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**AUSTRALIAN ATOMIC ENERGY COMMISSION**  
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**EDITAR: A MODULE FOR REACTION RATE EDITING**  
**AND CROSS-SECTION AVERAGING WITHIN THE**  
**AUS NEUTRONICS CODE SYSTEM**

by

**G.S. ROBINSON**

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ABSTRACT

The EDITAR module of the AUS neutronics code system edits one- and two-dimensional flux data pools produced by other AUS modules to form reaction rates for materials and their constituent nuclides, and to average cross sections over space and energy. The module includes a  $B_L$  flux calculation for application to cell leakage. The STATUS data pool of the AUS system is used to enable the 'unsmearing' of fluxes and nuclide editing with minimal user input. The module distinguishes between neutron and photon groups, and printed reaction rates are formed accordingly. Bilinear weighting may be used to obtain material reactivity worths and to average cross sections. Bilinear weighting is at present restricted to diffusion theory leakage estimates made using mesh-average fluxes.

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E CODE; FISSION; NEUTRON FLUX; NEUTRON LEAKAGE; NEUTRONS; REACTION KINETICS; REACTIVITY WORTHS; REACTOR KINETICS; TOTAL CROSS SECTIONS; WEIGHTING FUNCTIONS

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## 1. INTRODUCTION

The AUS neutronics system [Robinson 1975a] is a modular system of computer codes for application to a wide range of neutronics calculations including thermal reactors, fast reactors and fusion blankets. The EDITAR module of AUS is a general purpose module for the editing of fluxes, which may be used either for lattice cell calculations or for global calculations. EDITAR replaces the earlier EDIT module [Pollard 1975] which performed only some of the functions of the present module.

EDITAR may be used

- to print reaction rates for both the materials in the system and their constituent nuclides;
- to average cross sections over space and energy, and either print the results or produce an AUS cross-section data pool;
- to apply bilinear weighting when forming reaction rates and thus obtain material worths;
- to apply bilinear weighting in cross-section averaging;
- to distinguish between neutron and photon groups to print separate reaction rates for neutrons and photons;
- to calculate the leakage from a cell using a  $B_L$  calculation; and
- to 'unsmeared' the fluxes in one EDITAR calculation by using the smearing factors from a previous EDITAR calculation.

The module may be used, for instance, to obtain detailed reaction rates of nuclides within a cell from a global flux calculation which uses cell-average cross sections.

EDITAR may be applied to either one- or two-dimensional calculations. As most editing is done by material, the number of dimensions has little effect. An option is available to print reaction rates at each mesh interval. Under this option, all reactions are printed at each mesh interval for 1D cases but only one reaction, *e.g.* fission, is printed for 2D cases.

## 2. USE OF AUS DATA POOLS

The EDITAR module normally obtains cross-section input for materials and nuclides from the AUS data pool on FORTRAN unit 10. These cross sections may be supplemented from a second AUS cross-section data pool on FORTRAN unit 11. Where a second set of cross sections is used, it is intended that the first set of data should be for materials, whereas the second set should be for the nuclides present in those materials. The latter set may be in a different group structure from the first, if the first group structure is a condensation of the second. The purpose of this option is to enable a sequence of calculations in which a nuclide cross-section data pool remains uncondensed while the material cross sections are condensed and spatially smeared, possibly through more than one stage.

The geometry is obtained from the geometry data pool on FORTRAN unit 7. The input flux is obtained from the flux data pool on FORTRAN unit 9, which may be either of the FLUXA or FLUXB type. If a FLUXA type, which specifies mesh-edge fluxes, is given these are first converted to mesh-average fluxes. Although this conversion has no effect on reaction rates, it affects quantities related to leakage between material zones.

The data on material composition and possible material smearing are obtained from the pair of STATUS data pools ST1 and ST2. The ST1 data are taken from FORTRAN unit 4 and the ST2 data from unit 5. EDITAR may be used without these data, in which case only the materials directly in the geometry specification are considered to be in the system. The significance of entries on the STATUS data pool has been discussed by Robinson [1975b].

