig0=<1.0E8 1000000.0 100000.0 10000.0 5000.0 2000.0 1000.0 640.0 320.0 160.0 120.0 80.0 40.0 20.0
10.0 4.0 1.0 1.0E-6>
temps=<293.0 600.0 900.0 1200.0 2400.0>
input=input/bond_
evals=/neptune/home02/ykk/libraries/endl7.0/endl7.0.xml
end

=bind_mg
master=result/master_ neutron=result/neut_
addfree=yes addbond=yes bond=result/bond_
input=input/bind_
evals=/neptune/home02/ykk/libraries/endl7.0/endl7.0.xml
end

```

Figure 6.1.1 Sample template to generate the AMPX input files

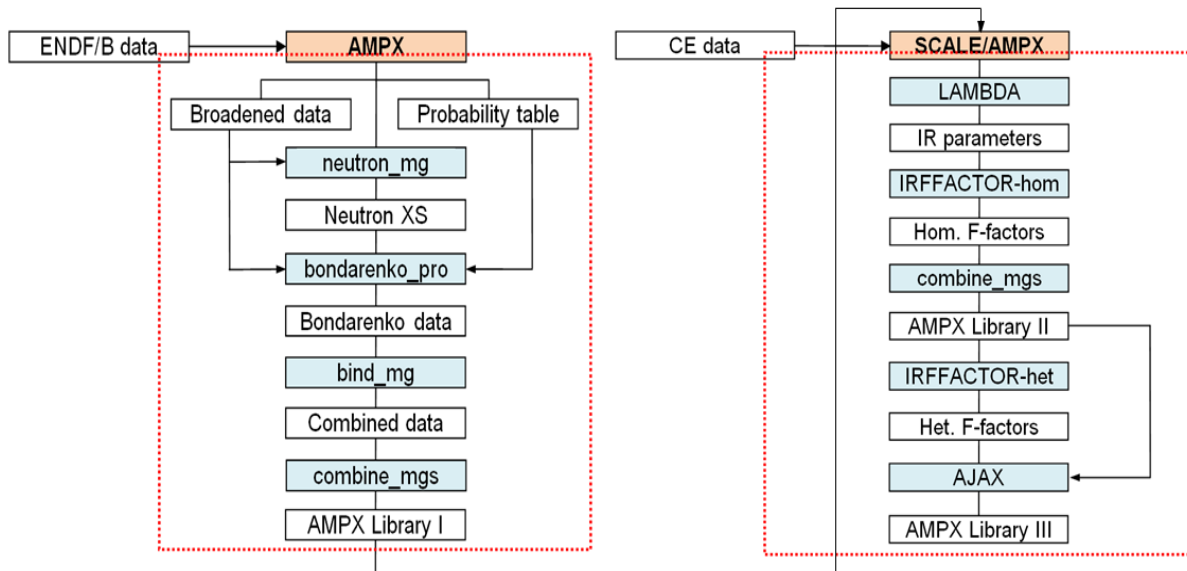


Figure 6.1.2 Flow chart of the AMPX procedure

After completing the AMPX calculations for all nuclides, individual MG data files can be merged into an AMPX MG library (I) by the AMPX AJAX module.

## 6.2 Intermediate resonance parameters and Homogeneous F-factors

The intermediate resonance (IR) parameters are generated for all nuclides by using the AMPX-LAMBDA module with a procedure introduced in Section 3.3 in which the IR parameters are estimated only for the specified energy groups (for example, the 47-g library for groups 10~46) and unity is assigned to other energy groups. In addition homogeneous F-factors are generated for the same energy groups for the IR parameters by the AMPX-IRFFACTOR-hom module with a procedure described in Section 3.4 which are substituted for the original F-factors.

Figure 6.2.1 provides a template to be expanded to generate the LAMBDA and IRFFACTOR-hom input files for all nuclides. An expanded sample input for  $^{235}\text{U}$  is shown in Appendix A.1. Figure 6.2.2 provides a flow chart of the resonance data generation procedure in which pointwise slowing down calculations are performed by CENTRM. The following data are included in the AMPX MG master library.

- Intermediate resonance parameters : all nuclides, all energy groups
- Homogeneous F-factors : Nuclides Atomic number  $\geq 40$  (Zr)

```

=ffactors
master=/home/ykk/libraries/endl7.0/mg/47g/ampx47g_70_07182014_00_ykk.bin
cedesc=/scale/scale_dev_data/ce_v7.0_endf
cedatadir=ln -fs /scale/scale_dev_data/cekenolib_7.0
kernels=/scale/scale_dev_data/endl_b/vers7
low=10 high=46
out=irflib/irf
input=input/in_
evals=/neptune/home02/ykk/libraries/endl7.0/endl7.0.xml
end
    
```

Figure 6.2.1 Sample template to generate the IR parameters and homogeneous F-factors

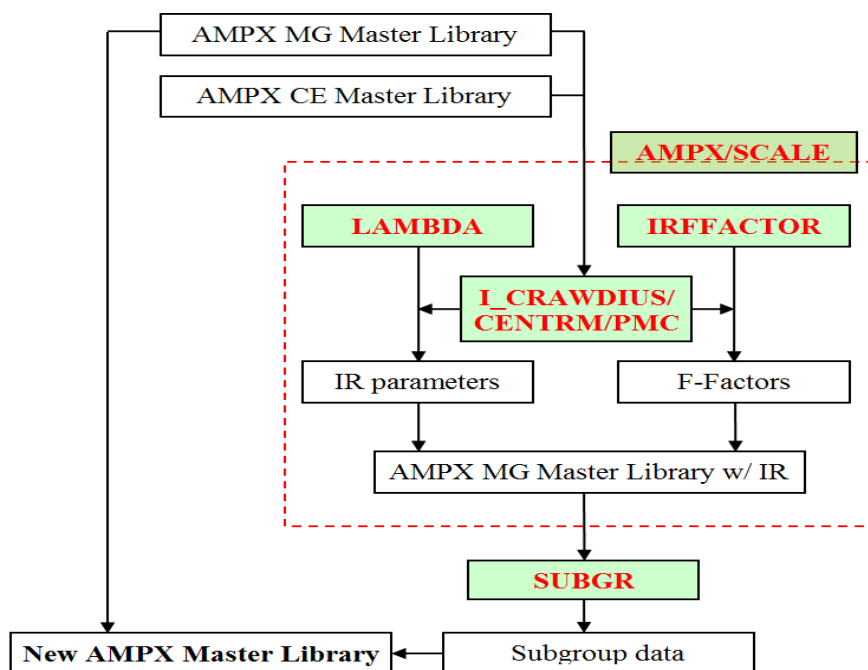


Figure 6.2.1 Flow chart of the resonance data generation procedure

### 6.3 Heterogeneous F-factors

Only selected nuclides include heterogeneous F-factors which are very important resonance nuclides significantly impacting on the neutronics result. The following 20 nuclides have been selected to have heterogeneous F-factors. Option for level dependent background cross sections has been used for all nuclides except for Ag, In and Cd nuclides.

- $^{107}\text{Ag}$ ,  $^{109}\text{Ag}$ ,  $^{113}\text{Cd}$ ,  $^{113}\text{In}$ ,  $^{115}\text{In}$ ,  $^{155}\text{Gd}$ ,  $^{156}\text{Gd}$ ,  $^{157}\text{Gd}$ ,  $^{158}\text{Gd}$ ,  $^{232}\text{Th}$ ,  $^{233}\text{U}$ ,  $^{235}\text{U}$ ,  $^{236}\text{U}$ ,  $^{238}\text{U}$ ,  $^{238}\text{Pu}$ ,  $^{239}\text{Pu}$ ,  $^{240}\text{Pu}$ ,  $^{241}\text{Pu}$ ,  $^{242}\text{Pu}$ ,  $^{241}\text{Am}$

The IRFFACTOR-het input file for  $^{238}\text{U}$  to generate heterogeneous F-factors is shown in Figure 6.3.1 and a companion input file for various geometrical and compositional configurations is provided in Appendix B.1. Since typically Bondarenko approach is not able to consider resonance interference effect explicitly, heterogeneous F-factors can be generated in two different ways with or without considering explicit resonance interference. ‘Single’ and ‘Multiple’ indicate without and with resonance interference, respectively. Both models are applied only to 5 heavy nuclides including  $^{235}\text{U}$ ,  $^{238}\text{U}$ ,  $^{239}\text{Pu}$ ,  $^{240}\text{Pu}$ , and  $^{241}\text{Pu}$ .

The Heterogeneous IRFFACTOR results in two F-factor tables of which the first one can be incorporated by the AMPX-AJAX and the second one (subgrpdata) is for SUBGR to generate subgroup data. The first one (or the AMPX MG master library) also can be utilized in the subgroup data generation. However, in order to generate subgroup data by using eq. (4.2.7) to conserve cross sections, subgroup level dependent background cross sections should be added to the F-factor tables (self-shielded cross section table). The AMPX MG library does not include the subgroup level dependent background cross sections, but the ‘subgrpdata’ file includes them.

```

' -----
' Heterogeneous irffactor for U-238 :: multiple absorber ENDF/B-VII.0
' -----
=shell
ln -sf $RTNDIR/./cases/U238-HetCases-new.inp het_input
ln -sf /home/ykk/libraries/endlf7.0/mg/47g/ffactor/ampx47g_70_07182014_01_ykk.bin ft88f001
end

=irffactor
in=88 out=92 fnuc=92238 medit=3 essm=yes nterp=1
absopt=1 mopt=3 removal=yes
cellfil="het_input"
ehres=20.0E+03
subgrxs= 1.0 5.0 10.0 50.0 100.0 500.0 1000.0 5000.0 10000.0 20000.0 end
end

=shell
mv ft92f001 $RTNDIR/U238_ffactors_multiple_m3
mv subgrpdata $RTNDIR/subgrpdata_U238_m3
end
    
```

Figure 6.3.1 Sample input to generate the heterogeneous F-factors

## 6.4 Generation of subgroup data

At first Bondarenko F-factors in the AMPX MG library should be converted into resonance integral tables to be used in the subgroup data generation. And then subgroup data are generated by using SUBGR. Subgroup data including weights and levels have been generated for important 49 resonance nuclides as shown in Table 6.4.1 and for all energy groups. Figures 6.4.1 and 6.4.2 provide the standard subgroup level and the SUBGR input files, respectively.

Table 6.4.1 List of nuclides including subgroup data

| No | Nuclide           | ID    | Method | $\sigma_a$ | $\nu\sigma_f$ | No | Nuclide           | ID    | Method | $\sigma_a$ | $\nu\sigma_f$ |
|----|-------------------|-------|--------|------------|---------------|----|-------------------|-------|--------|------------|---------------|
| 1  | <sup>91</sup> Zr  | 40091 | RI     | o          | x             | 26 | <sup>163</sup> Dy | 66163 | RI     | o          | x             |
| 2  | <sup>96</sup> Zr  | 40096 | RI     | o          | x             | 27 | <sup>164</sup> Dy | 66164 | RI     | o          | x             |
| 3  | <sup>95</sup> Mo  | 42095 | RI     | o          | x             | 28 | <sup>166</sup> Er | 68166 | RI     | o          | x             |
| 4  | <sup>99</sup> Tc  | 43099 | RI     | o          | x             | 29 | <sup>167</sup> Er | 68167 | RI     | o          | x             |
| 5  | <sup>103</sup> Rh | 45103 | RI     | o          | x             | 30 | <sup>176</sup> Hf | 72176 | RI     | o          | x             |
| 6  | <sup>108</sup> Pd | 46108 | RI     | o          | x             | 31 | <sup>177</sup> Hf | 72177 | RI     | o          | x             |
| 7  | <sup>107</sup> Ag | 47107 | RI     | o          | x             | 32 | <sup>178</sup> Hf | 72178 | RI     | o          | x             |
| 8  | <sup>109</sup> Ag | 47109 | RI     | o          | x             | 33 | <sup>179</sup> Hf | 72179 | RI     | o          | x             |
| 9  | <sup>113</sup> In | 49113 | RI     | o          | x             | 34 | <sup>180</sup> Hf | 72180 | RI     | o          | x             |
| 10 | <sup>115</sup> In | 49115 | RI     | o          | x             | 35 | <sup>182</sup> W  | 74182 | RI     | o          | x             |
| 11 | <sup>131</sup> Xe | 54131 | RI     | o          | x             | 36 | <sup>183</sup> W  | 74183 | RI     | o          | x             |
| 12 | <sup>133</sup> Cs | 55133 | RI     | o          | x             | 37 | <sup>184</sup> W  | 74184 | RI     | o          | x             |
| 13 | <sup>152</sup> Sm | 62152 | RI     | o          | x             | 38 | <sup>186</sup> W  | 74186 | RI     | o          | x             |
| 14 | <sup>151</sup> Eu | 63151 | RI     | o          | x             | 39 | <sup>232</sup> Th | 90232 | XS     | o          | x             |
| 15 | <sup>152</sup> Eu | 63152 | RI     | o          | x             | 40 | <sup>233</sup> U  | 92233 | XS     | o          | o             |
| 16 | <sup>153</sup> Eu | 63153 | RI     | o          | x             | 41 | <sup>235</sup> U  | 92235 | XS     | o          | o             |
| 17 | <sup>154</sup> Eu | 63154 | RI     | o          | x             | 42 | <sup>236</sup> U  | 92236 | XS     | o          | o             |
| 18 | <sup>155</sup> Eu | 63155 | RI     | o          | x             | 43 | <sup>238</sup> U  | 92238 | XS     | o          | x             |
| 19 | <sup>155</sup> Gd | 64155 | XS     | o          | x             | 44 | <sup>238</sup> Pu | 94238 | XS     | o          | o             |
| 20 | <sup>156</sup> Gd | 64156 | XS     | o          | x             | 45 | <sup>239</sup> Pu | 94239 | XS     | o          | o             |
| 21 | <sup>157</sup> Gd | 64157 | XS     | o          | x             | 46 | <sup>240</sup> Pu | 94240 | XS     | o          | o             |
| 22 | <sup>158</sup> Gd | 64158 | XS     | o          | x             | 47 | <sup>241</sup> Pu | 94241 | XS     | o          | o             |
| 23 | <sup>160</sup> Dy | 66160 | RI     | o          | x             | 48 | <sup>242</sup> Pu | 94242 | XS     | o          | o             |
| 24 | <sup>161</sup> Dy | 66161 | RI     | o          | x             | 49 | <sup>241</sup> Am | 95241 | XS     | o          | o             |
| 25 | <sup>162</sup> Dy | 66162 | RI     | o          | x             |    |                   |       |        |            |               |

RI : Resonance Integral conservation

XS: Cross section conservation

