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BURNUP MODULES OF THE AUS NEUTRONICS CODE SYSTEM

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ABSTRACT

In the AUS neutronics code system, the burnup module CHAR solves the nuclide depletion equations by an analytic technique in a number of spatial zones. CHAR is usually used as one component of a lattice burnup calculation but contains features which also make it suitable for some global burnup calculations. BURNMAC is a simple accounting module based on the assumption that cross sections for a reactor zone depend only on irradiation. BURNMAC is used as one component of a global calculation in which burnup is achieved by interpolation in the cross sections produced from a previous lattice calculation.

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ANALYTICAL SOLUTION; B CODES; C CODES; BURNUP; CROSS SECTIONS; NEUTRON FLUX ; NEUTRONS; NUCLEAR FUELS; REACTOR LATTICES; IRRADIATION; ISOTOPES

CONTENTS

1. INTRODUCTION

The AUS neutronics code system [Robinson 1975a] was developed at Lucas Heights for application to a wide range of thermal and fast fission reactors. AUS has been extended recently [Robinson 1984] to facilitate fusion blanket neutronics calculations. The modules CHAR and BURNMAC provide a flexible burnup capability for the AUS system.

The CHAR module solves the nuclide depletion equations for a number of reactor zones using an analytic technique. This method depends on ordering the nuclides such that, with few exceptions, burnup is in one direction within that heirarchy. This is not a real restriction on application of the method. CHAR is normally used as one component of a lattice bumup calculation. However, the ability to handle multiple sets of cross-section data for a nuclide and to unsmear fluxes makes CHAR suitable for global burnup calculations. Such global calculations require a severe restriction on one or more of the number of reactor zones, the number of nuclides considered, and the complexity of cross-section generation methods. CHAR may also be used for fission product inventory calculations which make use of a large set of fission products additional to the set for which cross sections are available on the standard library AUS.ENDFB [Robinson 1985]. An early version of the CHAR module, documented by Robinson [1975b], contains details of the method for solving the nuclide depletion equations.

Global burnup calculations within the AUS system can also be done with the BURNMAC module. This classical method is based on the assumption that the macroscopic cross sections are mainly a function of irradiation. Therefore global burnup calculations of reactivity and flux distributions can be performed using sets of macroscopic cross sections as a function of irradiation obtained from previous lattice burnup calculations. BURNMAC is a simple module to interpolate cross sections and perform fuel accounting for use with a flux calculation module to undertake this type of burnup calculation. The current version of BURNMAC does not include a dependence of cross sections on temperature or flux level. Therefore it is not well suited to calculations of large thermal power reactors where such effects may have a considerable effect on flux distributions.

2. DESCRIPTION OF CHAR MODULE

2.1 General

The CHAR module was developed to provide a method for solving nuclide depletion equations which may be applied to any burnup calculation in which individual nuclides are represented. Any number of spatial divisions and any number of nuclides may be represented. The required data on neutron fluxes, geometry, cross sections, burnup processes, and initial nuclide concentrations are obtained from AUS data pools [Robinson 1975a, 1975b].

The module applies an analytic solution of the depletion equations to burn up each discrete material of the system under study; this system may be either a single lattice cell or a whole reactor. The user needs to ensure that there is in the data pools a separate definition in terms of its constituent nuclides for each material which, for burnup, may be considered to have a spatially uniform flux. The module adds the updated nuclide concentrations to the AUS data pool and optionally remixes the macroscopic cross sections.

2.2 Solution Of The Nuclide Depletion Equations

At the heart of the calculation is the solution of the nuclide depletion equations for a given set of microscopic reaction rates using an analytic technique [Robinson 1975a]. The solution requires that the reaction rates remain invariant (constant flux) and the nuclides be numbered so that nuclides burn to form nuclides with a higher number. Nuclides may, however, burn to the immediately preceding nuclide so long as such occurrences are isolated. These restrictions cause no real problems in practice, and the constant flux assumption may be overcome by using a number of time steps and renormalising to a required power at the beginning of each step.

By using the standard nuclide ordering included in the burnup information on the data pools, only a few minor burnup reactions cannot be represented in the analytic solution. These are treated very simply by adding $\hat{N}_n\ a_{mn}\ \Delta t$ to the concentration of nuclide m at the end of a time step Δt, where a_{mn} is the microscopic reaction rate for producing nuclide m from nuclide n, $\hat{N}_n$ is the average concentration of nuclide n during the time step, and $m < n-1$. The analytic solution provides an analytic average concentration as well as the required concentrations at the end of the time step. Reactions requiring this simple treatment are noted in the printed output.