For cell calculations, the latest cell name is taken from ST2 and only the material definitions for that cell name are read from ST1. When a  $B_L$  calculation is performed, a GFAC entry giving the ratio of  $k_{eff}$  to  $k_{inf}$  flux in each group is added to both ST1 and ST2. When AUS cross sections are written for any material, a GRPS entry followed by a mixing rule entry for each output material are added to ST1. The mixing rule entry is made even if the material is simply transferred from the input to the output cross-section data pool. No entries are made for any output nuclides and their names remain unchanged. The 20-byte name of output materials is constructed to maintain uniqueness of material names.

### 3. CALCULATION OF LEAKAGE FROM A CELL

EDITAR includes a  $B_L$  flux calculation which may be applied to the calculation of leakage from a lattice cell. To obtain a homogeneous system to which the  $B_L$  equations can be applied, cell-average cross sections are formed by flux weighting with the input flux. Simple flux weighting is used even if input spatial currents are available. A transport corrected  $P_0$  calculation is performed rather than a  $B_0$  calculation. For  $P_0$  calculations, the input transport cross sections are flux weighted before forming the cell-average diffusion coefficient. EDITAR does not include a streaming treatment.

The solution of the  $B_L$  equations is exactly the same as that in the MIRANDA module [Robinson 1977] from which the flux solution subroutine used in EDITAR was taken. When bilinear weighting is requested, the adjoint  $B_L$  equations are also solved.

A flux within the cell which includes leakage effects is obtained by multiplying the homogeneous flux from the  $B_L$  calculation by the ratio of input spatial flux to the integrated input flux. The same ratio is used to multiply any homogeneous currents calculated from the  $B_L$  equations. Even if spatial currents are available, they are ignored in cell calculations unless the user directs otherwise (see section 5.2). In that case, any leakage currents are ignored. Input adjoint distributions are always ignored when a cell-leakage calculation is performed.

### 4. FORMULAE USED IN AVERAGING

#### 4.1 Flux Unsmearing

Although the unsmearing of fluxes is not explicitly represented in the formulae given, it is allowed in all applications. However, for bilinear weighting and for the production of an AUS cross-section data pool, there is a restriction that the cross sections and the input spatial flux must be in the same number of groups. Unsmearing of the adjoint flux is not supported, therefore a uniform adjoint is assumed in unsmearing. In the averaging formulae given below, any reaction rate at an interval  $i$  in group  $G$ ,  $N_i \sigma_{iG} \phi_{iG} v_i$ , may be simply that or, when unsmearing is performed, may represent the fuller expression for reaction rate,  $r$ ,

$$r = \sum_{m \in M} \sum_{g \in G} N_m f_m d_{mg} \sigma_{mg} \phi_{iG} v_i \quad (1)$$

where  $m$  represents one of the discrete materials which has been smeared into the material  $M$  present at the mesh interval  $i$ ;  $N_m$  is the atom density of a nuclide in material  $m$ ;  $f_m$  is the volume fraction of material  $M$  occupied by material  $m$ ;  $d_{mg}$  is the smearing factor in a group  $g$  condensed to form group  $G$ ; and  $\sigma_{mg}$  is the nuclide cross section to be used in material  $m$ .

The unsmearing factor  $d_{mg}$  is obtained from a previous EDITAR run via the STATUS data pool, and is given by

$$d_{mg} = \sum_{j \in m} V_j \psi_{jg} / f_m \sum_{j \in M} \sum_{g \in G} V_j \psi_{jg} \quad (2)$$

where  $V_j$ ,  $\psi_{jg}$  are volumes and fluxes used in the previous EDITAR run; and  $j \in m$  represents the mesh intervals at which  $m$  is present. When a microscopic reaction rate is being formed,  $N_m$  is set to 1 where  $N_m \neq 0$ .

The unsmearing routine used in EDITAR is more general than is indicated above. Any number of smear/condense steps may have been performed when forming the cross sections used in the current spatial flux calculation from the set of cross sections  $\sigma_{mg}$ .