```

$TIT:
  AMPX 47-G LIBRARY
$DIM:
  54      7      4
$NUC: 0
  00000
  L7A 0 -2.0 2.0 4.800000E+00 4.000000E+01 2.500000E+02 7.500000E+02 2.000000E+03 7.000000E+03 2.000000E+04
  L4A 0 -5.0 5.0 1.000000E+01 5.000000E+02 2.000000E+03 1.000000E+04
$NUC: 1
  40090
  L7A 0 -2.0 2.0 4.800000E+00 4.000000E+01 2.500000E+02 7.500000E+02 2.000000E+03 7.000000E+03 2.000000E+04
  L4A 0 -5.0 5.0 1.000000E+01 5.000000E+02 2.000000E+03 1.000000E+04
$NUC: 40
  92235
  L7A 0 -2.0 2.0 4.800000E+00 4.000000E+01 2.500000E+02 7.500000E+02 2.000000E+03 7.000000E+03 2.000000E+04
  L4A 0 -5.0 5.0 1.000000E+01 5.000000E+02 2.000000E+03 1.000000E+04
$NUC: 41
  92236
  L7A 0 -2.0 2.0 4.800000E+00 4.000000E+01 2.500000E+02 7.500000E+02 2.000000E+03 7.000000E+03 2.000000E+04
  L4A 0 -5.0 5.0 1.000000E+01 5.000000E+02 2.000000E+03 1.000000E+04
$NUC: 42
  92238
  L7A 0 -2.0 2.0 4.800000E+00 4.000000E+01 2.500000E+02 7.500000E+02 2.000000E+03 7.000000E+03 2.000000E+04
  L4A 0 -5.0 5.0 1.000000E+01 5.000000E+02 2.000000E+03 1.000000E+04
$END:
    
```

Figure 6.4.1 Sample of the initial subgroup level



```

%TITL
SUBGROUP DATA GENERATION
%IOPT
iop 0          !0/1/2: constant/variable(RI)/variable(sig)
con 0.05 0.005 0.005
grp 1 51
%FILE
sub  .\lib\subgr_51g_70s.lev
rit  .\lib\ssxs_51g_70s_10012015.dat
%RESO
nuc  1    40091  0  1  !
nuc  2    40096  0  1  !
nuc  3    42095  0  1  !
nuc  4    43099  0  1  !
nuc  5    45103  0  1  !
nuc  6    46108  0  1  !
nuc  7    47107  0  1  !
nuc  8    47109  0  1  !
nuc  9    49113  0  1  !
nuc 10    49115  0  1  !
nuc 11    54131  0  1  !
nuc 12    55133  0  1  !
nuc 13    62152  0  1  !
nuc 14    63151  0  1  !
nuc 15    63152  0  1  !
nuc 16    63153  0  1  !
nuc 17    63154  0  1  !
nuc 18    63155  0  1  !
nuc 19    64155  2  1  !
nuc 20    64156  2  1  !
nuc 21    64157  2  1  !
nuc 22    64158  2  1  !
nuc 23    66160  0  1  !
nuc 24    66161  0  1  !
nuc 25    66162  0  1  !
nuc 26    66163  0  1  !
nuc 27    66164  0  1  !
nuc 28    68166  0  1  !
nuc 29    68167  0  1  !
nuc 30    72176  0  1  !
nuc 31    72177  0  1  !
nuc 32    72178  0  1  !
nuc 33    72179  0  1  !
nuc 34    72180  0  1  !
nuc 35    74182  0  1  !
nuc 36    74183  0  1  !
nuc 37    74184  0  1  !
nuc 38    74186  0  1  !
nuc 39    90232  2  1  !
nuc 40    92233  2  2  !
nuc 41    92235  2  2  !
nuc 42    92236  2  2  !
nuc 43    92238  2  1  !
nuc 44    94238  2  2  !
nuc 45    94239  2  2  !
nuc 46    94240  2  2  !
nuc 47    94241  2  2  !
nuc 48    94242  2  2  !
nuc 49    95241  2  2  !
%FINE
    
```

Figure 6.4.2 Sample of the SUBGR input

## 6.5 Generation of the MPACT MG library

Appendices A and B provide the detailed data information for the MPACT MG library and the DECLIB input to generate the ENDF/B-7.1 based V4.0 MPACT 47-group library. The following data files are required for DECLIB to generate the MPACT MG library.

- The AMPX 47-g library
- ENDF/B files : Neutron data, decay constants, fission product yields
- Subgroup data and resonance integral table (49 nuclides, groups 10-26)
- Transport correction factors ( $^1\text{H}$ )
- Pre-determined background cross sections (105 nuclides)

- Transient data (21 nuclides)
- Subgroup data with epithermal upscattering ( $^{238}\text{U}$ )

Figure 6.5.1 provides a flow diagram of DECLIB to generate the MPACT MG library of which data structure is shown in Appendix C.

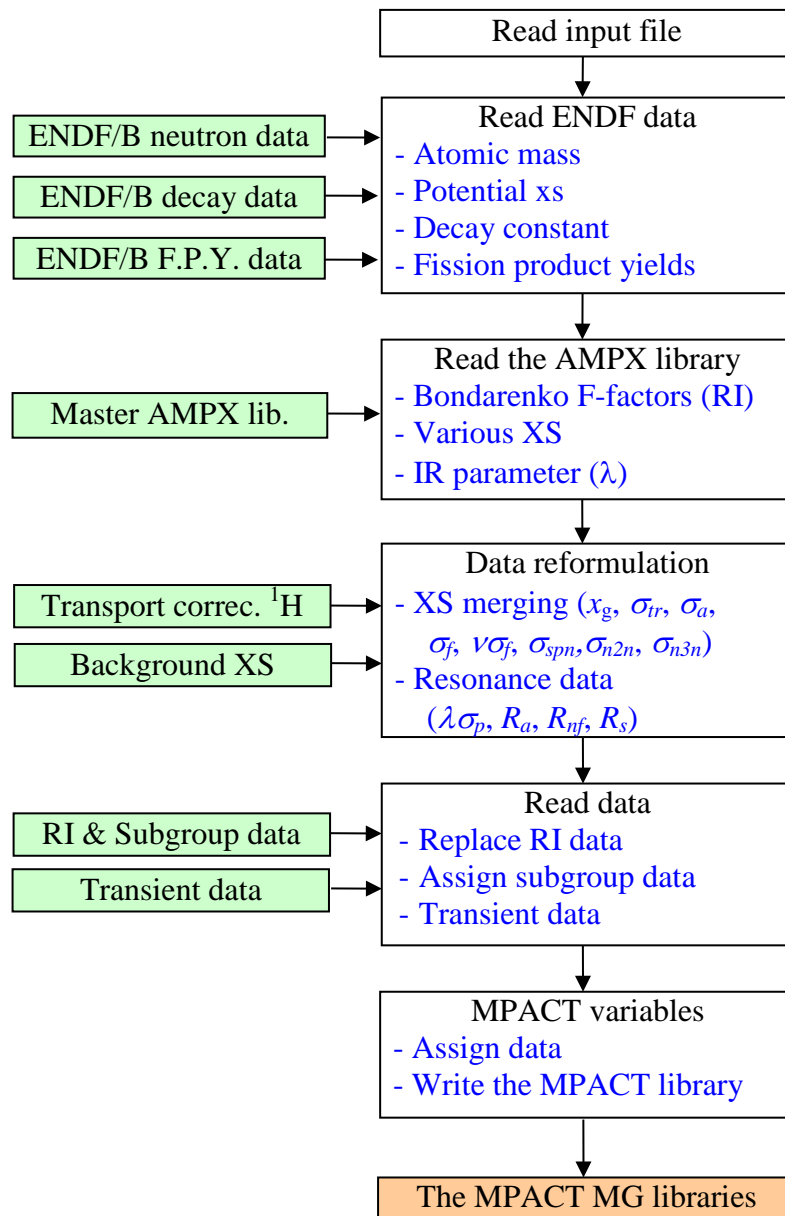


Figure 6.5.1 DECLIB calculation flow

Multigroup cross sections are directly edited from the AMPX master library. MPACT solves the time-independent Boltzmann equation as follows:

$$\hat{\Omega} \cdot \nabla \psi_g + \Sigma_{tr,g}(\vec{r})\psi_g(\vec{r}, \hat{\Omega}) = \sum_{g'} \Sigma_{sg'g} \phi_{g'} + \frac{1}{k_{eff}} \chi_g \sum_{g'} \nu \Sigma_{fg'} \phi_{g'}, \quad (6.5.1)$$

where

- $\hat{\Omega}$  : neutron direction
- $\vec{r}$  : space coordinate
- $g$  : neutron energy group
- $\psi_g$  : angular flux
- $\phi_g$  : scalar flux
- $\Sigma_{tr,g}$  : macroscopic transport cross section
- $\Sigma_{s,g'g}$  : transport corrected scattering matrix
- $\nu$  : the number of neutrons released from a fission
- $\Sigma_{f,g'}$  : fission cross section
- $\chi_g$  : fission spectrum
- $k_{eff}$  : effective multiplication factor.

The required data for the transport calculation are transport cross section, fission cross section, the number of neutrons released from a fission, and fission spectrum for each nuclide. Since the absorption and fission cross sections are modified through the resonance treatment and are needed for the depletion calculation, those cross sections should be included. And (n,2n) and (n,3n) cross sections are required for the depletion cross sections. High order (P<sub>1</sub>~P<sub>3</sub>) scattering matrices are to be included for the future use. Therefore, the multigroup data required in the library are as follows:

- a. Total cross section ( $\sigma_t$ )
- b. Absorption cross section ( $\sigma_a = \sigma_c + \sigma_f$ )
- c. Fission cross section ( $\sigma_f$ )
- d. Neutrons released from a fission ( $\nu$ )
- e. Scattering cross sections ( $\sigma_{s0-3}$ )
- f. (n,2n) cross section ( $\sigma_{n,2n}$ )
- g. (n,3n) cross section ( $\sigma_{n,3n}$ )
- h. P<sub>0-3</sub> scattering matrix ( $\sigma_{sngg'}$ )
- i. Fission spectrum ( $\chi_g$ )

Since the AMPX master library includes cross section data for various reactions, it is required to merge cross sections to generate the above underlined data. Table 6.5.1 provides which MT numbers are to be merged for the required data.

(n,2n) and (n,3n) cross sections are not treated explicitly, but included in the absorption cross section and scattering matrix as follows:

$$\sigma_{a,g}^c = \sigma_{c,g} + \sigma_{f,g} - \sigma_{n2n,g} - 2\sigma_{n3n,g}, \quad (6.5.2)$$

and

$$\sigma_{s,gg'}^c = \sigma_{s,gg'} + 2\sigma_{n2n,gg'} + 3\sigma_{n3n,gg'}. \quad (6.5.3)$$

The effective fission spectrum needs to be re-estimated by using MT=18, 452, 455, 1055 and 1099 by using the following formula.

$$\chi_g = \frac{\chi_g^{prompt} \sum_{g'} \nu_{g'}^{prompt} \sigma_{g'} \phi_{g'} + \chi_g^{delayed} \sum_{g'} \nu_{g'}^{delayed} \sigma_{g'} \phi_{g'}}{\sum_{g'} \nu_{g'}^{prompt} \sigma_{g'} \phi_{g'} + \sum_{g'} \nu_{g'}^{delayed} \sigma_{g'} \phi_{g'}}. \quad (6.5.4)$$

Table 6.5.1 MT numbers to be merged

| Cross section           | MT numbers in the AMPX master library |                                |                      |
|-------------------------|---------------------------------------|--------------------------------|----------------------|
|                         | F-factors                             | 1-D cross section              | 2-D cross section    |
| $\sigma_t$              | 1                                     | -                              | -                    |
| $\sigma_c$              | 102                                   | 103~117                        | -                    |
| $\sigma_f$              | 18                                    | -                              | -                    |
| $\nu$                   | -                                     | 452                            | -                    |
| $\sigma_{s0}$           | 2                                     | 4                              | -                    |
| $\sigma_{n,2n}$         | -                                     | 11, 16                         | 11, 16               |
| $\sigma_{n,3n}$         | -                                     | 17                             | 17                   |
| $\sigma_{sn,gg'}$       | -                                     | -                              | 2, 50~91, 1007, 1008 |
| $\chi_g^*$              | -                                     | 18, 452, 455, 1055, 1056, 1099 | -                    |
| $\lambda$               | -                                     | 2000                           | -                    |
| $\sigma_{s,g}^{within}$ | 2022                                  | -                              | -                    |

\*Since MT=1018 is just for prompt neutron,  $\chi_g$  is internally recalculated.

## 7. CHALLENGING ISSUES

### 7.1 Drawbacks of the MPACT MG library

The current MPACT 47-group library includes the following drawbacks.

- 49 nuclides include full resonance data such as subgroup data, RI table and IR parameters
- Groups 10 through 26 include resonance data.
- Pre-self-shielded cross sections should be utilized for non-resonance nuclides and non-resonance energy groups.
- Temperature independent elastic scattering matrices are available. (This is a drawback of the AMPX MG library.)
- Although the ENDF/B-7.0 AMPX MG library includes 419 nuclides, the MPACT MG library include 295 nuclides and in addition some of nuclides are duplicated for activation and fission product nuclides.
- Many of fission product nuclides include only absorption cross sections.
- There are many duplicated nuclides at activation and fission product nuclides.
- High order scattering matrices ( $\geq P_2$ ) are available only for important nuclides.

### 7.2 Resonance interference

Self-shielded cross section table can be generated by considering resonance interference effect between resonance nuclides such as  $^{238}\text{U}$  and  $^{235}\text{U}$  for the CENTRM pointwise slowing down calculation. This method may be able to provide better agreement for the specific cases in MG 1D self-shielded cross sections or reaction rates between reference solution and the MPACT result. However, resonance interferences are doubly considered due to the Bondarenko iteration of subgroup method for resonance interference, and this procedure would introduce significant  $^{235}\text{U}$  enrichment bias resulting in radial power tilt at the whole core calculation.

Resonance interference could not be considered explicitly in the conventional Bondarenko resonance self-shielding methods such as subgroup method, ESSM and Dancoff based methods when using coarse energy group structure. There can be several solutions as follows:

- a. Pre-calculated resonance interference factor tables [Kim11]
- b. ESSM+0D slowing down [Kim12]
- c. ESSM-X [Liu15]

However, these methods are still under development and include various pending technical issues associated with computing time and accuracy. These will be discussed later.

### 7.3 Resonance self-shielding effect on scattering matrix

In deterministic transport calculation, the accuracy is determined by quality of both 1D and 2D cross section data. Recently new F-factor table for removal correction has been included to

adjust scattering matrix in the AMPX MG master library. It is noted that this approach is very successful in resolving temperature bias problem due to temperature independent scattering matrix. At first a new functionality to utilize the F-factor table for removal correction needs to be incorporated into MPACT. However, since this approach is based on Bondarenko method, there should be some bias in estimating self-shielding effect in scattering matrix. New resonance self-shielding method is based on quasi 1D slowing down calculation. Therefore, removal correction factor for 2D data can be explicitly estimated simultaneously with an estimation of 1D data. These removal correction factors can be applied to scattering matrix.

#### 7.4 Intra pin self-shielding effect in ESSM

When estimating resonance self-shielded cross section by ESSM, absorption cross sections ( $\Sigma_{i,a,g}$ ) at fuel and cladding are iteratively determined by solving the fixed source transport eq. (7.4.1).

$$\hat{\Omega} \cdot \nabla \psi_g + \sum_i (\Sigma_{i,a,g} + \lambda_{i,g} \Sigma_{i,p}) \psi_g(\hat{\Omega}) = \sum_i \lambda_{i,g} \Sigma_{i,p} \cdot \quad (7.4.1)$$

Typical PWR and BWR fuel pins include UO<sub>2</sub> fuel and zirconium alloy fuel cladding in which <sup>235</sup>U, <sup>238</sup>U, <sup>91</sup>Zr and <sup>95</sup>Zr are important resonant nuclides requiring explicit resonance treatment. Since Zr nuclides are located at different zone from fuel, there is no direct resonance interference effect with fuel nuclides.

When generating F-factor tables by using homogeneous or heterogeneous models by the SCALE and AMPX procedures, only one resonant nuclide is considered to have absorption cross section and other background nuclides include only potential or scattering cross sections in slowing down and ESSM calculations. Multigroup cross sections are obtained from the slowing down calculations and the corresponding background cross sections are obtained by solving eq. (7.4.1) and then by using eq. (7.4.2).

$$\Sigma_{b,g} = \frac{\Sigma_{a,g} \phi_g}{1 - \phi_g} \rightarrow \sigma_{b,g}^i = \frac{\Sigma_{b,g}}{N_i} \cdot \quad (7.4.2)$$

When performing multigroup ESSM calculation, absorption cross sections at fuel and cladding are estimated simultaneously. Since the scalar flux ( $\phi_g$ ) at fuel is influenced by the cladding absorption cross section and vice versa, self-shielded cross sections at fuel and cladding introduce some bias because of inconsistency between how to generate F-factor table and how to use it.

In order to avoid this problem, the AMPX 56-group master library includes two sets of cross section data for <sup>91</sup>Zr and <sup>96</sup>Zr. The first set of F-factor table was prepared by using a conventional procedure, and the second set of F-factor table was generated by using a model shown in Figure

7.4.1 (a). The second set of F-factor tables cannot be utilized in other composition except for fuel cladding. While the impact on Zr nuclides of cladding by fuel is significant, the impact on U nuclides of fuel by cladding is not. In order to have the highest quality of self-shielded cross sections and a simplified procedure, this impact should be considered for both cladding and fuel with a utilization of only one set of F-factor table.

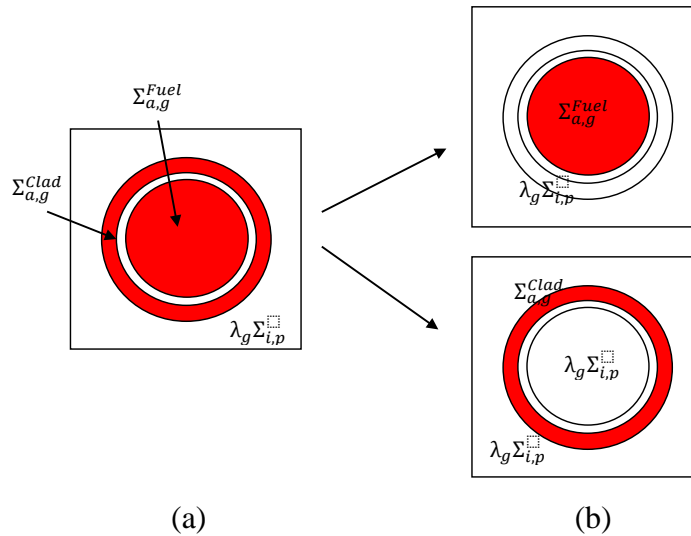


Figure 7.4.1 Counter influence between fuel and cladding

In order to solve this problem category concept in the subgroup method can be utilized. Or if fuel and cladding regions are separated from each other which is somewhat similar with category concept, this problem can be resolved which may be called a geometrical category.

Another idea is to include a special option to generate a table of macroscopic cross sections as a function of temperature for which resonance self-shielding effect, if necessary, can be considered by using a simple model such as single pin and homogeneous mixture. This option can be applied to cladding, detector and moderator. Similar approach can be applied to the subgroup method.

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Appendix A.1 Sample input of LAMBDA and IRFFACTOR-hom for  $^{235}\text{U}$ 