With few exceptions, nuclides are ordered using an integer decimal of the form PZZAAAI, where P is 1 for a fission product, else 0, ZZ is the nuclide charge number, AAA is the nuclide mass number, and I is 1 for the ground state when a metastable isomer is also given, else 0. The inclusion of alpha decay causes some problems. Thus ^{238}Pu is ordered after ^{242}Cm to represent properly the decay of ^{242}Cm to ^{238}Pu, and neutron capture in ^{238}Pu is

represented only approximately.

The analytic method is susceptible to round-off errors. This causes few problems when 64-bit arithmetic is used but the inclusion of large sets of fission products requires some care. The problem arises when the fission products in the set are all interconnected. In calculations of fast reactors using the set of 155 fission products on AUS.ENDFB it was advisable to break the interconnections by changing the burnup order indicator of ^{123}Te and ^{162}Dy to cause captures in these two nuclides to be treated approximately. Fast reactor calculations are more susceptible to this problem because one-group cross sections have less variation between nuclides than they have for thermal reactors. In calculations of fission product inventories, using the extended set of fission products described in Section 2.3, the approximate method of treating captures in ^{123}Te is built into the module.

2.3 Details Related to Data Pool Usage

The system geometry is obtained from an AUS geometry data pool on FORTRAN unit 4. The flux distribution for the geometry is obtained either from an AUS FLUXA data pool on FORTRAN unit 5 or from an AUS FLUXB data pool on FORTRAN unit 6. This information is combined to give the volume and average group fluxes for each material explicitly represented in the geometry.

The STATUS data pools ST1 and ST2 should be given on FORTRAN units 9 and 10, respectively. The current burnup time is obtained from the ST2 data set, and the information on the ST1 data set for that time is read. If a suitable GFAC entry which gives the ratio of a k_{eff} to a k_{inf} flux solution is present on ST2, it is used to modify the input flux.

The ST1 data give the current irradiation and composition of each discrete material and, if spatial smearing has been performed, the smearing factors and group condensation vector. A discrete material is simply one for which a definition in terms of nuclides is given on ST1. The smearing factors give the ratio of fluxes and volume of the discrete material relative to the material into which it has been smeared. Any number of smearing/condensation stages may be performed when constructing the materials present in the geometry from the discrete materials. The group structure in which nuclide cross sections are given may be the same as that of the flux data pool or it may correspond to a previous stage in the cross section preparation.

Macroscopic cross sections are given on FORTRAN unit 7 in an AUS cross-section data pool. Microscopic data for the nuclides may follow the macroscopic data or be given on an additional unit, normally 8. The cross-section data pool may contain several versions of a particular nuclide, since components of a material are identified by full 20-character names.

The cross sections of a nuclide may also be dependent on irradiation. This dependence is included within one version of the nuclide. The burnup mechanisms are obtained from the data pool which includes a bumup table giving the decay constant, fission product yields, and the simple nuclide name of the product of decay and two possible production reactions.

The cross sections required are absorption, fission and the production processes. The two production processes are normally (n,γ) and (n,2n), although the production of two isomers by (n,γ) has also been represented. The absorption cross section in AUS is that which gives neutron balance. To obtain a nuclide destruction cross section, the module adds to absorption twice the value of any (n,2n) cross section it can identify.

In most cases the required burnup information is taken from the AUS data pool. However, it is possible to include further data on fission products in an additional data set. This facility is intended for use in fission product inventory and decay heat calculations, whereas the standard data is intended for reactivity calculations. The additional data set contains burnup and decay energy information for a more complete set of fission products than that given in the cross-section library. When this data set is given, the burnup information replaces that given in the AUS cross-section data pool. The cross sections for nuclides not given in the AUS data pool are taken to be zero. A nuclide may have more than one mode of decay in this data set but not in the AUS data pool. An additional data set of 481 nuclides with half-lives greater than ten seconds has been generated from ENDF/B-IV and is available as the data set AUS.ENDF.FISSPROD.

After the burnup calculation, the ST2 data set is restarted with an updated TIME entry which gives the current burnup time. A TIME entry, an IRAD entry giving irradiations of each discrete material, and an updated nuclide composition for each discrete material are added to ST1. Any $DATA entries for the previous time which have the same module name as the previous TIME entry (normally the MIRANDA data generation module [Robinson 1986]) are added to ST1 with the material compositions, in the same order as before. If the option of remixing the macroscopic cross sections is specified, all entries in ST1 for the previous time are added to ST1 as it is assumed

that the modules which wrote these entries are to be bypassed for the current time.