#### 4.2 Flux Weighting Option

The simple flux weighting formula

$$\sigma_G = \sum_{g \in G} \sum_i N_i \sigma_{ig} \phi_{ig} v_i / \sum_{g \in G} \sum_i N_i \phi_{ig} v_i \quad (3)$$

is used for all reactions when forming cross sections that are to be printed, and the numerator is used for printed reaction rates. In this formula, and in those below,  $N_i$  takes the following values:

- for nuclides present in the system,  $N_i$  is the atom density in interval  $i$  if macroscopic weighting is used, or unity at mesh intervals where  $N_i \neq 0$  if microscopic weighting is used;
- for nuclides or materials not present in the system,  $N_i = 1$  for all  $i$ ;
- for materials present in the system,  $N_i = 1$  for intervals which include the material.

The term 'all reactions' refers to an AUS cross-section record, which normally includes, in the following sequence, cross sections for transport, absorption, fission emission, scattering, total, fission,  $(n, \gamma)$ ,  $(n, 2n)$ , and kerna factor.

The simple formula is also used when forming cross sections, other than the transport cross section, for an AUS data pool. In global calculations, the formula used for the transport cross section is

$$\sigma_G = \frac{\sum_{g \in G} \sum_i N_i \sigma_{ig} D_{ig} \phi_{ig} v_i}{\sum_{g \in G} \sum_i N_i D_{ig} \phi_{ig} v_i} \quad (4)$$

where  $D_{ig}$  is the diffusion coefficient of the material actually at interval  $i$ .

In cell calculations, the formulae for the transport cross section are the same as those used in MIRANDA. That is, for  $P_0$  calculations,

$$\sigma_G = \frac{\sum_{g \in G} D_g B_g^2 \sum_i N_i \sigma_{ig} \phi_{ig} v_i \sum_{g \in G} \psi_g}{\sum_{g \in G} \sum_i N_i \phi_{ig} v_i \sum_{g \in G} D_g B_g^2 \psi_g} \quad (5)$$

where  $D_g$  is the cell diffusion coefficient;  $B_g^2$  is the buckling; and  $\psi_g$  is the homogeneous cell flux. For  $B_L$  calculations, the  $D_g B_g^2 \psi_g$  term in the denominator is replaced by the  $B_L$  leakage estimate. Also, in  $B_L$  calculations, transport cross sections and hence the cell-diffusion coefficient are not taken directly from the cross-section input but formed from

$$\sigma_{trg} = \sigma_{tg} - \sum_h \sigma_{h \rightarrow g, 1} \psi_{h1} / \psi_{g1} \quad (6)$$

where  $\sigma_{trg}$ ,  $\sigma_{tg}$  are the transport and total cross section;  $\sigma_{h \rightarrow g, 1}$  is the  $P_1$  scattering matrix; and  $\psi_{g1}$  is the homogeneous cell current. It may be noted that the formula for averaging the transport cross section after a  $B_L$  calculation includes the ratio of the  $DB^2$  leakage estimate to the  $B_L$  leakage estimate. This is appropriate if the transport cross section is to be used in a diffusion theory calculation. Otherwise, the input option TRANSPORT P1 should be used to give

$$\sigma_G = \frac{\sum_{g \in G} (L_g / \psi_g) \sum_i N_i \sigma_{ig} \phi_{ig} v_i \sum_{g \in G} \psi_g}{\sum_{g \in G} \sum_i N_i \phi_{ig} v_i \sum_{g \in G} L_g} \quad (7)$$

where  $L_g$  is the  $B_L$  leakage estimate.

For averaging the scattering matrices, the simple formula is

$$\sigma_{G \rightarrow H} = \frac{\sum_{g \in G} \sum_i N_i \sum_{h \in H} \sigma_{i, g \rightarrow h} \phi_{ig} v_i}{\sum_{g \in G} \sum_i N_i \phi_{ig} v_i} \quad (8)$$

The self-scatter term of the  $P_0$  matrix is adjusted to be

$$\sigma_{G \rightarrow G} = \sigma_{tG} - \sigma_{aG} - \sum_{H \neq G} \sigma_{G \rightarrow H} \quad (9)$$

where  $\sigma_{aG}$  is the absorption cross section,  $\sigma_{tG}$  is a total cross section obtained by averaging  $\sigma_t$  using equation (4) for global calculations, or equation (7) for cell calculations, and  $\sigma_{tg}$  is  $\sigma_{ag} + \sum_h \sigma_{g \rightarrow h}$ .

