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LUCAS HEIGHTS

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A TWO DIMENSIONAL COUPLED HYDRAULIC-NEUTRONICS CODE FOR CALCULATING THE EQUILIBRIUM AND TIME DEPENDENT BURNUP STATES OF PRESSURE TUBE TYPE REACTORS

by

E.W. HESSE

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ABSTRACT

The Equicore II code calculates the burnup, power distribution and coolant conditions of pressure tube type reactors for both the equilibrium and time dependent core conditions.

The code has been designed to be both easy to use and economic in computing time. To achieve this, the overall core burnup and hydraulic performance is obtained by considering only a few representative fuel assemblies and by using flux distributions derived from an RZ-diffusion calculation, based on volume elements containing the average properties of these representative fuel assemblies.

A separate RZ supercell calculation model is available to determine the detailed flux and power variation between selected fuel assemblies and the average properties within radial zones of the core.

The code also incorporates a user-oriented facility which allows, at any stage of the burnup calculation, the determination of such things as space dependent temperature coefficients and time dependent variation of xenon concentration.

This report describes the models used in the code and gives the data requirements.

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BURNUP; COMPUTER CALCULATIONS; COOLANTS; CRITICAL HEAT FLUX; DISTURBANCES; E CODES; EQUILIBRIUM; FUEL ASSEMBLIES; HYDRAULICS; IBM COMPUTERS; MATHEMATICAL MODELS; NEUTRON FLUX; POWER DENSITY; PRESSURE TUBE REACTOR; REACTOR CORES; TIME DEPENDENCE; TWO DIMENSIONAL CALCULATIONS; XENON

CONTENTS

CONTENTS (Cont'd.)

1. INTRODUCTION

The EQUICORE II code calculates the burnup, the power distribution and the coolant conditions of pressure tube type reactors for both the equilibrium and the time dependent core conditions. It is an extension of the EQUICORE code (Bicevskis and Hesse) originally developed for a joint AAEC and UKAEA design study of a natural uranium SGHWR*. The code was originally designed to calculate the equilibrium core conditions of an on-load refuelling reactor. However, because many of the computer-time-saving features in the EQUICORE code could also be extended to calculate enriched batch-refuelled SGHWR's under design at that time by the UKAEA, it was requested that the code should be extended to include time dependent burnup calculations. During this work the equilibrium model was extended to cover more complex fuel management problems such as are encountered in CANDU (bi-directional refuelling) and ATUCHA (radial shuffling) reactors.

Although most of the coding was completed in the UK, further work was necessary after the author's return to Australia to include features which were necessary to study various aspects of the SGHW reactor. Other calculational models and improvements were then included for the assessment of tenders for the Jervis Bay reactor. Time dependent xenon variation was included to study possible axial instability in PWR's and a booster rod model was developed for calculating CANDU xenon override capability. Improvements to fuel temperature calculation and to the hydraulic models were also made. Finally, changes in the operation of the AAEC IBM360 computer necessitated changes to the whole structure of the code to make it more efficient for a multi-processing environment.

Although it is intended to develop the code further, particularly for calculations involving light water reactors, it has been decided to describe the code in its present form. This version which is fully operational on the IBM360 computer, is given the name of EQUICORE II to distinguish it from the earlier code, EQUICORE.

2. EQUICORE II CALCULATIONAL MODEL CONCEPT

The calculational models used in EQUICORE II have been developed with the object of producing a computer code which would be easy to use, economic in computing time and allow one to obtain an accurate assessment of the performance of many core designs or fuel management options. The earlier EQUICORE code was used to examine the effect of changing a wide range of core parameters

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* Steam Generating Heavy Water Reactor

to determine the optimum design of a natural uranium SGHWR core. During the transient phase between EQUICORE and EQUICORE II the code was used to assess rapidly a number of reactor designs operating under various conditions. Thus, for both these roles, the code has been required to produce information which would enable the rapid but sufficiently accurate assessment of the overall core performance and the detailed performance of typical highly rated channels.