```

=shell
ln -fs /home/ykk/libraries/endlf7.0/mg/47g/ampx47g_70_07182014_00_ykk.bin ft77f001
cp /scale/scale_dev_data/ce_v7.0_endf .
ln -fs /scale/scale_dev_data/cekenolib_7.0
cp /scale/scale_dev_data/endlf_b/vers7//1-0 .
end
=ajax
0$$ 20 e
1$$ 1 e t
2$$ 77 4 e t
3$$ 92238 1001 92235 92235 e t
end
=lambada
in=20 out=44 bnuc=1001 fnuc=92238
iddens=0.0238333 bdens=0.0357499 dens=0.00743805
low=10 high=46
temp=600.0
end
=ajax
0$$ 46 e 1$$ 1 e t
2$$ 44 1 e t
3$$ 92235 e t
end
=shell
cp ft46f001 ${RTNDIR}/../irflib/irf_lambda_u235
end
=irffachomo
in=44 out=45 fnuc=92235 bnuc=1001
dens=1.0 ehres=2.25000E+03
low=10 high=46
end
=ajax
0$$ 48 e 1$$ 1 e t
2$$ 45 1 e t
3$$ 92235 e t
end
=shell
cp ft48f001 ${RTNDIR}/../irflib/irf_irf_u235
end

```

Appendix B.1 Sample input of IRFFACTOR-het for  $^{238}\text{U}$ 

```
=csaslx parm=centrm
hetrocells to compute f-factors of U238;
ft88f001
read composition

|
| ***CELL-1 Composition
|
u-235 1 0 9.39468E-04 293 end
u-238 1 0 2.22624E-02 293 end
o-16 1 0 4.64223E-02 293 end
al-27 2 0 6.02620E-02 293 end
h-1 3 0 1.17012E-04 293 end
o-16 3 0 5.85142E-05 293 end

|
| ***CELL-2 Composition
|
u-235 4 0 9.39468E-04 293 end
u-238 4 0 2.22624E-02 293 end
o-16 4 0 4.64223E-02 293 end
al-27 5 0 6.02620E-02 293 end
h-1 6 0 9.36093E-03 293 end
o-16 6 0 4.68114E-03 293 end

|
| ***CELL-3 Composition
|
u-235 7 0 9.39468E-04 293 end
u-238 7 0 2.22624E-02 293 end
o-16 7 0 4.64223E-02 293 end
al-27 8 0 6.02620E-02 293 end
h-1 9 0 2.34023E-02 293 end
o-16 9 0 1.17028E-02 293 end

|
| ***CELL-4 Composition
|
u-235 10 0 9.39468E-04 293 end
u-238 10 0 2.22624E-02 293 end
o-16 10 0 4.64223E-02 293 end
al-27 11 0 6.02620E-02 293 end
h-1 12 0 3.51035E-02 293 end
o-16 12 0 1.75543E-02 293 end

|
| ***CELL-5 Composition
|
u-235 13 0 9.39468E-04 293 end
u-238 13 0 2.22624E-02 293 end
o-16 13 0 4.64223E-02 293 end
al-27 14 0 6.02620E-02 293 end
h-1 15 0 4.68046E-02 293 end
o-16 15 0 2.34057E-02 293 end

|
| ***CELL-6 Composition
|
u-235 16 0 9.39468E-04 293 end
```

```
u-238 16 0 2.22624E-02 293 end
o-16 16 0 4.64223E-02 293 end
al-27 17 0 6.02620E-02 293 end
h-1 18 0 4.68046E-02 293 end
o-16 18 0 2.34057E-02 293 end
```

```
'
' ***CELL-7 Composition
'
```

```
u-235 19 0 9.39468E-04 293 end
u-238 19 0 2.22624E-02 293 end
o-16 19 0 4.64223E-02 293 end
al-27 20 0 6.02620E-02 293 end
h-1 21 0 4.68046E-02 293 end
o-16 21 0 2.34057E-02 293 end
```

```
'
' ***CELL-8 Composition
'
```

```
u-235 22 0 9.39468E-04 293 end
u-238 22 0 2.22624E-02 293 end
o-16 22 0 4.64223E-02 293 end
al-27 23 0 6.02620E-02 293 end
h-1 24 0 4.68046E-02 293 end
o-16 24 0 2.34057E-02 293 end
```

```
'
' ***CELL-9 Composition
'
```

```
u-235 25 0 4.69734E-04 293 end
u-238 25 0 1.11312E-02 293 end
o-16 25 0 2.32111E-02 293 end
al-27 26 0 6.02620E-02 293 end
h-1 27 0 4.68046E-02 293 end
o-16 27 0 2.34057E-02 293 end
```

```
'
' ***CELL-10 Composition
'
```

```
u-235 28 0 2.34867E-04 293 end
u-238 28 0 5.56561E-03 293 end
o-16 28 0 1.16056E-02 293 end
al-27 29 0 6.02620E-02 293 end
h-1 30 0 4.68046E-02 293 end
o-16 30 0 2.34057E-02 293 end
```

```
'
' ***CELL-11 Composition
'
```

```
u-235 31 0 1.17434E-04 293 end
u-238 31 0 2.78280E-03 293 end
o-16 31 0 5.80278E-03 293 end
al-27 32 0 6.02620E-02 293 end
h-1 33 0 4.68046E-02 293 end
o-16 33 0 2.34057E-02 293 end
```

```
'
' ***CELL-12 Composition
'
```

```
u-235 34 0 5.87168E-05 293 end
u-238 34 0 1.39140E-03 293 end
o-16 34 0 2.90139E-03 293 end
al-27 35 0 6.02620E-02 293 end
```

```

h-1 36 0 4.68046E-02 293 end
o-16 36 0 2.34057E-02 293 end

'
' ***CELL-13 Composition
'
u-235 37 0 2.93584E-05 293 end
u-238 37 0 6.95701E-04 293 end
o-16 37 0 1.45070E-03 293 end
al-27 38 0 6.02620E-02 293 end
h-1 39 0 4.68046E-02 293 end
o-16 39 0 2.34057E-02 293 end

'
' ***CELL-14 Composition
'
u-235 40 0 9.39468E-06 293 end
u-238 40 0 2.22624E-05 293 end
o-16 40 0 4.64223E-04 293 end
al-27 41 0 6.02620E-02 293 end
h-1 42 0 4.68046E-02 293 end
o-16 42 0 2.34057E-02 293 end

'
' ***CELL-15 Composition
'
u-235 43 0 9.39468E-06 293 end
u-238 43 0 2.22624E-08 293 end
o-16 43 0 4.64223E-04 293 end
al-27 44 0 6.02620E-02 293 end
h-1 45 0 4.68046E-02 293 end
o-16 45 0 2.34057E-02 293 end

'
' ***CELL-16 Composition (infinitely dilute; Homo mixture)
'
u-235 70 0 9.3947E-05 293 end
u-238 70 0 1.0E-12 293 end
o-16 70 0 0.046422 293 end
al-27 70 0 0.060262 293 end
h-1 70 0 4.4183E-02 293 end
o-16 70 0 2.2095E-02 293 end
'-----
end composition

read celldata

'
' ***CELL-1 Geometry
latticecell squarepitch pitch=1.2620 3 fuelr=0.4025 1
          cladr=0.4759 2 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata

'
' ***CELL-2 Geometry
latticecell squarepitch pitch=1.2620 6 fuelr=0.4025 4
          cladr=0.4759 5 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata

'
' ***CELL-3 Geometry
latticecell squarepitch pitch=1.2620 9 fuelr=0.4025 7

```

```

        cladr=0.4759 8 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata
'
' ***CELL-4 Geometry
latticecell squarepitch pitch=1.2620 12 fuelr=0.4025 10
        cladr=0.4759 11 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata
'
' ***CELL-5 Geometry
latticecell squarepitch pitch=1.2620 15 fuelr=0.4025 13
        cladr=0.4759 14 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata
'
' ***CELL-6 Geometry
latticecell squarepitch pitch=1.5728 18 fuelr=0.4025 16
        cladr=0.4759 17 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata
'
' ***CELL-7 Geometry
latticecell squarepitch pitch=2.0581 21 fuelr=0.4025 19
        cladr=0.4759 20 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata
'
' ***CELL-8 Geometry
latticecell squarepitch pitch=2.2621 24 fuelr=0.4025 22
        cladr=0.4759 23 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata
'
' ***CELL-9 Geometry
latticecell squarepitch pitch=2.2621 27 fuelr=0.4025 25
        cladr=0.4759 26 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata
'
' ***CELL-10 Geometry
latticecell squarepitch pitch=2.2621 30 fuelr=0.4025 28
        cladr=0.4759 29 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata
'
' ***CELL-11 Geometry
latticecell squarepitch pitch=2.2621 33 fuelr=0.4025 31
        cladr=0.4759 32 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata
'
' ***CELL-12 Geometry
latticecell squarepitch pitch=2.2621 36 fuelr=0.4025 34
        cladr=0.4759 35 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata
'
' ***CELL-13 Geometry
latticecell squarepitch pitch=2.2621 39 fuelr=0.4025 37
        cladr=0.4759 38 end
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata

```

```
'  
' ***CELL-14 Geometry  
latticecell squarepitch pitch=2.2621 42 fuelr=0.4025 40  
                  cladr=0.4759 41 end  
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata  
  
'  
' ***CELL-15 Geometry  
latticecell squarepitch pitch=2.2621 45 fuelr=0.4025 43  
                  cladr=0.4759 44 end  
centrmdata demin=.0001 demax=9.E3 iup=15 npxs=6 end centrmdata  
  