To average the  $P_n$  expansion of the scattering matrix,  $\sigma_{g \rightarrow h, n}$  for  $n > 0$ , equation (8) is used, with  $\phi_{ig}$  being replaced by the appropriate term of the flux expansion. If that term is unavailable, the highest order available  $\phi$  term is used. The fission spectra are weighted with the fission emission rate.

### 4.3 Bilinear Weighting Option

The formulae used for bilinear weighting are based on the use of diffusion theory estimates of those quantities connected with particle transport. A minor adjustment was made to allow for a  $B_L$  calculation of cell leakage. The formulae used in averaging the transport cross sections are extensions of the homogeneous  $P_0$  formula

$$\sigma_{trg} = \frac{\sum_{g \in G} \sigma_{trg} D_g^2 B_g^2 \phi_g \phi_g^*}{\sum_{g \in G} B_g^2 \phi_g \phi_g^* / (\sum_{g \in G} D_g B_g^2 \phi_g \phi_g^*)^2} \quad (10)$$

which ensures that for homogeneous calculations the leakage contributions to material worths are preserved, and that

$$D_G = \frac{\sum_{g \in G} D_g B_g^2 \phi_g \phi_g^*}{\sum_{g \in G} B_g^2 \phi_g \phi_g^*} \quad (11)$$

All the formulae are consistent with defining the condensed adjoint as

$$\phi_G^* = \frac{\sum_{g \in G} \phi_g \phi_g^*}{\sum_{g \in G} \phi_g} \quad (12)$$

### 4.3.1 Reaction rates

A request for reaction rates results in a printout of the components of material reactivity worths in the condensed group structure. The perturbation denominator  $P$  is given by

$$P = (1/k_{eff}) \sum_i \sum_s (\nu\sigma_f)_{is}^M \phi_{is} \sum_h \chi_{ih}^M \phi_{ih}^* v_i , \quad (13)$$

where  $(\nu\sigma_f)_{is}^M$  and  $\chi_{ih}^M$  are, respectively, the fission emission cross section and the fission spectrum of the materials actually in the system. The components for absorption, fission emission and scattering are given by

$$-(1/P) \sum_{g \in G} \sum_i N_i \sigma_{aig} \phi_{ig} \phi_{ig}^* v_i , \quad (14)$$

$$(1/Pk_{eff}) \sum_{g \in G} \sum_i N_i (\nu\sigma_f)_{ig} \phi_{ig} \sum_h \chi_{ih} \phi_{ih}^* v_i , \quad (15)$$

$$(1/P) \sum_{g \in G} \sum_i N_i \sum_h \sigma_{i,g \rightarrow h} \phi_{ig} (\phi_{ih}^* - \phi_{ig}^*) v_i . \quad (16)$$

In global calculations, the leakage components are given by

$$(3/P) \sum_{g \in G} \sum_i N_i \sigma_{tr ig} D_{ig} A_{ig} , \quad (17)$$

where

$$A_{ig} = \int_{V_i} \nabla \phi_g^* \cdot D_g \nabla \phi_g dv \quad (18)$$

In cell calculations, the leakage components become

$$(3/P) \sum_{g \in G} \sum_i N_i \sigma_{tr ig} \phi_{ig} v_i D_g L_g \psi_g^* / \psi_g . \quad (19)$$

where  $D_g, \psi_g, \psi_g^*, L_g$  are for the homogeneous leakage calculation, and the leakage,  $L_g$ , reduces to  $D_g B_g^2 \psi_g$  in a  $P_0$  calculation. It should be noted that

$$\delta D = \delta \sigma_{tr} \frac{dD}{d \sigma_{tr}} \quad (20)$$

has been used to obtain these formulae.