For speed in execution the EQUICORE II code employs models which determine the burnup and hydraulic performance of the core by considering only a few representative fuel assemblies and by using flux distributions derived from an RZ diffusion calculation, based on volume elements containing the average properties of these representative fuel assemblies. A separate RZ supercell calculation model is available to determine the detailed flux and power variation between selected fuel assemblies and the average properties within radial zones of the core. These models, explained in detail in the next section, are more economical in computer running time and core storage than the more conventional explicit channel flux method normally used for SGHWR calculations. EQUICORE II uses only two dimensions with relatively few smeared volume elements requiring few mesh points in the diffusion calculations and treats only a few representative fuel assemblies for the burnup and hydraulic calculations. On the other hand the explicit treatment of every assembly in the core requires three dimensions (MAGOG, Hopkins and Oakes 1967) or synthesised three dimension (CALEB, Robinson and Fayers 1969) for both the diffusion calculation and for the determination of the burnup and coolant density distributions. The adoption of a two region RZ supercell model allows the use of sufficient mesh points to enable sufficiently accurate determination of the flux variation across the selected homogenised fuel assemblies without involving long program running times. In addition to the conventional time-dependent burnup calculation method, EQUICORE II retains the calculational model which allows the rapid determination of the fuel burnup and core power distribution at equilibrium of an on-load refuelled reactor such as CANDU.

It is explained in detail in the following sections that the use of these fast calculational models implies that the loss of detail by use of RZ geometry is small. This assumption has been shown to be valid for assessing the performance of large SGHWR reactors with many fuel channels and is superior to the conventional method where a coarse mesh must be used to obtain reasonable computer execution times, (Hesse 1972).

Reactor assessment often involves the examination of core performance at conditions other than at full power and at various times in the burnup history of the core. As EQUICORE II employs calculation models that require little computing time it has been possible to include a facility which allows the user to request a perturbation calculation at any stage during a burnup calculation. This enables the determination of such things as space dependent temperature coefficients and the time dependent variation of xenon concentration in the core.

A general description of the EQUICORE II calculational models is given in Section 3. Sections 4 and 5 provide an outline of the code structure together with a description of the calculation steps and the equations used in the code. These serve to fill in some details not covered in the EQUICORE II model description. In Section 6, a description of the input data required by the code is given.

3. DESCRIPTION OF THE EQUICORE II CALCULATIONAL MODELS

3.1 Reactor Core Description in EQUICORE II

The code sets up the outer dimensions of the core in RZ geometry from information of the height of the active core, number of fuel channels and cross sectional area of a fuel lattice cell including associated moderator (Figure 1). The specification of the associated radial and axial reflectors are given in conventional RZ diffusion calculational form. However, to understand the division of the active core it is necessary to define, with reference to Figure 1, the following concepts.

Radial zones. The fuel channels of the reactor core are grouped radially into cylindrical sections called 'radial zones' extending the full height of the active core. The number of radial zones (typically 6 to 9) is chosen such that they adequately describe the variation of power burnup and coolant density insofar as they significantly affect changes in the nuclear properties of the materials across the core. The thicknesses of the radial zones are specified in terms of number of channels in each zone and do not necessarily have to correspond to any physical identifiable regions of the reactor.

Fuel channel types. The fuel channels making up the radial zone are grouped into a specified number of channel types for each of which the power, burnup and coolant density distribution has to be determined. A 'fuel type' represents all fuel channels in a radial zone which contain fuel having the same enrichment, burnup and future fuel movement requirements. For example a CANDU reactor requires at least two fuel channel types per radial zone to represent the bi-axial loading, whereas a commercial SGHW reactor requires nine channel types to represent the typical nine burnup states of the batch refueling scheme. Although the EQUICORE II model does not position a fuel channel type in its exact physical location and assumes that the channels are randomly distributed in the radial zone, it has been shown (Hesse 1972) that it represents to a very good approximation the typical or average fuel channel in that radial zone. The mixture of channel types within each radial zone can be varied to allow for changes in the constitution of actual fuel assemblies used near an enrichment or reflector zone. The number of actual channels represented by a fuel channel type is specified in terms of the fraction of channels of that type to the number of channels in the radial zone and is defined as the channel type volume fraction.

Fuel assembly. In EQUICORE II a 'fuel assembly' is taken to mean the assembly or group of fuel pins which is loaded or moved in fuel channels as one unit during the whole fuel management cycle. For an SGHWR, where no axial fuel moves take place, the assembly length is equal to the active core length. In a typical case of CANDU a fuel assembly length is 1/6 of the active core, being equivalent to the two fuel bundles which move as a unit. All fuel assembly lengths must be the same and the length is specified in the input data in terms of the number of fuel assemblies per fuel channel.