When cross-section remixing is specified, the updated macroscopic cross sections are written on FORTRAN unit 7. A material is remixed if all of the nuclides comprising the material are present in the microscopic cross-section data pool and any required mixing rules are available. Otherwise the material is simply copied, and the fact is noted in the printed output. As in the case of unsmearing, a material may have passed through a number of condensing and smearing stages.

3. INPUT SPECIFICATION FOR CHAR MODULE

3.1 Data Layout

Input data is entered in free format on FORTRAN unit 1 using keywords to indicate the data type. Each keyword must begin on a new record and all data must be entered between columns 1 and 72. The module uses the input subroutine SCAN [Bennett & Pollard 1967] to process the data. Default values have been set for most variables. A list of data following a keyword may be shortened if the defaults are satisfactory. Data to be reproduced exactly have been given in upper case. Reference to an n-character word means up to n alphanumeric characters, of which the first is alphabetic.

3.2 Flux Normalisation

The flux level at which the system is to be burnt may be set by specifying a volume integrated flux, flux, power, or power density for the whole or a part of the system. The format of the entry is

FLUX (DENSITY) flux FOR name GRPS m TO n

or

POWER (DENSITY) power FOR name

where DENSITY is given only if a flux or power density is to be specified,

flux is the volume integrated flux (neutrons cm s^{-1}) or flux (neutrons cm^{-2} s^{-1}),

power is the fission power (W) or power density (W cm^{-3}),

name is $ALL, $FUEL or a 20-character name (given as two 8-character and one 4-character words) of a discrete material in the system. $ALL means the normalisation is for the whole system; $FUEL means the fuel materials; and a material name means the normalisation is for the named material only. $ALL is the default value,

m,n are the first and last energy groups of the nuclide group structure to be included in the flux normalisation. The default is all groups.

CONSTANT FLUX

is included with the POWER entry to burn at a constant flux which corresponds to the given initial power.

HEIGHT h

gives the ratio of actual volume to volume represented in the flux calculation. For example, h might be the core height in cm in an XY calculation. The default value is h=1.

Examples:

POWER 1.E+9

specifies a total power of 1000 MW.

FLUX DENSITY 1.E+14 FOR FUEL, CELLA,ORIG GRPS 5 TO 5

specifies a flux level of 1.E+14 in group 5 for the named material.

3.3 Time Step

STEP t n

specifies t the time step interval in days, and n the number of steps of length t to be taken (default 1). As the solution of the depletion equations is analytic, the time step is that at which power renormalisation is carried out. That is, the time step is limited only by the requirement that there is little change in the flux over the step. For burnup at constant flux, one step only is required unless intermediate printed output is desired.

3.4 Flux Data Pool

FLUXLIB FLn m

specifies the type of flux data pool. The integer n has the value 1 for an AUS FLUXA data pool (POW dump) on FORTRAN unit 5, or the value 2 for an AUS FLUXB (WDSN dump) on FORTRAN unit 6. If n is zero, a one point burnup is performed and the cross-section data must be given for one energy group. The default value of n is 2. The integer m is included for a multifile data pool from which no more than m files are to be read. The default is to use the last file. Thus if m is less than the number of files, the m^{th} file is used.

KINF

specifies that bumup is to be performed in a k_{inf} flux. That is, any GFAC entry on ST2 is ignored.

3.5 Cross Section and Burnup Data

XSLIB n m

specifies the FORTRAN unit number n of the microscopic nuclide cross-section data pool. The default is for microscopic data to follow the macroscopic data on FORTRAN unit 7. Where a separate data pool for nuclide data is given, this would normally be on unit 8. A separate data pool must be used if macroscopic data are to be remixed. A value m is specified in fission product inventory calculations to provide the FORTRAN unit number of an additional data set on which additional fission product bumup data are given. The data set AUS.ENDF.FISSPROD provides such data.

REMIX

is included if the macroscopic cross sections are to be remixed after burnup is performed.

FTYPE n

is included to change the fuel type for all nuclides to n. The fuel type on an AUS cross-section data pool specifies which of six possible sets of fission product yields is used. On current libraries, the six sets are for ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{241}Pu, and ^{239}Pu in a fast neutron spectrum.