The simple flux weighted reaction rate is printed out for those cross sections, such as fission and capture, not used directly in a transport calculation.

### 4.3.2 Cross sections

The formulae given below for cross sections are used both for AUS cross section output and for printed output. Because a scattering matrix which generates or destroys particles is not permitted in the AUS system, the absorption cross section is adjusted to include the loss or gain due to scattering which occurs when bilinear weighting is used. Therefore, in calculations which use bilinear weighted cross sections, absorption and scattering can not be considered as separate processes.

The absorption cross section is given by

$$\begin{aligned} \sigma_{aG} &= \sum_{g \in G} \sum_i N_i \sigma_{aig} \phi_{ig} \phi_{ig}^* v_i / \sum_{g \in G} \sum_i N_i \phi_i \phi_{ig}^* v_i \\ &+ \sigma_{sG} - \sum_H \sigma_{G \rightarrow H} , \end{aligned} \quad (21)$$

where the total scattering cross section  $\sigma_{sG}$  is given by

$$\sigma_{sG} = \sum_{g \in G} \sum_i N_i \sum_h \sigma_{i,g \rightarrow h} \phi_{ig} \phi_{ig}^* v_i / \sum_{g \in G} \sum_i N_i \phi_{ig} \phi_{ig}^* v_i , \quad (22)$$

the condensed scattering matrix  $\sigma_{G \rightarrow H}$  is given by

$$\sigma_{G \rightarrow H} = \sum_{g \in G} \sum_i N_i \sum_{h \in H} \sigma_{i,g \rightarrow h} \phi_{ig} \phi_{ih}^* v_i / \sum_{g \in G} \sum_i N_i \phi_{ig} \phi_{ih}^* v_i , \quad (23)$$

and the condensed adjoint  $\phi_{iH}^*$  is

$$\phi_{iH}^* = \sum_{h \in H} \phi_{ih} \phi_{ih}^* / \sum_{h \in H} \phi_{ih} . \quad (24)$$



The fission emission cross section is given by

$$(\nu\sigma_f)_G = \frac{\sum_{g \in G} \sum_i N_i (\nu\sigma_f)_{ig} \phi_{ig} \sum_h \chi_{ih} \phi_{ih}^* \nu_i}{\sum_{g \in G} \sum_i N_i \phi_{ig} \sum_n \chi_{in} \phi_{in}^* \nu_i} \quad (25)$$

and the fission spectrum by

$$\chi_G = \frac{\sum_{g \in G} \sum_i N_i \sum_h (\nu\sigma_f)_{ih} \phi_{ih} \chi_{ig} \phi_{ig}^* \nu_i}{\sum_H (\nu\sigma_f)_H \sum_{h \in H} N_i \phi_{ih} \phi_{ig}^* \nu_i} \quad (26)$$

where  $(\nu\sigma_f)_H$  is obtained from equation (25) and  $\phi_{ig}^*$  from equation (24). To conform to the normal usage for the fission spectrum normalisation, the quantities used are

$$\chi_G^1 = \chi_G / \sum_H \chi_H \quad (27)$$

and

$$(\nu\sigma_f)_G^1 = (\nu\sigma_f)_G \sum_H \chi_H \quad (28)$$

In global calculations, the transport cross section is given by

$$\sigma_{tr G} = \frac{\sum_{g \in G} \sum_i N_i \sigma_{tr ig} D_{ig} A_{ig} \sum_{g \in G} \sum_i \delta_i A_{ig} / D_{ig}}{\sum_{g \in G} \sum_i N_i A_{ig} \sum_{g \in G} \sum_i \delta_i A_{ig}} \quad (29)$$

where  $\delta_i = 1$  where  $N_i \neq 0$ , else  $\delta_i = 0$ .