Fuel segment. To allow for the variation of power and burnup along a channel the length of fuel assembly in a fuel assembly location is subdivided further into a specified number of 'fuel segments'. All the fuel segments have the same length, which is the smallest interval for which the average power, burnup and coolant density is determined.

Radial zone elements. To perform the RZ diffusion calculation the radial zones are divided axially into slices, corresponding to the fuel segments, to form 'radial zone elements'. The nuclear cross sections used for the radial zone elements (Σ_{RZE}) in the diffusion calculations are the average of the cross sections of the fuel segments (Σ_{FS}) of all the fuel channel types within each radial zone element. Cross sections for reflector regions are read in separately.

Although the radial zone elements can be subdivided for a more accurate finite difference solution, only the average neutron flux of the radial zone element (Φ_{RZE}) is used when deriving the channel power distributions.

3.2 Channel Power Distribution Calculation

The axial power distribution of each channel type in the radial zones is calculated from the fuel segment fission cross section ($(\Sigma_{FS})_f$) and normalised Φ_{RZE} applying along the channel. In calculating the power or fission rate distribution, the standard EQUICORE II model makes the assumption that each fuel segment in the same radial zone element 'sees' the same flux as determined by the full core diffusion calculation based on the average properties in each radial zone element. This assumption is reasonable if the fuel properties of the channel types are not too different or if the real fuel channels are mixed together sufficiently so that the local flux is not significantly affected by the individual fuel. In the case of a typical SGHW reactor where the ratio of the new fuel fission cross section to the average batch fission cross section is approximately 1.13, the ratio of the new fuel flux to the average batch flux is approximately 1.04 (Hesse 1972). It is assumed in the burnup calculation that this small flux error will be balanced out as the fuel burns up.

EQUICORE II includes an option which estimates the approximate flux variation between channel types in a radial zone, and flux-weights all Σ_{FS}'s when determining the volumetric average Σ_{RZE}'s. The flux variation is obtained from an interpolation of precalculated library values of flux weighting factors as a function of a K_∞ ratio and supercell size. Using these weighted $(\Sigma_{FS})_f$ leads to channel power distribution approximately adjusted for the flux variation between a specified fuel channel type and the other channel types in a radial zone. This calculation therefore allows a better estimate to be obtained of highly rated channels.

3.3 Hydraulic Calculation

Using the channel power distributions the coolant flowrates through each fuel channel type and associated feeder and riser sections can be determined to satisfy a given riser exit pressure, feeder inlet pressure and flow inlet enthalpy. An option is also available to determine the flow for a given riser exit quality and pressure.

Except for the fact that only representative channels are calculated these calculations are similar to those used in conventional explicit channel programs such as CALEB.

The calculation provides the average coolant density in each fuel segment of each channel type to be used for the determination of coolant density dependent cross sections. Critical heat flux ratio correlations are also available to determine the dryout margins of the channel types. The hydraulic model used in EQUICORE II is an improvement over the hydraulic model used in the earlier EQUICORE which did not have riser and feeder sections. For details of the hydraulic model and methods used to determine voidage, slip, two phase multipliers etc. see the HYDRO code(Green 1973). The hydraulic section in EQUICORE II has been arranged so that the hydraulic calculation need not be carried out at every flux iteration. For reactors having effectively a zero void coefficient it is possible to specify that the hydraulic calculation should only be done after a converged power distribution has been obtained and, only at specified steps in the burnup calculation. The hydraulics can also be bypassed completely.

3.4 Burnup Calculation

There are two burnup calculational models within EQUICORE II. One model is used to perform the more conventional time dependent calculation (marching process) used for off-load refuelled reactors, and the other is used to obtain the core equilibrium state applying to a continuously on-load refuelled reactor. Both models use similar methods and assumptions for determining the burnup reaction rates.