'  
' ***CELL-16 Geometry (infinitely dilute; Homo mixture)  
infhommedium 70 end  
centrmdata demin=.0001 demax=9.E3 iup=15 end centrmdata  
  
'-----  
end celldata  
  
end
```

## Appendix C.1 Data information in the MAPCT 47-group library

- NO1 0 non depletable  
 >0 depletable (w/ decay constant)  
 2 RI-a only (fissionable w/ kappa, beta, FPY)  
 3 RI -a & RI-vf (fissionable w/ kappa, beta, FPY)
- NO2 NFPY (Ordering no. of FPY) (+/- : cumulative/direct yield)
- NO3 Indicator for fission spectrum (0/>0 : no/yes)
- NO4 Indicator for (n,2n) & (n,3n) XS (0/1/2/3 : no/(n,2n)/(n,3n)/both)
- NO5 The number of temperatures for XS
- NO6 The number of temperatures for RI
- NO7 The number of temperatures for p2~p3
- NO8 The number of sig-0's for RI

1 – 49: Full XS + subgroup data

50 – 191: Full XS + resonance data such as IR parameter, potential XS

192 – end: Part XS

| No | NID   | Amass    | NO1 | NO2 | NO3 | NO4 | NO5 | NO6 | NO7 | NO8 | AID    |
|----|-------|----------|-----|-----|-----|-----|-----|-----|-----|-----|--------|
| 1  | 40091 | 90.9056  | 1   | 0   | 0   | 0   | 3   | 5   | 0   | 16  | Zr-91  |
| 2  | 40096 | 95.9083  | 1   | 0   | 0   | 0   | 3   | 5   | 0   | 16  | Zr-96  |
| 3  | 42595 | 94.9059  | 1   | 17  | 0   | 0   | 3   | 5   | 0   | 16  | Mo-95  |
| 4  | 43599 | 99.0005  | 1   | 23  | 0   | 1   | 3   | 5   | 0   | 16  | Tc-99  |
| 5  | 45103 | 102.9050 | 1   | 0   | 0   | 0   | 3   | 5   | 0   | 16  | Rh-103 |
| 6  | 46608 | 107.9039 | 1   | 37  | 0   | 0   | 3   | 5   | 0   | 16  | Pd-108 |
| 7  | 47107 | 106.9054 | 0   | 0   | 0   | 3   | 3   | 5   | 0   | 16  | Ag-107 |
| 8  | 47109 | 108.9045 | 0   | 0   | 0   | 3   | 3   | 5   | 0   | 16  | Ag-109 |
| 9  | 49113 | 112.9039 | 0   | 0   | 0   | 0   | 3   | 5   | 0   | 16  | In-113 |
| 10 | 49115 | 114.9041 | 0   | 0   | 0   | 0   | 3   | 5   | 0   | 16  | In-115 |
| 11 | 54631 | 130.9055 | 1   | 57  | 0   | 3   | 3   | 5   | 0   | 16  | Xe-131 |
| 12 | 55633 | 132.9057 | 1   | 63  | 0   | 1   | 3   | 5   | 0   | 16  | Cs-133 |
| 13 | 62152 | 151.9201 | 0   | 0   | 0   | 3   | 5   | 5   | 0   | 16  | Sm-152 |
| 14 | 63151 | 150.9195 | 0   | 0   | 0   | 3   | 5   | 5   | 0   | 16  | Eu-151 |
| 15 | 63152 | 151.9251 | 1   | 0   | 0   | 3   | 5   | 5   | 0   | 16  | Eu-152 |
| 16 | 63153 | 152.9217 | 0   | 0   | 0   | 3   | 5   | 5   | 0   | 16  | Eu-153 |
| 17 | 63154 | 153.9223 | 1   | 0   | 0   | 3   | 5   | 5   | 0   | 16  | Eu-154 |
| 18 | 63155 | 154.9229 | 1   | 0   | 0   | 3   | 5   | 5   | 0   | 16  | Eu-155 |
| 19 | 64155 | 154.9229 | 0   | 0   | 0   | 1   | 5   | 5   | 0   | 16  | Gd-155 |
| 20 | 64156 | 155.9225 | 0   | 0   | 0   | 1   | 5   | 5   | 0   | 16  | Gd-156 |
| 21 | 64157 | 156.9241 | 0   | 0   | 0   | 1   | 5   | 5   | 0   | 16  | Gd-157 |
| 22 | 64158 | 157.9236 | 0   | 0   | 0   | 1   | 5   | 5   | 0   | 16  | Gd-158 |
| 23 | 66160 | 159.9248 | 0   | 0   | 0   | 0   | 3   | 5   | 0   | 16  | Dy-160 |
| 24 | 66161 | 160.9274 | 0   | 0   | 0   | 0   | 3   | 5   | 0   | 16  | Dy-161 |
| 25 | 66162 | 161.9270 | 0   | 0   | 0   | 0   | 3   | 5   | 0   | 16  | Dy-162 |
| 26 | 66163 | 162.9286 | 0   | 0   | 0   | 0   | 3   | 5   | 0   | 16  | Dy-163 |
| 27 | 66164 | 163.9282 | 0   | 0   | 0   | 3   | 3   | 5   | 0   | 16  | Dy-164 |
| 28 | 68166 | 165.9304 | 0   | 0   | 0   | 0   | 3   | 5   | 0   | 16  | Er-166 |
| 29 | 68167 | 166.9320 | 0   | 0   | 0   | 0   | 5   | 5   | 0   | 16  | Er-167 |
| 30 | 72176 | 175.9414 | 0   | 0   | 0   | 1   | 3   | 5   | 0   | 16  | Hf-176 |
| 31 | 72177 | 176.9430 | 0   | 0   | 0   | 1   | 3   | 5   | 0   | 16  | Hf-177 |
| 32 | 72178 | 177.9436 | 0   | 0   | 0   | 1   | 3   | 5   | 0   | 16  | Hf-178 |
| 33 | 72179 | 178.9462 | 0   | 0   | 0   | 1   | 3   | 5   | 0   | 16  | Hf-179 |
| 34 | 72180 | 179.9468 | 0   | 0   | 0   | 1   | 3   | 5   | 0   | 16  | Hf-180 |
| 35 | 74182 | 181.9531 | 0   | 0   | 0   | 1   | 3   | 5   | 0   | 16  | W-182  |
| 36 | 74183 | 182.9517 | 0   | 0   | 0   | 1   | 3   | 5   | 0   | 16  | W-183  |
| 37 | 74184 | 183.9502 | 0   | 0   | 0   | 1   | 3   | 5   | 0   | 16  | W-184  |
| 38 | 74186 | 185.9575 | 0   | 0   | 0   | 1   | 3   | 5   | 0   | 16  | W-186  |



|    |       |          |   |   |    |   |    |   |    |    |        |
|----|-------|----------|---|---|----|---|----|---|----|----|--------|
| 39 | 90232 | 232.0333 | 2 | 0 | 1  | 3 | 5  | 5 | 5  | 16 | Th-232 |
| 40 | 92233 | 233.0450 | 3 | 0 | 2  | 3 | 5  | 5 | 5  | 16 | U-233  |
| 41 | 92235 | 235.0441 | 3 | 0 | 3  | 3 | 5  | 5 | 5  | 16 | U-235  |
| 42 | 92236 | 236.0458 | 3 | 0 | 4  | 3 | 3  | 5 | 3  | 16 | U-236  |
| 43 | 92238 | 238.0510 | 2 | 0 | 5  | 3 | 5  | 5 | 5  | 16 | U-238  |
| 44 | 94238 | 238.0495 | 3 | 0 | 6  | 3 | 3  | 5 | 0  | 16 | Pu-238 |
| 45 | 94239 | 239.0522 | 3 | 0 | 7  | 3 | 5  | 5 | 5  | 16 | Pu-239 |
| 46 | 94240 | 240.0542 | 3 | 0 | 8  | 3 | 5  | 5 | 5  | 16 | Pu-240 |
| 47 | 94241 | 241.0487 | 3 | 0 | 9  | 3 | 5  | 5 | 5  | 16 | Pu-241 |
| 48 | 94242 | 242.0584 | 3 | 0 | 10 | 3 | 3  | 5 | 3  | 16 | Pu-242 |
| 49 | 95241 | 241.0568 | 3 | 0 | 0  | 3 | 3  | 5 | 0  | 16 | Am-241 |
| 50 | 1001  | 1.0078   | 0 | 0 | 0  | 0 | 8  | 0 | 8  | 0  | H-1    |
| 51 | 1002  | 2.0141   | 0 | 0 | 0  | 1 | 5  | 0 | 5  | 0  | H-2    |
| 52 | 1006  | 1.0078   | 0 | 0 | 0  | 0 | 3  | 0 | 3  | 0  | H-1    |
| 53 | 1040  | 1.0078   | 0 | 0 | 0  | 0 | 3  | 0 | 3  | 0  | H-1    |
| 54 | 3006  | 4.0026   | 0 | 0 | 0  | 0 | 5  | 0 | 5  | 0  | Li-6   |
| 55 | 3007  | 6.0151   | 0 | 0 | 0  | 1 | 3  | 0 | 3  | 0  | Li-7   |
| 56 | 4009  | 7.0160   | 0 | 0 | 0  | 1 | 3  | 0 | 0  | 0  | Be-9   |
| 57 | 5000  | 9.0122   | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | B-Nat  |
| 58 | 5010  | 10.8120  | 1 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | B-10   |
| 59 | 5011  | 10.0129  | 1 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | B-11   |
| 60 | 6000  | 11.0093  | 0 | 0 | 0  | 0 | 5  | 0 | 5  | 0  | C-Nat  |
| 61 | 6001  | 12.0011  | 0 | 0 | 0  | 0 | 10 | 0 | 10 | 0  | C-Nat  |
| 62 | 7014  | 12.0011  | 0 | 0 | 0  | 1 | 3  | 0 | 0  | 0  | N-14   |
| 63 | 8001  | 14.0031  | 0 | 0 | 0  | 0 | 5  | 0 | 5  | 0  | O-UO2  |
| 64 | 8016  | 15.9905  | 0 | 0 | 0  | 0 | 5  | 0 | 5  | 0  | O-16   |
| 65 | 9019  | 15.9905  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | F-19   |
| 66 | 11023 | 18.9982  | 0 | 0 | 0  | 0 | 5  | 0 | 0  | 0  | Na-23  |
| 67 | 12000 | 22.9895  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Mg-Nat |
| 68 | 13027 | 24.3051  | 0 | 0 | 0  | 0 | 3  | 0 | 3  | 0  | Al-27  |
| 69 | 14000 | 26.9818  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Si-Nat |
| 70 | 15031 | 28.0853  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | P-31   |
| 71 | 16000 | 30.9741  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | S-Nat  |
| 72 | 17000 | 32.0636  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Cl-Nat |
| 73 | 19000 | 35.4526  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | K-Nat  |
| 74 | 20000 | 39.1019  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Ca-Nat |
| 75 | 22000 | 40.0803  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Ti-Nat |
| 76 | 23000 | 47.8789  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | V-Nat  |
| 77 | 24000 | 50.9416  | 0 | 0 | 0  | 1 | 3  | 0 | 0  | 0  | Cr-Nat |
| 78 | 24050 | 51.9957  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Cr-50  |
| 79 | 24052 | 49.9461  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Cr-52  |
| 80 | 24053 | 51.9402  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Cr-53  |
| 81 | 24054 | 52.9408  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Cr-54  |
| 82 | 25055 | 53.9394  | 0 | 0 | 0  | 1 | 3  | 0 | 0  | 0  | Mn-55  |
| 83 | 26000 | 54.9381  | 0 | 0 | 0  | 1 | 3  | 0 | 0  | 0  | Fe-Nat |
| 84 | 26054 | 55.8446  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Fe-54  |
| 85 | 26056 | 53.9394  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Fe-56  |
| 86 | 26057 | 55.9345  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Fe-57  |
| 87 | 26058 | 56.9351  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Fe-58  |