FER f

is included to change the energy release per fission to f for all nuclides. The units of f may be either joules or MeV. The power in the CHAR module is fission power only.

INFER (nuc_1 f_1) (nuc_2 f_2)...

is included to change the fission energy release of specified nuclides. The data are given as the pairs nuclide name nuc_i and energy release f_i in joules or MeV.

3.6 Irradiation-dependent Nuclide Cross Sections

The microscopic nuclide data, which is normally on FORTRAN unit 8, may be a type-4 AUS cross-section data pool in which irradiation dependence replaces dependence on effective scattering cross section on a type-2 data pool. The irradiation values at which the data are tabulated are not entered in the data pool but are given by the following entry:

TABLED cell n $r_1, r_2, ..., r_n$

where cell is the 6-character cell-name for which data is tabulated as a function of irradiation,
n is the number of irradiation points, and
r_i are irradiation values in W day cm^{-3}.

It should be noted that a cell-name forms part (characters 9 to 14) of a full 20-character material or nuclide name, and that cross sections for all materials of a lattice cell are tabulated against irradiation. The irradiation values are those for the (first) fuel material in the lattice cell, which are the values printed out by CHAR in a previous burnup calculation in which the nuclide data were prepared. For a given material, the irradiation value used to interpolate in this table is the first non-zero irradiation value of any material for which characters 9 to 14 of the material name are the same as those of the given material. Thus the fuel, can and coolant data in a cell may all depend on the fuel irradiation value.

3.7 Dimension Data

The only dimension data possibly required are the maximum number of burnable materials. Other data for dimensions are obtained from the various data pools.

MAXM m
is included to alter the maximum number of burnable materials from the default value of 100.

3.8 Printed Output Options

PRINT n m
is included to control that section of the printed output not edited by nuclide (Section 3.9). If $n \geq k$, printed output of type k is produced where the print types are

0 a minimal print of materials and irradiations for each time step,
1 a print of concentrations and time average concentrations at the end of each time step for each nuclide for each material,
2 a print of group fluxes for each material, and
4 a print of one-group microscopic reaction rates for each nuclide for each material.

The default value of n is 1. The integer m is intended for use in fission product inventory calculations using additional fission product data. The values for m are

0 nuclide concentrations are printed in units of 10^{24} atoms cm^{-3},
1 nuclide concentrations are printed in grams,
2 rather than the print of type n=1 above, a full nuclide print including nuclide activities and decay power is produced.

The default value of m is 0.

3.9 Editing Facilities

Editing facilities are available to provide printouts of the concentrations of selected nuclides and the reactions for mixtures of nuclides. Spatial averages of the above quantities may also be printed. The print of reactions gives loss (absorption plus decay), fission and capture for the selected mixtures in units of events per 10^{-24} cm^3 over the burnup step. The edit facilities would normally be used in conjunction with a PRINT 0 entry.

FOLLOW n (n nuclide names)
provides n, the number of nuclides for which concentrations are to be printed, and the set of named nuclides.

REACTIONS n (m_i, (m_i nuclide names), i=1,n)
provides n the number of mixtures to be formed, m_i the number of nuclides in the i^{th} mixture, and the nuclide names which must be taken from the set named in FOLLOW.

AVERAGE n (n sets of two 4-character names)
provides n the number of spatial averages to be formed, and two names for each average which give the first halves of the first two 8-character words of the full material name. The names may also have the values $ALL which means everything or $FUEL which means all fuel materials. The materials which have the specified half-names are included in the average. The use of AVERAGE requires an appropriate convention for forming material names. Average results are printed for both concentrations and reactions.

Examples:

```
FOLLOW 6 U235 U238 PU239 PU240 PU241 PU242

REACTIONS 4 3 U235 PU239 PU241 2 U238 PU240 1 PU239 1 PU240

AVERAGE 3 FUEL CORE FUEL BLNK FUEL $ALL
```

would form averages for the core, blanket and whole reactor from the materials (FUEL,CORE01), (FUEL,CORE02), (FUEL,BLNK01), (FUEL,BLNK02) for two core cells labelled CORE01 and CORE02 and two blanket cells labelled BLNK01 and BLNK02.

3.10 START

START
is included to cause the calculation set up by the previous entries to begin.