The EQUICORE II burnup models are based on the use of the UKAEA PATRIACH (Hicks 1967) type cross section library. Each fuel assembly type and associated pressure tube and moderator (Figure 1(c)) is represented in terms of homogenised two group cell cross sections as a function of irradiation (MWD/TU) and coolant density. Cross section correction factors are included in the library to allow for changes in xenon concentration, fuel temperature, and any additional poisons. The calculation used to estimate the fuel segment temperatures is based on a relationship relating the local fuel rating burnup and physical pin dimensions to the temperature (see Section 5.2.2 (i)(d)). The xenon concentration calculation is based on a simple relationship between the concentration and the fuel segment fission rate and effective xenon cross section (see Section 5.2.2 (i) (b)). Thus every axial fuel segment in all channel types in the core is represented by its library fuel assembly type and its irradiation state, rather than by its isotopic compositions. When fuel is moved to another fuel assembly location the fuel type and irradiation state of each segment is simply transferred. The irradiation state of a particular fuel segment is advanced over a given time interval by calculating the fission rate as a product of Σ_{RZE} and the particular $(\Sigma_{FS})_f$ obtained via fuel type and interpolation of the library for irradiation, and coolant density. The average coolant density of the segment is determined from the channel hydraulic calculation of that particular fuel channel type. If the flux weighting option described in 3.2 is used, the $(\Sigma_{FS})_f$ used for the burnup advancement would have been suitably weighted.

3.4.1 Time dependent burnup

In the time dependent burnup model the user specifies the fuel assembly types to be loaded into all fuel assembly locations in the core. The code allocates zero irradiation states to all fuel segments and then determines the core RZ flux distribution for a critical poison loading using Σ_{RZE}'s equivalent to the zero irradiation of fuel segments. In the iterative procedure for determining the poison loading the code will also establish cross sections which have been adjusted for the correct fuel temperature, xenon, and coolant density in each fuel segment.

When a converged flux distribution for the specified core k_{eff} has been established, the fission rate and hence fuel segment irradiation can be determined. All fuel segment irradiation states are then advanced and the process repeated until burnup has reached a state where the core cannot be made critical (i.e. all poison removed). Once this state has been reached the code branches back to read more input data to obtain further refuelling instructions. This will normally be in the form of some fuel movement to other assembly positions in the core together with some loading of new fuel. Before re-entering the burnup calculation process, the code will give details of the burnup achieved in the rejected fuel assemblies.

The specification of fuel movement is flexible allowing fuel assemblies to be inverted. Repetitive features are used to reduce input data specification. In addition it is possible to specify that the core should be operated for a set time interval rather than be operated until the core can no longer be made critical.

3.4.2 Equilibrium burnup

The calculational model used to determine the equilibrium burnup was developed for the earlier EQUICORE code. The advantages of the model and the assumptions used are explained in detail in the earlier EQUICORE report (Bicevskis and Hesse 1969). The main changes incorporated in EQUICORE II have been to increase the flexibility of fuel movement in the core and to allow more than one fuel channel type in a radial zone.

The object of the equilibrium model is to obtain both the core power distribution and the speed of fuel movement through the core which would keep the core critical. The speed of fuel movement can then be related to the core average fuel burnup.

The equilibrium model assumes that the average flux distribution in the core has reached a fixed state. This is taken to mean that although fuel is being continuously loaded and moved in the core, the average irradiation of the fuel segments in any radial zone element remains constant. Therefore the average Σ_{RZE}'s remain constant and can be used in the diffusion calculation to determine the average macroscopic flux distribution and hence Φ_{RZE}'s. The equilibrium burnup model assumes that the constant Φ_{RZE} can be used as the burnup flux level 'seen' by the fuel segment while in the radial zone element. The flux level is the same for all fuel segments of whatever type within a radial zone element and is not effected by changes in cross sections as the fuel segment burns up. The average Σ_{FS}'s are obtained by using a time averaging procedure of the fuel segment cross section variation while it dwells in that radial zone element.