| 88 | 27059 | 57.9337  | 0 | 0 | 0  | 1 | 3  | 0 | 0  | 0  | Co-59  |
| 89 | 28000 | 58.9332  | 0 | 0 | 0  | 1 | 3  | 0 | 0  | 0  | Ni-Nat |
| 90 | 28058 | 58.6936  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Ni-58  |
| 91 | 28060 | 57.9357  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Ni-60  |
| 92 | 28061 | 59.9308  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Ni-61  |
| 93 | 28062 | 60.9314  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Ni-62  |
| 94 | 28064 | 61.9280  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Ni-64  |
| 95 | 29063 | 63.9282  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Cu-63  |
| 96 | 29065 | 62.9296  | 0 | 0 | 0  | 0 | 3  | 0 | 0  | 0  | Cu-65  |
| 97 | 40000 | 64.9278  | 0 | 0 | 0  | 1 | 3  | 0 | 3  | 0  | Zr-Nat |

|     |       |          |   |     |   |   |   |   |   |   |         |
|-----|-------|----------|---|-----|---|---|---|---|---|---|---------|
| 98  | 40001 | 91.2196  | 0 | 0   | 0 | 1 | 8 | 0 | 8 | 0 | Zr-Zrh2 |
| 99  | 40090 | 91.2196  | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Zr-90   |
| 100 | 40092 | 89.9043  | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Zr-92   |
| 101 | 40094 | 91.9050  | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Zr-94   |
| 102 | 41093 | 93.9063  | 0 | 0   | 0 | 0 | 5 | 0 | 0 | 0 | Nb-93   |
| 103 | 42000 | 92.9032  | 0 | 0   | 0 | 3 | 3 | 0 | 0 | 0 | Mo-Nat  |
| 104 | 44601 | 95.9402  | 1 | 25  | 0 | 0 | 3 | 0 | 0 | 0 | Ru-101  |
| 105 | 45103 | 100.9058 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Rh-103  |
| 106 | 45603 | 102.9050 | 1 | 31  | 0 | 1 | 3 | 0 | 0 | 0 | Rh-103  |
| 107 | 45605 | 104.9062 | 1 | 32  | 0 | 0 | 3 | 0 | 0 | 0 | Rh-105  |
| 108 | 46605 | 104.9052 | 1 | 34  | 0 | 0 | 3 | 0 | 0 | 0 | Pd-105  |
| 109 | 46607 | 106.9054 | 1 | 36  | 0 | 0 | 3 | 0 | 0 | 0 | Pd-107  |
| 110 | 47609 | 108.9045 | 1 | 38  | 0 | 3 | 3 | 0 | 0 | 0 | Ag-109  |
| 111 | 48000 | 112.4258 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Cd-Nat  |
| 112 | 48110 | 109.9031 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Cd-110  |
| 113 | 48111 | 110.9047 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Cd-111  |
| 114 | 48112 | 111.9033 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Cd-112  |
| 115 | 48113 | 112.8999 | 0 | 0   | 0 | 1 | 3 | 0 | 0 | 0 | Cd-113  |
| 116 | 48114 | 113.9035 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Cd-114  |
| 117 | 49000 | 114.9071 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | In-Nat  |
| 118 | 49615 | 114.9071 | 1 | 44  | 0 | 0 | 3 | 0 | 0 | 0 | In-115  |
| 119 | 50000 | 118.7107 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-Nat  |
| 120 | 50112 | 111.9013 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-112  |
| 121 | 50114 | 113.8984 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-114  |
| 122 | 50115 | 114.9031 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-115  |
| 123 | 50116 | 115.9017 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-116  |
| 124 | 50117 | 116.9033 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-117  |
| 125 | 50118 | 117.9018 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-118  |
| 126 | 50119 | 118.9035 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-119  |
| 127 | 50120 | 119.9020 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-120  |
| 128 | 50122 | 121.9032 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-122  |
| 129 | 50124 | 123.9054 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sn-124  |
| 130 | 51000 | 121.7577 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sb-Nat  |
| 131 | 51121 | 120.9036 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sb-121  |
| 132 | 51123 | 122.9038 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sb-123  |
| 133 | 54634 | 133.9053 | 1 | 60  | 0 | 3 | 3 | 0 | 0 | 0 | Xe-134  |
| 134 | 60643 | 142.9097 | 1 | 81  | 0 | 3 | 3 | 0 | 0 | 0 | Nd-143  |
| 135 | 60645 | 144.9129 | 1 | 83  | 0 | 3 | 3 | 0 | 0 | 0 | Nd-145  |
| 136 | 61647 | 146.9151 | 1 | 88  | 0 | 3 | 3 | 0 | 0 | 0 | Pm-147  |
| 137 | 61748 | 147.9177 | 1 | 92  | 0 | 0 | 3 | 0 | 0 | 0 | Pm-148m |
| 138 | 62153 | 152.9217 | 1 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Sm-153  |
| 139 | 62647 | 146.9151 | 1 | 93  | 0 | 3 | 3 | 0 | 0 | 0 | Sm-147  |
| 140 | 62649 | 148.9173 | 1 | 95  | 0 | 3 | 3 | 0 | 0 | 0 | Sm-149  |
| 141 | 62650 | 149.9169 | 1 | 96  | 0 | 1 | 3 | 0 | 0 | 0 | Sm-150  |
| 142 | 62651 | 150.9195 | 1 | 97  | 0 | 3 | 3 | 0 | 0 | 0 | Sm-151  |
| 143 | 62652 | 151.9201 | 1 | 98  | 0 | 3 | 3 | 0 | 0 | 0 | Sm-152  |
| 144 | 63156 | 155.9245 | 1 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Eu-156  |
| 145 | 63157 | 156.9251 | 1 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Eu-157  |
| 146 | 63653 | 152.9217 | 1 | 102 | 0 | 3 | 3 | 0 | 0 | 0 | Eu-153  |
| 147 | 63654 | 153.9223 | 1 | 103 | 0 | 3 | 3 | 0 | 0 | 0 | Eu-154  |
| 148 | 63655 | 154.9229 | 1 | 104 | 0 | 3 | 3 | 0 | 0 | 0 | Eu-155  |
| 149 | 64152 | 151.9201 | 0 | 0   | 0 | 1 | 3 | 0 | 0 | 0 | Gd-152  |
| 150 | 64154 | 153.9213 | 0 | 0   | 0 | 1 | 3 | 0 | 0 | 0 | Gd-154  |
| 151 | 64160 | 159.9268 | 0 | 0   | 0 | 1 | 3 | 0 | 0 | 0 | Gd-160  |
| 152 | 64655 | 154.9229 | 1 | 108 | 0 | 1 | 3 | 0 | 0 | 0 | Gd-155  |
| 153 | 64656 | 155.9225 | 1 | 109 | 0 | 1 | 3 | 0 | 0 | 0 | Gd-156  |
| 154 | 64657 | 156.9241 | 1 | 110 | 0 | 1 | 3 | 0 | 0 | 0 | Gd-157  |
| 155 | 64658 | 157.9236 | 1 | 111 | 0 | 1 | 3 | 0 | 0 | 0 | Gd-158  |
| 156 | 68168 | 167.9414 | 0 | 0   | 0 | 0 | 3 | 0 | 0 | 0 | Er-166  |

|     |       |          |   |    |    |   |   |   |   |   |         |
|-----|-------|----------|---|----|----|---|---|---|---|---|---------|
| 157 | 71176 | 175.9414 | 0 | 0  | 0  | 0 | 3 | 0 | 0 | 0 | Lu-176  |
| 158 | 72174 | 173.9402 | 0 | 0  | 0  | 1 | 3 | 0 | 0 | 0 | Hf-174  |
| 159 | 73181 | 180.9545 | 0 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Ta-181  |
| 160 | 73182 | 181.9500 | 0 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Ta-182  |
| 161 | 74000 | 183.8564 | 0 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | W-Nat   |
| 162 | 79197 | 196.9660 | 0 | 0  | 0  | 0 | 3 | 0 | 0 | 0 | Au-197  |
| 163 | 82206 | 205.9694 | 0 | 0  | 0  | 0 | 3 | 0 | 0 | 0 | Pb-206  |
| 164 | 82207 | 206.9780 | 0 | 0  | 0  | 0 | 3 | 0 | 0 | 0 | Pb-207  |
| 165 | 82208 | 207.9766 | 0 | 0  | 0  | 0 | 3 | 0 | 0 | 0 | Pb-208  |
| 166 | 83209 | 208.9803 | 0 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Bi-209  |
| 167 | 90230 | 230.0361 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Th-230  |
| 168 | 91231 | 231.0347 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Pa-231  |
| 169 | 91233 | 233.0399 | 3 | 0  | 0  | 3 | 5 | 0 | 0 | 0 | Pa-233  |
| 170 | 92232 | 232.0333 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | U-232   |
| 171 | 92234 | 234.0405 | 3 | 0  | 11 | 3 | 3 | 0 | 3 | 0 | U-234   |
| 172 | 92237 | 237.0484 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | U-237   |
| 173 | 93237 | 237.0482 | 3 | 0  | 12 | 3 | 3 | 0 | 0 | 0 | Np-237  |
| 174 | 93238 | 238.0510 | 3 | 0  | 0  | 0 | 3 | 0 | 0 | 0 | Np-238  |
| 175 | 93239 | 239.0526 | 3 | 0  | 13 | 3 | 3 | 0 | 0 | 0 | Np-239  |
| 176 | 94236 | 236.0458 | 3 | 0  | 14 | 3 | 3 | 0 | 0 | 0 | Pu-236  |
| 177 | 95242 | 242.0594 | 3 | 0  | 0  | 0 | 3 | 0 | 0 | 0 | Am-242  |
| 178 | 95342 | 242.0594 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Am-242m |
| 179 | 95243 | 243.0610 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Am-243  |
| 180 | 96242 | 242.0584 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cm-242  |
| 181 | 96243 | 243.0610 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cm-243  |
| 182 | 96244 | 244.0626 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cm-244  |
| 183 | 96245 | 245.0652 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cm-245  |
| 184 | 96246 | 246.0668 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cm-246  |
| 185 | 96247 | 247.0725 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cm-247  |
| 186 | 96248 | 248.0721 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cm-248  |