The total cross section obtained from the  $P_0$  scattering matrix ( $\sigma_{ig} = \sigma_{ag} + \sum_h \sigma_{g \rightarrow h}$ ) is weighted in the same way. In cell calculations, the transport cross section is given by

$$\sigma_{tr G} = \frac{\sum_{g \in G} \sum_i N_i \sigma_{tr ig} \phi_{ig} \nu_i D_g L_g \psi_g^* / \psi_g \sum_{g \in G} \psi_g \sum_{g \in G} \psi_g \psi_g^* B_g^2}{\sum_{g \in G} \sum_i N_i \phi_{ig} \nu_i (\sum_{g \in G} L_g \psi_g^*)^2} \quad (30)$$

and the total cross section by a similar formula but with  $D_g$  replaced by  $L_g / \psi_g B_g^2$ .

The self-scatter term of the  $P_0$  scattering matrix is adjusted as usual:

$$\sigma_{G \rightarrow G} = \sigma_{tG} - \sigma_{aG} - \sum_{H \neq G} \sigma_{G \rightarrow H} \quad (31)$$

For averaging the  $P_n$  expansion of the scattering matrix for  $n > 0$ , equation (23) is used with  $\phi_{ig}$  replaced by the appropriate term of the flux expansion.

Simple flux weighting (equation (3)) is used for those cross sections not used directly in a transport calculation.

## 5. INPUT DESCRIPTION

### 5.1 General Layout

Input data to EDITAR are in free format and read with the SCAN input routine [Bennett and Pollard 1967]. The data are given in the form of a number of entries, each of which consists of a keyword followed by a string of numeric or alphabetic data items. More than one entry may be given on each input record from which only the first 72 characters are processed.

Most of the input parameters have been given default values. Thus, if the standard value is required, an entire entry may be omitted or trailing data may be left off the end of a data string. The data entries are grouped into three sets. The first set gives the major parameters which establish the type of calculation to be performed. This set must be given first because they are processed before the AUS data pools are read. The second set provides parameters for the leakage flux calculation if this is required. The third set specifies the output requirements. In the following description, information which is to be reproduced exactly is in upper case.

### 5.2 Major Parameters

#### 5.2.1 USE cross-section data specification

USE iuxs1,iuxs2  
where

iuxs1 is the FORTRAN unit number of the main input cross-section data pool (default 10), and  
iuxs2 is the FORTRAN unit number of an additional input cross-section data pool giving the isotope data (default 0).

The number of groups on iuxs1 and iuxs2 may be different, with the iuxs1 group structure being collapsed from the iuxs2 structure, except when bilinear weighting is used.

#### 5.2.2 CALC calculation type

CALC CELL or MULTICELL or GLOBAL or REACTOR

where only one of the four types should be given:

CELL indicates a calculation in which only one set of data is required from the STATUS data pool and a cell-diffusion coefficient is used in weighting transport cross sections.

MULTICELL is similar to CELL but all data are required from the STATUS data pool.

GLOBAL or REACTOR indicates a complete calculation as opposed to a CELL calculation. Current weighting of transport related cross sections is used if currents are available on the flux data pool.

The default calculation type, which is GLOBAL if the geometry includes a free boundary and CELL if there is no free boundary, should be satisfactory for most calculations.

#### 5.2.3 WEIGHT cross-section weighting

WEIGHT FLUX or BILINEAR or CURRENT

where only one of the three weights should be given:

FLUX causes flux weighting to be used.

BILINEAR causes bilinear, *i.e.* the product of flux and adjoint, weighting to be used.

CURRENT causes current weighting to be used if input currents are available.

This specification overrides the FLUX or CURRENT weighting which is implicit for each calculation type.

#### 5.2.4 PRINT control of printed output

PRINT i

where  $i=-1$  for a minimal print,

$i=0$  for a normal print (the default),

$i=1$  for additional print including nuclide names, mixing and smearing table, and

$i=2$  for debug prints.

Although it is not a major parameter, PRINT may be given with this set of entries to print the information from the input data pools.

### 5.3 Flux Calculation Parameters

This set of parameters controls the leakage flux calculation which is applied by default to calculations with no free boundary condition. Most of the entries are the same as the MIRANDA module.