To perform an equilibrium burnup calculation the user must specify the paths which various fuel assemblies take through the core together with their fuel assembly library type. The path is specified in terms of a series of 3 or 4 digit numbers indicating the fuel assembly location in the core (see Section 3.1). When determining the Σ_{RZE}'s to be used in the diffusion calculation the code must trace each specified fuel assembly through the core to determine its burnup history from which the time average Σ_{FS}'s can be determined. The code will load the first fuel assembly into its first specified core assembly location and allocate zero irradiation states to the fuel segments making up the assembly. Using Φ_{RZE} for a previous iteration and $(\Sigma_{FS})_f$ equivalent to the zero irradiation state, the irradiation of each fuel segment of the assembly can be advanced for a small time interval. (This time interval is estimated at the beginning of a complete sweep such that the combined effect of all the fuel movements through the core will make the core just critical). Using $(\Sigma_{FS})_f$ applicable to the new irradiation states and the same Φ_{RZE} the fuel is again advanced in burnup. The procedure is repeated for a set number of steps after which the fuel assembly is moved together with its irradiation distribution to its next specified core location where the whole procedure is repeated. In this way the complete path of each specified fuel assembly is traced through the core. During this process the Σ_{RZE}'s are obtained as before by volume averaging all the channel type Σ_{FS}'s derived during the burnup calculation. The flux weighting calculation can also be included.

The Σ_{RZE}'s derived during the burnup calculation are used in the RZ diffusion code to determine a better estimate of the core flux distribution and the core K_{eff}. The code iterates between the cross section and flux calculations to determine the burnup time interval; this produces a critical core as well as a consistent core flux distribution. The converged calculation provides a burnup of each fuel assembly.

The essential difference in the cross section-flux calculation between the equilibrium and time dependent models is in the way the average cross sections are determined for the RZ diffusion calculation. For the time dependent burnup, the Σ_{RZE}'s are the average of the instantaneous Σ_{FS}'s of all the fuel channel types in that element whereas in the equilibrium model the Σ_{RZE}'s are the average of the time averaged Σ_{FS}'s of all fuel channel types in that element. Thus in the equilibrium model the fuel burnup state is advanced while the cross sections are being determined because the burnup flux remains constant. In the time dependent model the burnup is advanced using the calculated flux distribution at each time interval.

For the equilibrium calculation, in addition to the specification of the channel type volume fractions of each fuel channel type in a radial zone, the ratios of fuel residence time in each channel type can be specified. The absolute values of the fuel residence times are determined by the code to satisfy the reactivity requirement. This allows, for example, slower feed rates to be specified for inner core channels to obtain irradiation flattening as in the CANDU reactor. A facility is also available to convert the irradiation distribution results of a converged equilibrium calculation to form the start of a time dependent burnup calculation.

3.5 Supercell Calculation

The EQUICORE II code includes a facility which calculates the variation of flux between a specified fuel channel type and the surrounding fuel channel types within any radial zone. By choosing the channel type with the highest rating in that radial zone this calculation allows one to determine an improved power peaking factor occuring in a batch in that zone.

The calculation is performed using a two dimensional RZ supercell approach where the specified channel, in the form of a homogenised cell, is surrounded by an annulus containing the average properties of the remaining channel types. The model is similar to a 2 radial zone EQUICORE II calculation except that the outer boundary of the surrounding region has reflective boundary conditions thus representing an infinite array of fuel channel types making up the batch refuelling scheme. The axial geometry similar to that used in the main calculation is used in the supercell calculation.

By normalising the power distribution of the supercell calculation to give the same average power as was obtained in the radial zone, the channel power of the particular channel type can be obtained. Although the surrounding channels have been smeared together, it has been shown (Hesse 1972) that for an SGHWR the model leads to accurate estimates of batch peaking factor.

The supercell calculation can be requested at any stage in the normal time dependent burnup calculation. During the supercell calculation the normal iteration procedure between the cross section and flux calculations is carried out, as well as the hydraulic calculation of each channel type making up the batch of channel types.

To assess the power rise in a refuelled channel in a continuously on-load refuelled reactor, this supercell model has been adapted for the equilibrium type calculation. This adapted supercell model now replaces the earlier EQUICORE centre channel calculation which had become impractical with the inclusion of a more complex fuel management calculation. If a supercell calculation is requested following an equilibrium type calculation the supercell calculation is set up in such a way as to simulate the effect of only one channel being refuelled in a large region whose average properties remain constant. The specified fuel channel type in this case is surrounded by a region containing the average cross sections which were used in the diffusion calculation for the specified radial zone. The supercell radius in this case is specified and the power distribution is normalised such that the power at the outer edge of the supercell is equal to the average power in the radial zone.

By normalising the power at the edge of the supercell the assumption is made that the disturbance caused by refuelling one channel does not affect the rest of the reactor. This assumption gives reasonable results and is simpler than treating the whole reactor as was done in the earlier EQUICORE centre channel model.