| 187 | 97249 | 249.0797 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Bk-249  |
| 188 | 98249 | 249.0797 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cf-249  |
| 189 | 98250 | 250.0763 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cf-250  |
| 190 | 98251 | 251.0799 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cf-251  |
| 191 | 98252 | 252.0815 | 3 | 0  | 0  | 3 | 3 | 0 | 0 | 0 | Cf-252  |
| 192 | 1003  | 3.0160   | 0 | 0  | 0  | 0 | 3 | 0 | 0 | 0 | T-3     |
| 193 | 2003  | 3.0149   | 0 | 0  | 0  | 0 | 3 | 0 | 0 | 0 | He-3    |
| 194 | 35581 | 80.9163  | 1 | 1  | 0  | 0 | 1 | 0 | 0 | 0 | Br-81   |
| 195 | 36582 | 81.9135  | 1 | 2  | 0  | 1 | 1 | 0 | 0 | 0 | Kr-82   |
| 196 | 36583 | 82.9141  | 1 | 3  | 0  | 3 | 1 | 0 | 0 | 0 | Kr-83   |
| 197 | 36584 | 83.9114  | 1 | 4  | 0  | 1 | 1 | 0 | 0 | 0 | Kr-84   |
| 198 | 36585 | 84.9125  | 1 | 5  | 0  | 0 | 1 | 0 | 0 | 0 | Kr-85   |
| 199 | 36586 | 85.9106  | 1 | 6  | 0  | 3 | 1 | 0 | 0 | 0 | Kr-86   |
| 200 | 38589 | 88.9075  | 1 | 7  | 0  | 0 | 1 | 0 | 0 | 0 | Sr-89   |
| 201 | 38590 | 89.9078  | 1 | 8  | 0  | 0 | 1 | 0 | 0 | 0 | Sr-90   |
| 202 | 39589 | 88.9058  | 1 | 9  | 0  | 1 | 1 | 0 | 0 | 0 | Y-89    |
| 203 | 39590 | 89.9072  | 1 | 10 | 0  | 0 | 1 | 0 | 0 | 0 | Y-90    |
| 204 | 39591 | 90.9074  | 1 | 11 | 0  | 0 | 1 | 0 | 0 | 0 | Y-91    |
| 205 | 40591 | 90.9029  | 1 | 12 | 0  | 1 | 1 | 0 | 0 | 0 | Zr-91   |
| 206 | 40593 | 92.9064  | 1 | 13 | 0  | 0 | 1 | 0 | 0 | 0 | Zr-93   |
| 207 | 40595 | 94.9080  | 1 | 14 | 0  | 0 | 1 | 0 | 0 | 0 | Zr-95   |
| 208 | 40596 | 95.9049  | 1 | 15 | 0  | 1 | 1 | 0 | 0 | 0 | Zr-96   |
| 209 | 41595 | 94.9068  | 1 | 16 | 0  | 0 | 1 | 0 | 0 | 0 | Nb-95   |
| 210 | 42095 | 94.9059  | 0 | 0  | 0  | 0 | 1 | 0 | 0 | 0 | Mo-95   |
| 211 | 42596 | 95.9047  | 1 | 18 | 0  | 0 | 3 | 0 | 0 | 0 | Mo-96   |
| 212 | 42597 | 96.9061  | 1 | 19 | 0  | 0 | 1 | 0 | 0 | 0 | Mo-97   |
| 213 | 42598 | 97.9055  | 1 | 20 | 0  | 0 | 1 | 0 | 0 | 0 | Mo-98   |
| 214 | 42599 | 98.9078  | 1 | 21 | 0  | 0 | 1 | 0 | 0 | 0 | Mo-99   |
| 215 | 42600 | 99.9075  | 1 | 22 | 0  | 0 | 1 | 0 | 0 | 0 | Mo-100  |

|     |       |          |   |     |   |   |   |   |   |   |         |
|-----|-------|----------|---|-----|---|---|---|---|---|---|---------|
| 216 | 44600 | 99.9042  | 1 | 24  | 0 | 0 | 1 | 0 | 0 | 0 | Ru-100  |
| 217 | 44602 | 101.9044 | 1 | 26  | 0 | 0 | 1 | 0 | 0 | 0 | Ru-102  |
| 218 | 44603 | 102.9060 | 1 | 27  | 0 | 0 | 1 | 0 | 0 | 0 | Ru-103  |
| 219 | 44604 | 103.9056 | 1 | 28  | 0 | 0 | 1 | 0 | 0 | 0 | Ru-104  |
| 220 | 44605 | 104.9082 | 1 | 29  | 0 | 0 | 1 | 0 | 0 | 0 | Ru-105  |
| 221 | 44606 | 105.9078 | 1 | 30  | 0 | 0 | 1 | 0 | 0 | 0 | Ru-106  |
| 222 | 46604 | 103.9036 | 1 | 33  | 0 | 0 | 1 | 0 | 0 | 0 | Pd-104  |
| 223 | 46606 | 105.9038 | 1 | 35  | 0 | 0 | 1 | 0 | 0 | 0 | Pd-106  |
| 224 | 47710 | 109.9061 | 1 | 39  | 0 | 0 | 1 | 0 | 0 | 0 | Ag-110m |
| 225 | 47611 | 110.9057 | 1 | 40  | 0 | 0 | 1 | 0 | 0 | 0 | Ag-111  |
| 226 | 48610 | 109.9031 | 1 | 41  | 0 | 0 | 1 | 0 | 0 | 0 | Cd-110  |
| 227 | 48611 | 110.9047 | 1 | 42  | 0 | 0 | 1 | 0 | 0 | 0 | Cd-111  |
| 228 | 48613 | 112.8999 | 1 | 43  | 0 | 1 | 1 | 0 | 0 | 0 | Cd-113  |
| 229 | 50125 | 124.9080 | 0 | 0   | 0 | 0 | 1 | 0 | 0 | 0 | Sn-125  |
| 230 | 51621 | 120.9087 | 1 | 45  | 0 | 0 | 1 | 0 | 0 | 0 | Sb-121  |
| 231 | 51625 | 124.9050 | 1 | 46  | 0 | 0 | 1 | 0 | 0 | 0 | Sb-125  |
| 232 | 51627 | 126.9069 | 1 | 47  | 0 | 0 | 1 | 0 | 0 | 0 | Sb-127  |
| 233 | 52727 | 126.9052 | 1 | 48  | 0 | 0 | 1 | 0 | 0 | 0 | Te-127m |
| 234 | 52729 | 128.9074 | 1 | 49  | 0 | 0 | 1 | 0 | 0 | 0 | Te-129m |
| 235 | 52632 | 131.9082 | 1 | 50  | 0 | 0 | 1 | 0 | 0 | 0 | Te-132  |
| 236 | 53627 | 126.9042 | 1 | 51  | 0 | 3 | 1 | 0 | 0 | 0 | I-127   |
| 237 | 53629 | 128.9054 | 1 | 52  | 0 | 0 | 1 | 0 | 0 | 0 | I-129   |
| 238 | 53631 | 130.9066 | 1 | 53  | 0 | 0 | 1 | 0 | 0 | 0 | I-131   |
| 239 | 53635 | 134.9100 | 1 | 54  | 0 | 0 | 1 | 0 | 0 | 0 | I-135   |
| 240 | 54628 | 127.9038 | 1 | 55  | 0 | 3 | 1 | 0 | 0 | 0 | Xe-128  |
| 241 | 54630 | 129.9039 | 1 | 56  | 0 | 3 | 1 | 0 | 0 | 0 | Xe-130  |
| 242 | 54632 | 131.9041 | 1 | 58  | 0 | 3 | 1 | 0 | 0 | 0 | Xe-132  |
| 243 | 54633 | 132.9057 | 1 | 59  | 0 | 0 | 1 | 0 | 0 | 0 | Xe-133  |
| 244 | 54635 | 134.9069 | 1 | 61  | 0 | 0 | 1 | 0 | 0 | 0 | Xe-135  |
| 245 | 54636 | 135.9075 | 1 | 62  | 0 | 3 | 1 | 0 | 0 | 0 | Xe-136  |
| 246 | 55634 | 133.9073 | 1 | 64  | 0 | 0 | 1 | 0 | 0 | 0 | Cs-134  |
| 247 | 55635 | 134.9059 | 1 | 65  | 0 | 0 | 1 | 0 | 0 | 0 | Cs-135  |
| 248 | 55636 | 135.9075 | 1 | 66  | 0 | 0 | 1 | 0 | 0 | 0 | Cs-136  |
| 249 | 55637 | 136.9071 | 1 | 67  | 0 | 0 | 1 | 0 | 0 | 0 | Cs-137  |
| 250 | 56634 | 133.9043 | 1 | 68  | 0 | 0 | 1 | 0 | 0 | 0 | Ba-134  |
| 251 | 56637 | 136.9051 | 1 | 69  | 0 | 0 | 1 | 0 | 0 | 0 | Ba-137  |
| 252 | 56640 | 139.9109 | 1 | 70  | 0 | 0 | 1 | 0 | 0 | 0 | Ba-140  |
| 253 | 57639 | 138.9063 | 1 | 71  | 0 | 0 | 1 | 0 | 0 | 0 | La-139  |
| 254 | 57640 | 139.9099 | 1 | 72  | 0 | 0 | 1 | 0 | 0 | 0 | La-140  |
| 255 | 58640 | 139.9059 | 1 | 73  | 0 | 0 | 1 | 0 | 0 | 0 | Ce-140  |
| 256 | 58641 | 140.9085 | 1 | 74  | 0 | 0 | 1 | 0 | 0 | 0 | Ce-141  |
| 257 | 58642 | 141.9091 | 1 | 75  | 0 | 0 | 1 | 0 | 0 | 0 | Ce-142  |
| 258 | 58643 | 142.9127 | 1 | 76  | 0 | 0 | 1 | 0 | 0 | 0 | Ce-143  |
| 259 | 58644 | 143.9133 | 1 | 77  | 0 | 0 | 1 | 0 | 0 | 0 | Ce-144  |
| 260 | 59641 | 140.9075 | 1 | 78  | 0 | 3 | 1 | 0 | 0 | 0 | Pr-141  |
| 261 | 59643 | 142.9107 | 1 | 79  | 0 | 0 | 1 | 0 | 0 | 0 | Pr-143  |
| 262 | 60642 | 141.9081 | 1 | 80  | 0 | 0 | 1 | 0 | 0 | 0 | Nd-142  |
| 263 | 60644 | 143.9103 | 1 | 82  | 0 | 0 | 1 | 0 | 0 | 0 | Nd-144  |
| 264 | 60646 | 145.9135 | 1 | 84  | 0 | 3 | 1 | 0 | 0 | 0 | Nd-146  |
| 265 | 60647 | 146.9161 | 1 | 85  | 0 | 0 | 1 | 0 | 0 | 0 | Nd-147  |
| 266 | 60648 | 147.9167 | 1 | 86  | 0 | 3 | 1 | 0 | 0 | 0 | Nd-148  |
| 267 | 60650 | 149.9209 | 1 | 87  | 0 | 3 | 1 | 0 | 0 | 0 | Nd-150  |
| 268 | 61648 | 147.9177 | 1 | 89  | 0 | 0 | 1 | 0 | 0 | 0 | Pm-148  |
| 269 | 61649 | 148.9183 | 1 | 90  | 0 | 0 | 1 | 0 | 0 | 0 | Pm-149  |
| 270 | 61651 | 150.9215 | 1 | 91  | 0 | 0 | 1 | 0 | 0 | 0 | Pm-151  |
| 271 | 62648 | 147.9147 | 1 | 94  | 0 | 0 | 1 | 0 | 0 | 0 | Sm-148  |
| 272 | 62653 | 152.9217 | 1 | 99  | 0 | 0 | 1 | 0 | 0 | 0 | Sm-153  |
| 273 | 62654 | 153.9223 | 1 | 100 | 0 | 0 | 1 | 0 | 0 | 0 | Sm-154  |
| 274 | 63651 | 150.9195 | 1 | 101 | 0 | 3 | 1 | 0 | 0 | 0 | Eu-151  |

|     |       |          |   |     |   |   |   |   |   |   |        |
|-----|-------|----------|---|-----|---|---|---|---|---|---|--------|
| 275 | 63656 | 155.9245 | 1 | 105 | 0 | 0 | 1 | 0 | 0 | 0 | Eu-156 |
| 276 | 63657 | 156.9251 | 1 | 106 | 0 | 0 | 1 | 0 | 0 | 0 | Eu-157 |
| 277 | 64654 | 153.9213 | 1 | 107 | 0 | 1 | 1 | 0 | 0 | 0 | Gd-154 |
| 278 | 64660 | 159.9268 | 1 | 112 | 0 | 1 | 1 | 0 | 0 | 0 | Gd-160 |
| 279 | 65159 | 158.9252 | 0 | 0   | 0 | 0 | 1 | 0 | 0 | 0 | Tb-159 |
| 280 | 65160 | 158.9273 | 1 | 0   | 0 | 0 | 1 | 0 | 0 | 0 | Tb-160 |
| 281 | 65161 | 160.9276 | 1 | 0   | 0 | 0 | 1 | 0 | 0 | 0 | Tb-161 |
| 282 | 65659 | 158.9252 | 1 | 113 | 0 | 0 | 1 | 0 | 0 | 0 | Tb-159 |
| 283 | 65660 | 158.9273 | 1 | 114 | 0 | 0 | 1 | 0 | 0 | 0 | Tb-160 |
| 284 | 65661 | 160.9276 | 1 | 115 | 0 | 0 | 1 | 0 | 0 | 0 | Tb-161 |
| 285 | 66660 | 159.9248 | 1 | 116 | 0 | 0 | 1 | 0 | 0 | 0 | Dy-160 |
| 286 | 66661 | 160.9274 | 1 | 117 | 0 | 0 | 1 | 0 | 0 | 0 | Dy-161 |
| 287 | 66662 | 161.9270 | 1 | 118 | 0 | 0 | 1 | 0 | 0 | 0 | Dy-162 |
| 288 | 66663 | 162.9286 | 1 | 119 | 0 | 0 | 1 | 0 | 0 | 0 | Dy-163 |
| 289 | 66664 | 163.9282 | 1 | 120 | 0 | 3 | 1 | 0 | 0 | 0 | Dy-164 |
| 290 | 67165 | 164.9298 | 0 | 0   | 0 | 3 | 1 | 0 | 0 | 0 | Ho-165 |
| 291 | 67665 | 164.9298 | 1 | 121 | 0 | 3 | 1 | 0 | 0 | 0 | Ho-165 |
| 292 | 77191 | 190.9781 | 0 | 0   | 0 | 0 | 1 | 0 | 0 | 0 | Ir-191 |
| 293 | 77193 | 192.9803 | 0 | 0   | 0 | 0 | 1 | 0 | 0 | 0 | Ir-193 |
| 294 | 91232 | 232.0386 | 1 | 0   | 0 | 0 | 1 | 0 | 0 | 0 | Pa-232 |
| 295 | 91234 | 234.0433 | 1 | 0   | 0 | 0 | 1 | 0 | 0 | 0 | Pa-234 |

## Appendix C.2 DECLIB input to generate the ENDF/B-7.1 V4.0 MPACT 47-group library