#### 5.3.1 KINF no leakage calculation

KINF

where this keyword is given if no leakage calculation is required.

#### 5.3.2 BUCK group buckling

BUCK Bn bgsq(1) bgsq(2) ...

where Bn specifies the order of the  $B_L$  flux solution and is optional; the default is the order of the input cross section.

bgsq(i) gives the buckling for energy group i.

If a short list of bgsq(i) is given, the last value in the list is used for the remaining groups. Thus one value is sufficient for a constant group buckling. The default buckling is  $0.001 \text{ cm}^{-2}$ . A transport corrected  $P_0$  rather than  $B_0$  calculation is performed.

#### 5.3.3 SEARCH search for critical buckling

SEARCH ON BSQ FOR k dk  
OFF

where ON,OFF turns the search on or off and the default is ON,

k is the value of keff required - default 1.,

dk is the accuracy to which k is required - default 0.0002.

#### 5.3.4 FLUXPARM flux options seldom required

FLUXPARM limfl limsea nscale acc

where limfl is the limit on the iterations to converge the flux calculation - default 100,

limsea is not used,

nscale is the number of broad groups used in scaling the thermal flux to improve convergence - default 3, and

acc is the required accuracy - default 0.0001.

### 5.4 Output Requirements

#### 5.4.1 OUTPUT type of output required

OUTPUT RR XS AUS Pn iout SCAN irm

where a selection of the parameters should be made,

RR specifies printed reaction-rate output,

XS specifies printed cross-section output,

AUS specifies AUS cross-section data and STATUS data output,

Pn is the scattering order for AUS output - default is of the same order as input cross sections,

iout is the FORTRAN unit number for AUS cross-section output - default 12,

SCAN specifies printed one-group microscopic reaction rate output at each mesh interval,

irm is the AUS reaction number for which the SCAN is given, which is meaningful only for 2D geometry and has a default value of 6 (fission). A value for irm must only be entered immediately after SCAN.

The default output type is AUS. RR output with bilinear weighting gives material worths and also fission and capture flux-weighted reaction rates. The output from SCAN is flux-weighted only.

#### 5.4.2 GROUPS condensed group structure

GROUPS n iu il(1) il(2) ... il(n)

where n is the number of condensed groups,

iu is the first input group of condensed group 1, and

il(i) is the last input group of condensed group i.

### 5.4.3 WHERE mesh interval labelling

Mesh interval labels are used to denote a set of mesh intervals over which a material may be averaged. EDITAR generates a label for each material directly specified in the geometry, which labels the intervals containing a material with the name of that material. Thus if a 5-interval calculation had a material named FUEL in intervals 1 and 2, and a material named COOL in intervals 3,4 and 5, then the interval set (1,2) is given the label FUEL and the interval set (3,4,5) is given the label COOL. The label ALL is also automatically given to the complete set of intervals. Additional interval labelling may be established using a number of WHERE entries, which should all be given before the first START keyword.

WHERE name j(1) j(2) ... j(n)

where name is the label to be given to the set of n intervals,

j(i) is the *i*th interval of the set. In 2D geometry the intervals are imagined to form a single vector.

### 5.4.4 REQD output material selection

REQD name(1)@lab(1) name(2)@lab(2) ...

where name(i) is :

- an input material or nuclide name,
- or a WHERE interval label,
- or the name CELL for a material which is the cell average,
- or the name ALLMAT1 which implies all materials and nuclides on the first input cross-section data pool,
- or the name ALLMAT2 which implies all nuclides on the second input cross section data pool,
- @ is the character@to indicate an optional lab(i) is given,
- lab(i) is a WHERE interval label over which the name(i) material or nuclide is to be averaged.

If a REQD entry is not given, the default is ALLMAT1. If @ lab(i) is not given, name(i) is averaged over the intervals in which it is present, or over @ ALL when it is not in any intervals. If name(i) is a WHERE interval label rather than a material, a material is formed by averaging the materials in each of the intervals specified by the WHERE. Thus the name ALL is equivalent to CELL. The material CELL is renamed with the name of the cell.