The time dependent supercell radius is normally made equivalent to the number of channel types within a radial zone. However the number of fuel channels making up the supercell can be specified on radial zones where many similar channel types would lead to misleading results. This specified supercell size would also be used for the flux weighting calculation.

3.6 Perturbation Calculation Facility and Time Dependent Xenon Treatment

To examine the core behaviour at conditions other than full power, the code allows the user to branch to calculation routines which perturb some operating conditions and calculate the resultant effects. For example, a user may wish to examine the power distribution and possible change in reactivity at some stage of the reactor's lifetime if the core power and coolant flow are reduced. As it is difficult to forsee the full range of calculations that will be required, the EQUICORE II code has been arranged to make it relatively simple to code and incorporate the special purpose calculation or routine into the main perturbation routine at the time it is required. If they are likely to remain useful these routines are then incorporated permanently in the code and can be expanded later if required.

For many perturbation studies it is necessary to allow for the change in reactivity due to the variation of xenon concentration with time. In the normal EQUICORE II calculation the equilibrium xenon concentration is built into the cross sections and the cross sections are only adjusted for a change in equilibrium concentration with change of power. Thus for the cases where the change in xenon with time is required for perturbation studies, subroutines have been made available which keep account of the xenon and iodine concentration with time.

The perturbation routine can be called up at any time during an EQUICORE II calculation by the use of the instruction key word PERT followed by any necessary perturbation data. The code will then branch to the perturbation routine when the necessary changes to normal data are made (i.e. reactor power or drum pressure) and the various calculation routines called to calculate the new result. Special subroutines are available which make the creation of a perturbation routine relatively easy, requiring only a general knowledge of the working structure of the EQUICORE II code (see Section 5.5)

A perturbation routine has been included permanently in the code which calculates:

(a) the new core conditions for a change in core power,

(b) the variation of xenon and power distribution during the return to full power using booster rods (or removal of absorber) and

(c) the xenon axial oscillation after a power perturbation.

4. OUTLINE OF EQUICORE II CODE STRUCTURE

Before proceeding with a description of the calculation steps and equations used in the code, a brief description of the code structure is given and the main sections and subroutines affecting the calculational process are pointed out.

Although EQUICORE II is a large code made up of over 90 subroutines it consists of four distinct calculational or working sections. These are:

(a) Cross section preparation and burnup calculation.

(b) Two group RZ diffusion calculation.

(c) A flux editing and channel power distribution calculation.

(d) A pressure tube hydraulic calculation.

In addition to these main sections the code has large sections for automatically setting up the EQUICORE II calculation model interpreting the fuel management instructions and for providing the data required by the working sections. For example most of the data required by a conventional diffusion code (mesh interval dimensions, boundary conditions, region data, etc.) are automatically generated by the code from simple information such as number of fuel channels, height of core, number of radial zones, etc.

During the development of EQUICORE II it became more convenient to reorganise the codes into separate sections to cover each phase in the calculation. These sections were arranged to be virtually independent of each other, to make further development easier. Also the old method of using fixed storage for large arrays of variables in the code was inflexible. The coding was thus changed to use variable dimensioning in conjunction with the DARRAY (G.W. Cox DARRAY NARRAY subroutines for core management in FORTRAN (unpublished AAEC report)) facility to allow dynamic allocation and release of storage in each working section. Each section of the code has been arranged to obtain its required storage, read its required information from disc, perform its calculation task, write its calculated information on disc and then release its storage for use in the next section. Under the IBM360 MVT operation system there is little penalty for the time used during these input and output operations. This method, however, has the advantage that it not only uses the minimum core storage but also that each section of the code can be modified and developed separately once a set of data has been established on disc. Temporary disc files are used for this purpose, (FORTRAN unit numbers 10 to 18).

Figure 2 shows the arrangement of the main parts of the code. To keep the figure reasonably simple only the outline of the sections and subroutines effecting the calculational sequence are shown. The following is a brief summary of the main purpose of each section and subroutine. More details are given in the next section.

The working sections are controlled by either CALC (for burnup calculations) or PERT (for perturbation calculations) and the reading and setting up sections are controlled by DATAIN. The sections on the right of the dotted line require their own special storage and are called via an intermediate subroutine which sets up the storage and brings in and dumps out information.