```

%TITL
DATE::FEBRUARY 26, 2015; GROUP::47; MASTER::ENDF/B-VII R1; SUBGROUP::YES; TRC::YES
%IOPT
! LIBRARY VERSION
VER 4.0
! 0: Unformatted master library -> Formatted master library
! 1: Formatted master library -> Unformatted master library
! 2: Unformatted master library -> DeCART library
! 3: Formatted master library -> DeCART library
LIB 2
! 0: lambda*sig-p is not included in sig-b
! 1: lambda*sig-p is included in sig-b
BXS 0
! 0: No transport corrected total XS
! 1: Transport corrected total XS
! 2: Transport correction is applied to pl scattering matrix only for testing.
XST 1
%FILE
AMP ..\lib\ampx47g_71_07202014_02m_ykk.bin lib47_HetFfactors_VII.1.fmt
DEC mpact47g_71s_v4.1m1_03102015.fmt mpact47g_71s_v4.1m1_03102015.bin
END ..\lib\n-endfb7.1
DCY ..\lib\dec-endfb7.1
FPY ..\lib\fpv-endfb7.1
SUB ..\lib\subgr_47g_71s_02252015_merge.sub !Subgroup data
TRC ..\lib\trcorr_47g_71_10242014_5n.dat !Transport correction
XSB ..\lib\backxs_47g_03022015_final.dat !Background XS
DLY ..\lib\totbeta_e71_10062014.dat !Transient data
DBR ..\lib\subgr_47g_71s_10312014_u238epi.sub !Epithermal upscattering
%COLL
FLX ..\lib\fluxcur_47g_fa.dat
SUB e:\ornl\declib\lib\subgr.sub_moc49g_mod
LAM e:\ornl\declib\lib\rilamb.irp_e705_49g
%RESO !# of fast / end of reso.
GRP 9 26
%NUCL
! NRES NELR NELT NYLD
49 191 295 121
! 48 191 295 121
!
! -----
! NID NUCLIDE ID 1000*Z+500*A+100*B+A
! A=0: ACTIVATION & HEAVY NUCLIDES / A=1: F.P. NUCLIDES
! B=0: STABLE / B=1: METASTABLE
! AMASS ATOMIC MASS
! AID ALPHANUMERIC NUCLIDE ID.
! NO0 1/2/3 COLLAPSING SPECTRA (MODERATOR/FUEL/STRUCTURE)
! NO1 0 NON DEPLETABLE
! >0 DEPLETABLE (W/ DECAY CONSTANT)
! 1
! 2 RI-A ONLY (FISSIONABLE W/ KAPPA, BETA, FPY)
! 3 RI-A & RI-NF (FISSIONABLE W/ KAPPA, BETA, FPY)
! NO2 NPPY (ORDERING NO. OF FPY) (+/- : CUMULATIVE/DIRECT YIELD)
! NO3 NPI (WITH P1/P2/P3)
! NO4 NCHIX (WITH FISSION SPECTRA)
! NO5 N2N (WITH N2N)
! NO6 N3N (WITH N3N)
! NO7 EPU (WITH EPITHERMAL UPSCATTERING RESONANCE DATA)
! NO8 TRC (TRANSPORT CORRECTION) (0/1/2 : OUTSCATT/NLC/INSCATT)
! AMPX # OF AMPX NUCLIDES TO BE MERGED
! #T # OF TEMPERATURES
!
! -----
! NO NID AMASS AID NO0 NO1 NO2 NO3 NO4 NO5 NO6 NO7 NO8 AMPX #T TEMPERATURE(K)
!
1 40091 90.9056 'Zr-91 ' 3 1 0 0 0 0 0 0 0 0 0 0 1 3 293 900 2000
1 40091 100.0
2 40096 95.9083 'Zr-96 ' 3 1 0 0 0 0 0 0 0 0 0 0 1 3 293 900 2000
1 40096 100.0
3 42595 94.9059 'Mo-95 ' 2 1 17 0 0 0 0 0 0 0 0 0 1 3 293 900 2000
1 42095 100.0
4 43599 99.0005 'Tc-99 ' 2 1 23 0 0 2 0 0 0 0 1 3 293 900 2000
1 43099 100.0
5 45103 102.9050 'Rh-103 ' 3 1 0 0 0 0 0 0 0 0 1 3 293 900 2000
1 45103 100.0
6 46608 107.9039 'Pd-108 ' 2 1 37 0 0 0 0 0 0 0 1 3 293 900 2000
1 46108 100.0
7 47107 106.9054 'Ag-107 ' 3 0 0 0 0 0 3 1 0 0 1 3 293 900 2000
1 47107 100.0
8 47109 108.9045 'Ag-109 ' 3 0 0 0 0 0 4 2 0 0 1 3 293 900 2000
1 47109 100.0
9 49113 112.9039 'In-113 ' 3 0 0 0 0 0 0 0 0 0 1 3 293 900 2000
1 49113 100.0
10 49115 114.9041 'In-115 ' 3 0 0 0 0 0 0 0 0 0 1 3 293 900 2000
1 49115 100.0
11 54631 130.9055 'Xe-131 ' 2 1 57 0 0 5 3 0 0 0 1 3 293 900 2000
1 54131 100.0
12 55633 132.9057 'Cs-133 ' 2 1 63 0 0 6 0 0 0 0 1 3 293 900 2000
1 55133 100.0
13 62152 151.9201 'Sm-152 ' 3 0 0 0 0 0 7 4 0 0 1 5 293 600 900 1200 2400
1 62152 100.0
14 63151 150.9195 'Eu-151 ' 3 0 0 0 0 0 8 5 0 0 1 5 293 600 900 1200 2400
1 63151 100.0

```

|                                          |       |          |           |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
|------------------------------------------|-------|----------|-----------|---|---|---|----|----|----|----|---|---|---|---|-----|-------|------|------|------|-----|------|------|-----|-----|
| 15                                       | 63152 | 151.9251 | 'Eu-152 ' | 3 | 1 | 0 | 0  | 0  | 9  | 6  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 63152    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 16                                       | 63153 | 152.9217 | 'Eu-153 ' | 3 | 0 | 0 | 0  | 0  | 10 | 7  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 63153    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 17                                       | 63154 | 153.9223 | 'Eu-154 ' | 3 | 1 | 0 | 0  | 0  | 11 | 8  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 63154    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 18                                       | 63155 | 154.9229 | 'Eu-155 ' | 3 | 1 | 0 | 0  | 0  | 12 | 9  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 63155    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 19                                       | 64155 | 154.9229 | 'Gd-155 ' | 2 | 0 | 0 | 0  | 0  | 13 | 0  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 64155    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 20                                       | 64156 | 155.9225 | 'Gd-156 ' | 2 | 0 | 0 | 0  | 0  | 14 | 0  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 64156    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 21                                       | 64157 | 156.9241 | 'Gd-157 ' | 2 | 0 | 0 | 0  | 0  | 15 | 0  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 64157    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 22                                       | 64158 | 157.9236 | 'Gd-158 ' | 2 | 0 | 0 | 0  | 0  | 16 | 0  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 64158    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 23                                       | 66160 | 159.9248 | 'Dy-160 ' | 3 | 0 | 0 | 0  | 0  | 0  | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 66160    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 24                                       | 66161 | 160.9274 | 'Dy-161 ' | 3 | 0 | 0 | 0  | 0  | 0  | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 66161    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 25                                       | 66162 | 161.9270 | 'Dy-162 ' | 3 | 0 | 0 | 0  | 0  | 0  | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 66162    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 26                                       | 66163 | 162.9286 | 'Dy-163 ' | 3 | 0 | 0 | 0  | 0  | 0  | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 66163    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 27                                       | 66164 | 163.9282 | 'Dy-164 ' | 3 | 0 | 0 | 0  | 0  | 17 | 10 | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 66164    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 28                                       | 68166 | 165.9304 | 'Er-166 ' | 2 | 0 | 0 | 0  | 0  | 0  | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 68166    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 29                                       | 68167 | 166.9320 | 'Er-167 ' | 2 | 0 | 0 | 0  | 0  | 0  | 0  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 68167    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 30                                       | 72176 | 175.9414 | 'Hf-176 ' | 3 | 0 | 0 | 0  | 0  | 18 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 72176    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 31                                       | 72177 | 176.9430 | 'Hf-177 ' | 3 | 0 | 0 | 0  | 0  | 19 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 72177    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 32                                       | 72178 | 177.9436 | 'Hf-178 ' | 3 | 0 | 0 | 0  | 0  | 20 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 72178    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 33                                       | 72179 | 178.9462 | 'Hf-179 ' | 3 | 0 | 0 | 0  | 0  | 21 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 72179    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 34                                       | 72180 | 179.9468 | 'Hf-180 ' | 3 | 0 | 0 | 0  | 0  | 22 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 72180    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| ----- Newly added on 05/22/2013(b) ----- |       |          |           |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 35                                       | 74182 | 181.9531 | 'W-182 '  | 3 | 0 | 0 | 0  | 0  | 23 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 74182    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 36                                       | 74183 | 182.9517 | 'W-183 '  | 3 | 0 | 0 | 0  | 0  | 24 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 74183    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 37                                       | 74184 | 183.9502 | 'W-184 '  | 3 | 0 | 0 | 0  | 0  | 25 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 74184    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 38                                       | 74186 | 185.9575 | 'W-186 '  | 3 | 0 | 0 | 0  | 0  | 26 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 74186    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| ----- Newly added on 05/22/2013(e) ----- |       |          |           |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 39                                       | 90232 | 232.0333 | 'Th-232 ' | 2 | 2 | 0 | 18 | 1  | 27 | 11 | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 90232    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 40                                       | 92233 | 233.0450 | 'U-233 '  | 2 | 3 | 0 | 19 | 2  | 28 | 12 | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 92233    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 41                                       | 92235 | 235.0441 | 'U-235 '  | 2 | 3 | 0 | 21 | 3  | 29 | 13 | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 92235    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 42                                       | 92236 | 236.0458 | 'U-236 '  | 2 | 3 | 0 | 22 | 4  | 30 | 14 | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 92236    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 43                                       | 92238 | 238.0510 | 'U-238 '  | 2 | 2 | 0 | 23 | 5  | 31 | 15 | 1 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 92238    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 44                                       | 94238 | 238.0495 | 'Pu-238 ' | 2 | 3 | 0 | 0  | 6  | 32 | 16 | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 94238    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 45                                       | 94239 | 239.0522 | 'Pu-239 ' | 2 | 3 | 0 | 24 | 7  | 33 | 17 | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 94239    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 46                                       | 94240 | 240.0542 | 'Pu-240 ' | 2 | 3 | 0 | 25 | 8  | 34 | 18 | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 94240    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 47                                       | 94241 | 241.0487 | 'Pu-241 ' | 2 | 3 | 0 | 26 | 9  | 35 | 19 | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 94241    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 48                                       | 94242 | 242.0584 | 'Pu-242 ' | 2 | 3 | 0 | 27 | 10 | 36 | 20 | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 94242    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 49                                       | 95241 | 241.0568 | 'Am-241 ' | 2 | 3 | 0 | 0  | 0  | 37 | 21 | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 95241    | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| -----                                    |       |          |           |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 50                                       | 1001  | 1.0078   | 'H-1 '    | 1 | 0 | 0 | 1  | 0  | 0  | 0  | 0 | 0 | 1 | 1 | 9   | 293.6 | 350  | 400  | 450  | 500 | 550  | 600  | 650 | 800 |
|                                          | 1     | 1001     | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 51                                       | 1002  | 2.0141   | 'H-2 '    | 1 | 0 | 0 | 2  | 0  | 38 | 0  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 1002     | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 52                                       | 1006  | 1.0078   | 'H-1 '    | 1 | 0 | 0 | 3  | 0  | 0  | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 5001001  | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 53                                       | 1040  | 1.0078   | 'H-1 '    | 1 | 0 | 0 | 4  | 0  | 0  | 0  | 0 | 0 | 1 | 3 | 296 | 400   | 500  | 600  | 700  | 800 | 1000 | 1200 |     |     |
|                                          | 1     | 7001001  | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 54                                       | 2004  | 4.0026   | 'He-4 '   | 1 | 0 | 0 | 0  | 0  | 0  | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 2004     | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 55                                       | 3006  | 6.0151   | 'Li-6 '   | 1 | 0 | 0 | 7  | 0  | 0  | 0  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 3006     | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 56                                       | 3007  | 7.0160   | 'Li-7 '   | 1 | 0 | 0 | 8  | 0  | 39 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 3007     | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 57                                       | 4009  | 9.0122   | 'Be-9 '   | 1 | 0 | 0 | 0  | 0  | 40 | 0  | 0 | 0 | 1 | 3 | 293 | 900   | 2000 |      |      |     |      |      |     |     |
|                                          | 1     | 4009     | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 58                                       | 5000  | 10.8120  | 'B-nat '  | 1 | 0 | 0 | 28 | 0  | 0  | 0  | 0 | 0 | 2 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 5010     | 19.90     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
|                                          | 2     | 5011     | 80.10     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 59                                       | 5010  | 10.0129  | 'B-10 '   | 1 | 1 | 0 | 29 | 0  | 0  | 0  | 0 | 0 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |
|                                          | 1     | 5010     | 100.0     |   |   |   |    |    |    |    |   |   |   |   |     |       |      |      |      |     |      |      |     |     |
| 60                                       | 5011  | 11.0093  | 'B-11 '   | 1 | 1 | 0 | 30 | 0  | 0  | 0  | 0 | 2 | 1 | 5 | 293 | 600   | 900  | 1200 | 2400 |     |      |      |     |     |







|     |       |          |           |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
|-----|-------|----------|-----------|---|---|-----|---|---|----|----|---|---|---|---|---|-----|-----|---------------|
| 124 | 50117 | 116.9033 | 'Sn-117 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 50117    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 125 | 50118 | 117.9018 | 'Sn-118 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 50118    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 126 | 50119 | 118.9035 | 'Sn-119 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 50119    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 127 | 50120 | 119.9020 | 'Sn-120 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 50120    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 128 | 50122 | 121.9032 | 'Sn-122 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 50122    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 129 | 50124 | 123.9054 | 'Sn-124 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 50124    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 130 | 51000 | 121.7577 | 'Sb-nat ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 2 | 3 | 293 | 900 | 2000          |
|     | 1     | 51121    | 57.210    |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
|     | 2     | 51123    | 42.790    |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 131 | 51121 | 120.9036 | 'Sb-121 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 51121    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 132 | 51123 | 122.9038 | 'Sb-123 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 51123    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 133 | 54634 | 133.9053 | 'Xe-134 ' | 2 | 1 | 60  | 0 | 0 | 53 | 24 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 54134    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 134 | 60643 | 142.9097 | 'Nd-143 ' | 2 | 1 | 81  | 0 | 0 | 54 | 25 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 60143    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 135 | 60645 | 144.9129 | 'Nd-145 ' | 2 | 1 | 83  | 0 | 0 | 55 | 26 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 60145    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 136 | 61647 | 146.9151 | 'Pm-147 ' | 2 | 1 | 88  | 0 | 0 | 56 | 27 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 61147    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 137 | 61748 | 147.9177 | 'Pm-148m' | 2 | 1 | 92  | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 1061148  | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 138 | 62153 | 152.9217 | 'Sm-153 ' | 3 | 1 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 62153    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 139 | 62647 | 146.9151 | 'Sm-147 ' | 2 | 1 | 93  | 0 | 0 | 57 | 28 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 62147    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 140 | 62649 | 148.9173 | 'Sm-149 ' | 2 | 1 | 95  | 0 | 0 | 58 | 29 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 62149    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 141 | 62650 | 149.9169 | 'Sm-150 ' | 2 | 1 | 96  | 0 | 0 | 59 | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 62150    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 142 | 62651 | 150.9195 | 'Sm-151 ' | 2 | 1 | 97  | 0 | 0 | 60 | 30 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 62151    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 143 | 62652 | 151.9201 | 'Sm-152 ' | 2 | 1 | 98  | 0 | 0 | 61 | 31 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 62152    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 144 | 63156 | 155.9245 | 'Eu-156 ' | 3 | 1 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 63156    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 145 | 63157 | 156.9251 | 'Eu-157 ' | 3 | 1 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 63157    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 146 | 63653 | 152.9217 | 'Eu-153 ' | 2 | 1 | 102 | 0 | 0 | 62 | 32 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 63153    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 147 | 63654 | 153.9223 | 'Eu-154 ' | 2 | 1 | 103 | 0 | 0 | 63 | 33 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 63154    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 148 | 63655 | 154.9229 | 'Eu-155 ' | 2 | 1 | 104 | 0 | 0 | 64 | 34 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 63155    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 149 | 64152 | 151.9201 | 'Gd-152 ' | 3 | 0 | 0   | 0 | 0 | 65 | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 64152    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 150 | 64154 | 153.9213 | 'Gd-154 ' | 3 | 0 | 0   | 0 | 0 | 66 | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 64154    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 151 | 64160 | 159.9268 | 'Gd-160 ' | 3 | 0 | 0   | 0 | 0 | 67 | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 64160    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 152 | 64655 | 154.9229 | 'Gd-155 ' | 2 | 1 | 108 | 0 | 0 | 68 | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 64155    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 153 | 64656 | 155.9225 | 'Gd-156 ' | 2 | 1 | 109 | 0 | 0 | 69 | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 64156    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 154 | 64657 | 156.9241 | 'Gd-157 ' | 2 | 1 | 110 | 0 | 0 | 70 | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 64157    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 155 | 64658 | 157.9236 | 'Gd-158 ' | 2 | 1 | 111 | 0 | 0 | 71 | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 64158    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 156 | 68168 | 167.9414 | 'Er-166 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 68168    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 157 | 71176 | 175.9414 | 'Lu-176 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 71176    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 158 | 72174 | 173.9402 | 'Hf-174 ' | 3 | 0 | 0   | 0 | 0 | 72 | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 72174    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 159 | 73181 | 180.9545 | 'Ta-181 ' | 3 | 0 | 0   | 0 | 0 | 73 | 35 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 73181    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 160 | 73182 | 181.9500 | 'Ta-182 ' | 3 | 0 | 0   | 0 | 0 | 74 | 36 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 73182    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 161 | 74000 | 183.8564 | 'W-nat '  | 3 | 0 | 0   | 0 | 0 | 75 | 37 | 0 | 0 | 0 | 4 | 3 | 293 | 900 | 2000          |
|     | 1     | 74182    | 26.62     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
|     | 2     | 74183    | 14.31     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
|     | 3     | 74184    | 30.64     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
|     | 4     | 74186    | 28.43     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 162 | 79197 | 196.9660 | 'Au-197 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 79197    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 163 | 82206 | 205.9694 | 'Pb-206 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 82206    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 164 | 82207 | 206.9780 | 'Pb-207 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 82207    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 165 | 82208 | 207.9766 | 'Pb-208 ' | 3 | 0 | 0   | 0 | 0 | 0  | 0  | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 82208    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 166 | 83209 | 208.9803 | 'Bi-209 ' | 3 | 0 | 0   | 0 | 0 | 76 | 38 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 83209    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 167 | 90230 | 230.0361 | 'Th-230 ' | 2 | 3 | 0   | 0 | 0 | 77 | 39 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 90230    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 168 | 91231 | 231.0347 | 'Pa-231 ' | 2 | 3 | 0   | 0 | 0 | 78 | 40 | 0 | 0 | 0 | 1 | 3 | 293 | 900 | 2000          |
|     | 1     | 91231    | 100.0     |   |   |     |   |   |    |    |   |   |   |   |   |     |     |               |
| 169 | 91233 | 233.0399 | 'Pa-233 ' | 2 | 3 | 0   | 0 | 0 | 79 | 41 | 0 | 0 | 0 | 1 | 5 | 293 | 600 | 900 1200 2400 |