### 5.4.5 AVERAGE averaging of nuclide data

More than one set of data for a nuclide may be included as input to EDITAR. The default option is to treat each set of data as if it were a distinct nuclide. If average output for some nuclides is desired, this is obtained with the entry

AVERAGE name(1) name(2) ...

where name(i) is the name of a nuclide for which the output data are to be averaged to form a single set of data for that nuclide. If the entry AVERAGE is given without a following nuclide list, all nuclides will be averaged. An AVERAGE entry must not be given before the REQD entry to which it applies.

### 5.4.6 MICRO and MACRO weighting of nuclide data

The default option (MACRO) is to print macroscopic reaction rates for those nuclides which are a constituent of geometry materials. The concentrations are obtained from the STATUS data pool. The output cross sections of a nuclide are microscopic but the concentrations are used in spatial averaging. To obtain microscopic reaction rates and to ignore concentrations in spatial averaging, the required keyword is

MICRO

Microscopic reaction rates are divided by the region volume. The keyword

MACRO

is used to restore the standard state.

#### 5.4.7 SCANWT group weights used with a reaction SCAN

SCANWT gw(1) gw(2) ... gw(n)

where gw(i) is the weight to be given to input group i in a reaction rate SCAN, and  
n is the number of input groups.

The default weights are all unity.

#### 5.4.8 TRANSPORT transport cross-section option

Transport cross sections after a  $B_L$  calculation are normally adjusted to give the correct leakage in a diffusion theory calculation of a reactor. If this is not appropriate, the required entry is

TRANSPORT P1

#### 5.4.9 SMEAR smearing factor options

When it is necessary to smear materials in one calculation for use in a following calculation, and reaction rate editing by nuclide is required in the second calculation, the normal procedure within AUS is not to condense the nuclide data and to unsmear into group fluxes in the original group structure. (Note that unsmearing of the adjoint flux is not supported.) If the nuclide data are being group-condensed, smearing factors in the condensed groups are required on the STATUS data pool. Alternatively, the STATUS data pool may be ignored completely and the volume fraction, smearing factors and concentration must then be included in the nuclide cross section. To use either of these alternative treatments, the entry in the EDITAR calculation which performs the smearing is

SMEAR jsrn ksm

where jsrn=1 if the the smearing factors are to be written in the condensed group structure - default 0,

ksm=1 if everything is to be included in the nuclide cross-section - default 0.

#### 5.4.10 START initiate calculation

The keyword

START

is required to initiate the calculation set up by the preceding input. More than one START keyword may be given. In this case, only the changed input needs to be given for the following START directives.

#### 5.4.11 STOP termination

The keyword

STOP

completes the input to an EDITAR run. It is not essential.

### 5.5 Sample Entries

The sample input given below is a collection of sample entries which is not intended to be a sample EDITAR run.

```
USE 11,10    CALC GLOBAL    WEIGHT BILINEAR
PRINT 1
BUCK B1 3.5E-3    SEARCH OFF
SEARCH ON BSQ FOR 0.98
FLUXPARM 150 0.1 1.E-5
OUTPUT AUS P1 11
OUTPUT XS SCAN 7
GROUPS 4 1 3 7 12 16
WHERE SOME=4(1)8
REQD CELL SOME ALLMAT1 U235 U235    @ SOME
AVERAGE U235 PU239
SCANWT 7*0.9*1.
SMEAR 1 0
```

## 6. REFERENCES

- Bennett, N.W. & Pollard, J.P. [1967] - SCAN - a free input subroutine for the IBM360. AAEC/TM399.
- Pollard, J.P. [1975] - AUS module EDIT. AAEC unpublished report.
- Robinson, G.S. [1975a] - AUS - the Australian modular scheme for reactor neutronics computations. AAEC/E369.
- Robinson, G.S. [1975b] - AUS burnup module CHAR and the associated STATUS data pool. AAEC/E372.
- Robinson, G.S. [1977] - AUS module MIRANDA - a data preparation code based on multiregion resonance theory. AAEC/E410.