The main working sections required for both a burnup and perturbation calculation are:

- CROSEC to perform the fuel assembly burnup calculation and to obtain Σ_{FS}'s and Σ_{RZE}'s used in DIFSEC and EDISEC. These cross sections are corrected for the calculated fuel temperature and xenon concentration.
- DIFSEC to perform the RZ diffusion calculation to obtain the core k_{eff} and flux distribution.
- EDISEC to obtain the normalised Φ_{RZE} from the results of DIFSEC, and

to calculate the channel type power distributions.

- HYDSEC to calculate the coolant flow and density distribution in each channel type.

The main burnup calculation control subroutines are:

- CONTRL to estimate and adjust poison concentration or equilibrium fuel residence time to achieve a critical core.
- MOVE to advance the core burnup time.

For a perturbation calculation additional subroutines are available. These are:

- PXEN to calculate the time and flux dependent xenon and iodine concentrations and to perturb the xenon concentration.
- PCONT to estimate and adjust poison concentration or Σ_{RZE}'s (booster reactivity) to achieve a critical core.

The part of the code concerned with the reading and setting up of the problem is made up of:

- DIFDAT to provide core geometry data required by DIFSEC.
- MANDAT to interpret fuel management data required by CROSEC.

(also used to give summary of rejected fuel assembly fissile concentration)

- HYDDAT to provide data for hydraulic calculation.
- LIBRA to read in basic cross section library for use in CROSEC.
- GUESS to provide a flux distribution guess required by DIFSEC and to set up initial cross section data for EDISEC.
- SUPDAT to set up new geometry and radial zone data required for a supercell model.

In addition to these sections there are subroutines to read free format input data (RIED) and to write out (DUMPIT, using FORTRAN unit number 20) and read in (COMIN, using FORTRAN unit number 8) all relevant EQUICORE II information to and from disc for possible later restart.

5. DESCRIPTION OF CALCULATIONAL STEPS AND EQUATIONS USED IN EQUICORE II

5.1 General

The sequence of operations adopted during the two burnup model calculations as well as in the supercell and perturbation calculations are in many ways similar. Therefore the sequence used in the time dependent calculations will be first described in detail followed by a description of the difference in the other calculations. As many of the equations used are similar to those given in the earlier EQUICORE report (Biceviskis and Hesse 1969) reference will be made to section 5.3 of that report for those areas where no significant changes have been made, and only the new method or equations given in this section.

5.2 Time Dependent Burnup Calculation

The time dependent burnup calculation can be considered to be made up of four distinct parts within which there are a number of steps.

5.2.1 Reading and preparation of data

This part of the operation takes place in DATAIN (see Figure 2). All input data, except that of the cross section library (normally residing on disc), is by means of free format and key words. Data is specified in a similar manner to that in the earlier EQUICORE report. Details of required data are given in Section 6. The input and preparation consists of the following steps:

(i) In the DIFDAT section the radial zone element and geometry specifications (radial, axial mesh, extrapolation lengths) are set up for use in the diffusion calculation based on read in core dimensions and from the number, size and mesh allocation of radial zones. Reflector and any additional radial zone element cross sections are also read in.

(ii) In the MANDAT section the fuel channel type arrangements are set up according to the specified number of fuel channel types and from the channel type volume fractions in each radial zone. The fuel assembly locations are set according to the specified fuel material types and the initial irradiation of the fuel segments are set to zero. A check is made that every fuel assembly location is loaded.

(iii) If a hydraulic calculation is specified, the hydraulic model is set up and the feeder, channel and riser properties are read in and stored in the HYDDAT section. Feeder and riser data can be specified for each radial zone.

(iv) In the LIBRA section the cross section library is read in and stored. The library is checked against the fuel materials required.

(v) An initial guess of the core flux distribution is calculated in GUESS based on a cosine axial distribution and a flattened cosine radial distribution. For equilibrium type calculations starting values of Σ_{RZE}'s are provided from the cross section library based on the specified guess of average core burnup. A guess of the coolant density up the core is also generated for coolant dependent cross section calculations.

(vi) Other miscellaneous data such as burnup time step interval, poison concentration and code print out instructions are read in.