```

1 60146 100.0
265 60647 146.9161 'Nd-147 ' 2 1 85 0 0 0 0 0 0 1 1 1100
1 60147 100.0
266 60648 147.9167 'Nd-148 ' 2 1 86 0 0 115 71 0 0 1 1 1100
1 60148 100.0
267 60650 149.9209 'Nd-150 ' 2 1 87 0 0 116 72 0 0 1 1 1100
1 60150 100.0
268 61648 147.9177 'Pm-148 ' 2 1 89 0 0 0 0 0 0 1 1 1100
1 61148 100.0
269 61649 148.9183 'Pm-149 ' 2 1 90 0 0 0 0 0 0 1 1 1100
1 61149 100.0
270 61651 150.9215 'Pm-151 ' 2 1 91 0 0 0 0 0 0 1 1 1100
1 61151 100.0
271 62648 147.9147 'Sm-148 ' 2 1 94 0 0 0 0 0 0 1 1 1100
1 62148 100.0
272 62653 152.9217 'Sm-153 ' 2 1 99 0 0 0 0 0 0 1 1 1100
1 62153 100.0
273 62654 153.9223 'Sm-154 ' 2 1 100 0 0 0 0 0 0 1 1 1100
1 62154 100.0
274 63651 150.9195 'Eu-151 ' 2 1 101 0 0 117 73 0 0 1 1 1100
1 63151 100.0
275 63656 155.9245 'Eu-156 ' 2 1 105 0 0 0 0 0 0 1 1 1100
1 63156 100.0
276 63657 156.9251 'Eu-157 ' 2 1 106 0 0 0 0 0 0 1 1 1100
1 63157 100.0
277 64654 153.9213 'Gd-154 ' 2 1 107 0 0 118 0 0 0 1 1 1100
1 64154 100.0
278 64660 159.9268 'Gd-160 ' 2 1 112 0 0 119 0 0 0 1 1 1100
1 64160 100.0
279 65159 158.9252 'Tb-159 ' 3 0 0 0 0 0 0 0 0 1 1 1100
1 65159 100.0
280 65160 158.9273 'Tb-160 ' 3 1 0 0 0 0 0 0 0 1 1 1100
1 65160 100.0
281 65161 160.9276 'Tb-161 ' 3 1 0 0 0 0 0 0 0 0 1 1100
282 65659 158.9252 'Tb-159 ' 2 1 113 0 0 0 0 0 0 1 1 1100
1 65159 100.0
283 65660 158.9273 'Tb-160 ' 2 1 114 0 0 0 0 0 0 1 1 1100
1 65160 100.0
284 65661 160.9276 'Tb-161 ' 2 1 115 0 0 0 0 0 0 0 1 1100
285 66660 159.9248 'Dy-160 ' 2 1 116 0 0 0 0 0 0 1 1 1100
1 66160 100.0
286 66661 160.9274 'Dy-161 ' 2 1 117 0 0 0 0 0 0 1 1 1100
1 66161 100.0
287 66662 161.9270 'Dy-162 ' 2 1 118 0 0 0 0 0 0 1 1 1100
1 66162 100.0
288 66663 162.9286 'Dy-163 ' 2 1 119 0 0 0 0 0 0 1 1 1100
1 66163 100.0
289 66664 163.9282 'Dy-164 ' 2 1 120 0 0 120 74 0 0 1 1 1100
1 66164 100.0
290 67165 164.9298 'Ho-165 ' 3 0 0 0 0 121 75 0 0 1 1 1100
1 67165 100.0
291 67665 164.9298 'Ho-165 ' 2 1 121 0 0 122 76 0 0 1 1 1100
1 67165 100.0
292 77191 190.9781 'Ir-191 ' 3 0 0 0 0 0 0 0 0 1 1 1100
1 77191 100.0
293 77193 192.9803 'Ir-193 ' 3 0 0 0 0 0 0 0 0 1 1 1100
1 77193 100.0
294 91232 232.0386 'Pa-232 ' 2 1 0 0 0 0 0 0 0 1 1 1100
1 91232 100.0
295 91234 234.0433 'Pa-234 ' 2 1 0 0 0 0 0 0 0 0 1 1100
%FINE

```

## Appendix C.3 Structure of the V4.0 MPACT MG library

[Structure]**%VERSION:**

[iversion, adte]

**%DIM:**

[nog, nofg, norg, nchi, nelt, nelr, nres, nyld, nsubg, nflxhel]

**%GRP:**[enbhel(ig)]<sub>ig=1,nog</sub>**%CHI:**[chix(ig)]<sub>ig=1,nog</sub>**%DIR:**

do i=1,nelt

[ix, nid, amass, ityp, ifpy, ichi, inmn, ntemp, nrtemp, np1temp, npot, aid]

enddo

**%NUC:** [i, nid, amass, ityp, ifpy, ichi, inmn, ntemp, nrtemp, np1temp, npot, aid]

["TP1+"]

[temp(it)]<sub>it=1, ntemp</sub>

["XSD+"]

do ig=1,nog

do it=1,ntemp

[ig, it, siga, {sigf, signf, sigtr, sigs, ib,ie, sm(jg)}<sub>jg= ib, ie</sub>]

enddo

enddo

[" SP1+"]

do ig=1,nog

do it=1,np1temp

[ig, it, sigsp1(ig,it), ib, ie, smp1(jg,it)]<sub>jg= ib, ie</sub>

enddo

enddo

["POT+"]

[[lamsigp(ig)]<sub>ig=igresb,igrese</sub>[sigres(ig)]<sub>ig=igresb,igrese</sub>[nu(ig)]<sub>ig=igresb,igrese</sub>

[" TP2+"]

[rtemp(it)]<sub>it=1,nrtemp</sub>

[" XS0+"]

[sigpot(ip)]<sub>ip=1,npot</sub>

[" RIA+"]

do ig=igresb,igrese

do it=1,liso(i)%nrtemp

[ig, it, ria(ip,it,ig)]<sub>ip=1, npot</sub>

enddo

enddo

["RIS+"]

do ig=igresb,igrese

do it=1,liso(i)%nrtemp

[ig, it, ris(ip,it,ig)]<sub>ip=1, npot</sub>

enddo

```

    enddo
["RNF+"]
  do ig=igresb,igrese
    do it=1,ldiso(i)%nrtemp
      [ig, it, rinf(ip,it,ig)ip=1, npot]
    enddo
  enddo
["SA7+"]
  do ig=igresb,igrese
    [ig, sgsa(ix,ig)ix=1, nsubg]
  enddo
  do ig=igresb,igrese
    do it=1,ldiso(i)%nrtemp
      [ig, it, sgwa(ix,it,ig)ix=1, nsubg]
    enddo
  enddo
["SF7+"]
  do ig=igresb,igrese
    [ig, sgsnf(ix,ig)ix=1, nsubg]
  enddo
  do ig=igresb,igrese
    do it=1,ldiso(i)%nrtemp
      [ig, it, sgwnf(ix,it,ig)ix=1, nsubg]
    enddo
  enddo
["SA4+"]
  do ig=igresb,igrese
    [ig, sgsafsp(ix,ig)ix=1, nflx]
  enddo
  do ig=igresb,igrese+1
    do it=1,ldiso(i)%nrtemp
      [ig, it, sgwafsp(ix,it,ig)ix=1, nflx]
    enddo
  enddo
["CHI+"]
  [chi(ig)]ig=1, nog
["CHI-"]
  [chid(ig)]ig=1, nog
[" FIS+"]
  [kappa, kappa0]
  [beta(ib)]ib=0,6
  [dcybeta(ib)]ib=1,6
  [yield(iy)]iy=1, nyld
[" DCY+"]
  [ldiso(i)%dcy]
[" TP3+"]
  [ldiso(i)%p1temp(it)]it=1, np1temp
[" SP2+"]
  do ig=1, nog
    do it=1, np1temp
      [ig, it, sigsp1(ig,it), ib, ie, smp2(jg,it)]jg=ib, ie
    enddo
  enddo
[" SP3+"]
  do ig=1, nog
    do it=1, np1temp

```

```

[ig, it, sigsp1(ig,it), ib, ie, smp3(jg,it)]jg=ib, ie
enddo
enddo
[" N2N+"]
[ldiso(i)%sign2n(ig)]ig=1,nog
[" N3N+"]
[ldiso(i)%sign3n(ig)]ig=1,nog
**Repeat %NUC: for melt nuclides.

```

**%END:**

| Variables                      | Type | Description                                                             |
|--------------------------------|------|-------------------------------------------------------------------------|
| nog                            | I    | The number of energy groups                                             |
| nofg                           | I    | The number of fast energy groups                                        |
| norg                           | I    | The number of resonance energy groups                                   |
| nchi                           | I    | The number of energy groups w/ fission spectra                          |
| nelt                           | I    | The total number of nuclides                                            |
| nelr                           | I    | The number of nuclides w/ full xs sets                                  |
| nyld                           | I    | The number of fission product nuclides                                  |
| nres                           | I    | The number of resonant nuclides                                         |
| nsubg                          | I    | The number of subgroup levels                                           |
| nflx                           | I    | The number of coarse subgroup levels                                    |
| igresb                         | I    | Starting group for resonance (=nofg+1)                                  |
| igrese                         | I    | Ending group for resonance (=nofg+norg)                                 |
| u(ig)                          | R    | Lethargy energy group boundaries (ig=1,nog)                             |
| enb(ig)                        | R    | Energy group boundaries (ig=1,nog)                                      |
| chix(ig)                       | R    | Common fission spectrum (ig=1,nog)                                      |
| sm(p?)(i,j)%                   | T    | [scatmat] P0~p3 scattering matrix                                       |
| %ib                            | I    | Stating energy group (in)                                               |
| %ie                            | I    | Ending energy group (in)                                                |
| %ioutsb                        | I    | Stating energy group (out)                                              |
| %ioutse                        | I    | Ending energy group (out)                                               |
| %from(k)                       | R    | Scattering matrix (k=%ib,%ie)                                           |
| ldiso(i)%                      | T    | [libdata] Nuclidewise multigroup data (i=1,nelt)                        |
| %nid                           | I    | Nuclide id.                                                             |
| %ityp                          | I    | Indicator for data type                                                 |
|                                |      | 0 non depletable                                                        |
|                                |      | >0 depletable (w/ decay constant)                                       |
|                                |      | 2 ri-a (fissionable w/ kappa, beta, fpy)                                |
| 3 ri-a & ri-nf (same as above) |      |                                                                         |
| %ichi                          | I    | Indicator for fission spectrum (0/>0 : no/yes)                          |
| %ifpy                          | I    | Indicator & ordering no. for f.p.y.                                     |
| %inmn                          | I    | Indicator for (n,2n) & (n,3n) xs (0/1/2/3 : no/(n,2n)/(n,3n)/both)      |
| %ntemp                         | I    | The number of temperatures for xs                                       |
| %nrtemp                        | I    | The number of temperatures for ri                                       |
| %npot                          | I    | The number of sig-0's for ri                                            |
| %np1temp                       | I    | The number of temperatures for p1~p3                                    |
| %aid                           | A8   | Alphanumeric nuclide id.                                                |
| %aw                            | R    | Atomic mass                                                             |
| %kappa                         | R    | Energy release per fission (w/sec) only w/o neutrino energy             |
| %kappa0                        | R    | Energy release per fission (w/sec) w/o decay and neutron kinetic energy |
| %crit_nd                       | R    | Critical pnd                                                            |
| %dcy                           | R    | Decay constant (/sec)                                                   |



|                 |   |                                                                              |
|-----------------|---|------------------------------------------------------------------------------|
| %chi(j)         | R | Fission spectrum (j=1,nog)                                                   |
| %chid(j)        | R | Delayed neutron spectrum (j=1,nog)                                           |
| %beta(j)        | R | Delayed neutron yields (j=1,6)                                               |
| %dcybeta(j)     | R | Delayed neutron decay constant (j=1,6)                                       |
| %rtemp(j)       | R | Square root temperatures for ri (j=1,%nrtemp)                                |
| %sigpot(j)      | R | Square root sig-0's for ri (j=1,%npot)                                       |
| %temp(j)        | R | Temperatures for xs (j=1,%ntemp)                                             |
| %p1temp(j)      | R | Temperatures for p1~p3 (j=1,%np1temp)                                        |
| %lamsigp(j)     | R | Lambda*potential xs (j=igresb,igrese)                                        |
| %sigsres(j)     | R | Scattering xs (j=igresb,igrese)                                              |
| %nu(j)          | R | The number of neutrons released (j=igresb,igrese)                            |
| %yield(j,k)     | R | Fission product yields (j=1, ;k=1, )                                         |
| %siga(j,k)      | R | Absorption xs (j=1,nog; k=1,%ntemp)                                          |
| %sigf(j,k)      | R | Fission xs (j=1,nog; k=1,%ntemp)                                             |
| %signf(j,k)     | R | Nu*fission xs (j=1,nog; k=1,%ntemp)                                          |
| %sigtr(j,k)     | R | Transport xs (j=1,nog; k=1,%ntemp)                                           |
| %sigs(j,k)      | R | Scattering xs (j=1,nog; k=1,%ntemp)                                          |
| %sigstr(j,k)    | R | Transport correct scattering xs (j=1,nog; k=1,%ntemp)                        |
| %sigsp1(j,k)    | R | P1 scattering xs (j=1,nog; k=1,%np1temp)                                     |
| %sigsp2(j,k)    | R | P2 scattering xs (j=1,nog; k=1,%np1temp)                                     |
| %sigsp3(j,k)    | R | P3 scattering xs (j=1,nog; k=1,%np1temp)                                     |
| %sign2n(j)      | R | (n,2n) xs (j=1,nog)                                                          |
| %sign3n(j)      | R | (n,3n) xs (j=1,nog)                                                          |
| %sm(j,k)%       | T | [scatmat] P0 corrected scattering matrix (j=1,nog; k=1,%np1temp)             |
| %smp1(j,k)%     | T | [scatmat] P1 scattering matrix (j=1,nog; k=1,%np1temp)                       |
| %smp2(j,k)%     | T | [scatmat] P2 scattering matrix (j=1,nog; k=1,%np1temp)                       |
| %smp3(j,k)%     | T | [scatmat] P3 scattering matrix (j=1,nog; k=1,%np1temp)                       |
| %ria(j,k,l)     | R | Absorption ri (j=1,%npot; k=1,%nrtemp; l=igresb,igrese)                      |
| %rinf(j,k,l)    | R | Nu*fission ri (j=1,%npot; k=1,%nrtemp; l=igresb,igrese)                      |
| %ris(j,k,l)     | R | Scattering ri (j=1,%npot; k=1,%nrtemp; l=igresb,igrese)                      |
| %sgsa(j,k)      | R | Subgroup levels of absorption ri (j=1,nsbg; k=igresb,igrese)                 |
| %sgsnf(j,k)     | R | Subgroup levels of nu*fission ri (j=1,nsbg; k=igresb,igrese)                 |
| %sgsafsp(j,k)   | R | Subgroup levels of absorption ri (j=1,nflx; k=igresb,igrese+1)               |
| %sgwa(j,k,l)    | R | Subgroup weights of absorption ri (j=1,nsbg; k=1,%nrtemp; k=igresb,igrese)   |
| %sgwnf(j,k,l)   | R | Subgroup weights of nu*fission ri (j=1,nsbg; k=1,%nrtemp; k=igresb,igrese)   |
| %sgwafsp(j,k,l) | R | Subgroup weights of absorption ri (j=1,nflx; k=1,%nrtemp; k=igresb,igrese+1) |