4. DESCRIPTION OF THE BURNMAC MODULE

The BURNMAC module has been written to enable reactor burnup calculations to be undertaken using the simple assumption that cross sections for each zone of the reactor are a function of irradiation only. The module performs three functions:

(a) For initial runs only, the user's geometry description is written as a geometry data pool on FORTRAN unit 4. This includes the layout of the burnup zones in which the flux is assumed to be constant in determining burnup.

(b) A core loading pattern of fuel type and irradiation for each zone is maintained and updated. Burnup is simply a matter of advancing the irradiation map. This requires a flux distribution on unit 5 in the form of an AUS FLUXA or FLUXB data pool (the last one on the data set is used) and a macroscopic fission cross section which is obtained by interpolation in irradiation in the tabulated input cross sections on unit 10.

(c) Cross sections for each zone are obtained by linear interpolation in irradiation in the input cross sections on unit 10 and the interpolated cross sections are written on unit 11 for use by a flux calculation module.

The STATUS data pool on unit 9 is used to pass information from one call of the module to the next. The normal usage is an initial link to BURNMAC followed by alternating links to the POW diffusion module [Pollard 1974] and BURNMAC. Any serious burnup calculation would require many AUS runs and all five data pools (GM1, FL1, ST1, XS1, XS2) on units 4, 5, 9, 10 and 11, respectively, should be saved to assist in restarts. With all data pools saved, restarts require no additional input.

The following information is necessary for error recovery. The STATUS data pool is organised as a TIME entry followed by loading pattern entries and a zone flux entry for that time. (The TIME and loading pattern entries are written by one link to BURNMAC and the flux entry by the next link to BURNMAC - after the flux calculation for that time.) The zone flux entry is used only if a flux data pool is not given. Thus, if a calculation is correct to time T, error recovery is possible by using AUSYS [Robinson 1975a] to insert an end-of-file before the entry for time $T + \Delta T$ and by using the zone flux in the first link to BURNMAC.

5. INPUT SPECIFICATION FOR BURNMAC MODULE

5.1 Data Layout

The input data are read using SCAN. The data are given in the form of a number of entries each of which consists of a keyword followed by a string of data items. More than one entry may be given on each input record. The entries are of four types: (a) fuel specification; (b) geometry description; (c) bumup directives; and (d) fuel movement directives. The entries of each type are described in the following subsections. Information to be reproduced exactly is in upper case.

5.2 Fuel Specification

Fuel specification entries must be given first in all links to BURNMAC. A FUEL entry is given for each type of fuel in the reactor.

FUEL name LOADING f $i_1,i_2,...,i_n$ $r_1,r_2,...,r_n$
DENSITY

where name is an alphabetic 4-character label,
either LOADING or DENSITY must be given,
f is the fuel loading per bumup zone (LOADING) or the fuel loading per unit volume (DENSITY),
n is the number of sets of cross sections on unit 10 for this fuel type,
i_j is a position number on unit 10 of the j^{th} set of cross sections,
r_j is the corresponding value of irradiation, and
units of f and r_j are discussed below.

A REFL entry is given to identify the non-fuel materials on unit 10.

REFL $l_1,...,l_n$

where n is the number of materials on unit 10 to be passed directly on to unit 11, and

l_j is the position of the j^{th} material on unit 10.

The last (optional) entry is

FACTORS h u

where h (default 1) is the ratio of actual volume to volume represented in the flux calculation, and
u (default 1) is the ratio of the user's burnup units to power-time.

A value h is specified to allow real power values. Thus, in an XY model, h would be the core height. In an RZ model reflected at the mid-plane, h would be 2.

The units used are largely at the discretion of the user. However, the following system is recommended as a basis. Throughout AUS, macroscopic cross sections are in cm^{-1} and length is in cm. With the power in watts and the time in days (as specified by the POWER and STEP entries below), the natural units for irradiation r_j are watt-days (Wd) per fuel loading unit. Thus with f in grams or g cm^{-3}, r_j is in Wd g^{-1} or MWd t^{-1}. Alternatively, f may be fuel elements or fuel elements per cm^3, in which case r_j is in Wd per fuel element. The factor u may be used to make an additional transformation, *e.g.* to fissile content in grams for which u would take a value $\sim -1.27 \times 10^{-6}$. (The value is negative since mass decreases as irradiation increases.) Then r_j would be in grams per element as used with the HIFAR research reactor.

Conventionally the first set of cross sections for a fuel type is xenon free and used only for irradiation values R, for which $R < r_1 < r_2$ (or $R > r_1 > r_2$). Cross sections for values of R in the range $r_1 \leq R < r_2$ (or $r_1 \geq R > r_2$) are obtained by extrapolation using the data at r_2 and r_3.

5.3 Geometry

Geometry data are given only once in an initial BURNMAC run. The data specify the mesh intervals and boundary conditions used in the flux calculation, the layout of burnup zones and reflector materials and an initial fuel loading pattern. Where appropriate, the data entries are of the POW type.

The mesh intervals and boundary conditions are given exactly as in POW using XM, YM, RM, ZM, RM(SPHERE), as appropriate, where each entry is of the form

XM d_L, $\delta_1,\ldots,\delta_n$, d_R

where d_L, d_R are left and right boundary conditions, and δ_i is the i^{th} mesh interval of a set of n intervals.

As in POW, REG entries are used to specify the reflector material layout and may also be used to specify the burnup zones. The form is

REG MX $i_1,\ldots,i_I$ MY $j_1,\ldots,j_J$ M(m)

or REG MR $i_1,\ldots,i_I$ MZ $j_1,\ldots,j_J$ M(m)

where the i and j are interval numbers, m is the m^{th} material as given on the REFL entry for $m > 0$, and $|m|$ is the m^{th} burnup zone for $m < 0$. Use of REG entries becomes tedious for large numbers of burnup zones. An alternative specification in terms of channel boundaries may be used for regular layouts. The entries are

LX mx p_1, $p_2,\ldots,p_{mx+1}$
LY my q_1, $q_2,\ldots,q_{my+1}$

where mx is the number of channels in the X or R direction, the intervals k satisfying $p_i \leq k < p_{i+1}$ are in the i^{th} channel in the X or R direction, and similarly for the Y or Z direction. The burnup zones are then numbered from left to right and top to bottom for each channel which does not contain reflector material.

Example:

```
REG  MX 1(1)10  MY  1(1)10  M(1)
     MX 1(1)6   MY  1(1)6   M(0)    ... to clear
     MX  5 6    MY   5 6    M(1)
```

```
LX 3 1 3 5 7  LY 3 1 3 5 7
```

The result is eight burnup zones:

```
1 1  2 2
1 1  2 2
3 3  4 4  5 5
3 3  4 4  5 5
6 6  7 7  8 8
6 6  7 7  8 8
```

The initial loading pattern is given by TYPE, IRRAD, NUMBER entries:

TYPE $i_1,...,i_{NZ}$

where $i_1,...,i_{NZ}$ are the fuel types in each of the NZ burnup zones. The fuel type is a number giving the position in the set of FUEL entries.

IRRAD $\nu_1,...,\nu_{NZ}$

where ν_1 etc. are the irradiation values for each burnup zone. The default values are the r_1 values for the corresponding FUEL entries.

NUMBER $n_1,...,n_{NZ}$

where n_1 etc. are identifying fuel element numbers. The default values are 1 to NZ. Where there is a one to one correspondence between burnup zones and fuel elements, the concept of a fuel element number which identifies the element throughout fuel movements should be useful. It may be ignored by the user.

No further data may be given in an initial BURNMAC run. The number of materials written on unit 11 is NZ plus the number of reflector materials.

5.4 Burnup Directives

POWER p

sets the power level.

STEP Δt n

where Δt is the time step and n (default 1) is the number of steps each of length Δt.

The STEP directive causes burnup to take place. For $n > 1$, the power map is recalculated at the beginning of each step using the updated fission cross sections for each zone.

5.5 Fuel Movement Directives

The module provides facilities for moving fuel elements and also for saving elements for later reloading.

DISCHARGE [NO] / m n ν

where NO is given only if elements are referred to by number rather than position,
/ is the burnup zone or fuel element number to be discharged,
m is the fuel type to be loaded,
n is the new fuel element number (default is 1 more than last number generated), and
ν is the irradiation value (default is r_1 value for fuel type m).

Alternatively, if m is negative, the fuel element number $|m|$ which was previously saved is loaded.

SAVE NO / m n ν

is identical to DISCHARGE except that the element is saved for reloading at some later time.

RELOAD NO $l_1, l_2,..., l_{NZ}$

where NO is given if elements are referred to by number rather than position, and
l_i is complete set of old fuel positions or fuel element numbers to be loaded in zones 1 to NZ.

MOVE $(i_1,j_1)...(i_n,j_n)$

causes the elements in positions i_1 to i_n to be moved to positions j_1 to j_n.

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