(vii) Print out of the interpreted data to be used by the fuel burnup, diffusion and hydraulic sections of the code is given. The core storage required by each calculation section is also provided.

5.2.2 Calculation of the power distribution for a critical core at a burnup time step

Control is passed to CALC where the following steps occur:

(i) In CROSEC the Σ_{FS}'s of each channel type are obtained for the particular fuel assembly type via interpolation of the library for irradiation and coolant density. Using the channel power distribution from the previous iteration, the fission rating of each fuel segment is used to determine the fuel temperature and xenon concentration. The cross sections can then be adjusted for changes from the library reference values of temperature and xenon concentration. For the first iteration before channel rating distributions are available no corrections are made to the library values.

The methods used for interpolating and correcting the cross sections are the same as in EQUICORE except for the following:

(a) The plutonium content correction with coolant density change, has been removed because it was not entirely satisfactory and it made the cross section library more difficult to prepare. The correction is only of interest for natural uranium systems having a high positive void coefficient.

(b) The correction made to the absorption cross section (Δa) to allow for the difference in equilibrium xenon concentration at a point (XE_p) from that of the xenon concentration in library cross sections (XE_r) is now approximated from

$$\Delta s = (XE_p - XE_R)\ \delta_{Xe}$$

where,

$$XE_p = \Phi\ \gamma_{XI}\ [1.0/(\lambda_{Xe} + \delta_{Xe}\ (\Phi_{RZE})_2)] \qquad \ldots(1)$$

$$XE_r = \Phi\ \gamma_{XI}\ [RR/(\lambda_{Xe} + \delta_{Xe}\ (\Phi_{RZE})_2\ {*}RR)] \qquad \ldots(2)$$

$$\Phi = (\Phi_{RZE})_1 * (\Sigma_{FS})_{f1} + (\Phi_{RZE})_2 * (\Sigma_{FS})_{f2}$$

$(\Phi_{RZE})_1$ = Radial zone element fast flux

$(\Phi_{RZE})_2$ = Radial zone element thermal flux

$(\Sigma_{FS})_{f1}$ = Fuel segment fast fission cross section

$(\Sigma_{FS})_{f2}$ = Fuel segment thermal fission cross section

γ_{XI} = Effective equilibrium fractional xenon yield

λ_{Xe} = Xenon decay constant = 2.11×10^{-5}

δ_{Xe} = Xenon thermal cross section (lattice cell averaged)

RR = Ratio of fuel rating used to derive the cross section library to that of the fuel segment rating.

(c) *The temperature* correction is now made to all cross sections rather than applying a simple Doppler effect to the fast absorption cross section. The cross section library now contains corrections to be made to each cross section for a change in fuel temperature from that of the library temperature (see library preparation Section 6.6). The corrected cross section(Σ_{FS}) is now derived from:

$$\Sigma_{FS} = \Sigma_L \ (A * DELT^2 + B * DELT + 1) \qquad \ldots(3)$$

where A and B are library coefficients interpolated for irradiation and coolant density,

Σ_L = uncorrected interpolated library cross section

$DELT = \sqrt{T_{FS}} - \sqrt{T_L}$

T_{FS} = Absolute temperature of fuel segment

T_L = Absolute temperature for which the library was prepared.

No correction is made for the diffusion coefficient and the same correction is applied to both Σ_f and $\mu\Sigma_f$. If A is set to -1., the temperature correction for that cross section is bypassed.

(d) *The mean* fuel segment temperature (T_{FS}) is now calculated more accurately from:

$$T_{FS} = 4/9 \ (T_{FS})_c + 5/9 \ (T_{FS})_s$$

where

$(T_{FS})_c$ = mean fuel segment centre temperature (^{o}C)

$(T_{FS})_s$ = mean fuel segment surface temperature (^{o}C)

and the values of $(T_{FS})_c$ and $(T_{FS})_s$ are obtained by the following semi emperical relationships (W.J. Green - A method of calculating fuel temperatures and its application to a CANDU 600 MWe design. AAEC unpublished report).

$$(T_{FS})_s = \frac{-N + \sqrt{N^2 + 4 \ YM}}{2M}$$

and

$N = h(I * J + K) - L * I \ (A/20.3)$

$Y = h * q + I * J * L + K * L$

$M = h * I\ (A/20.3)$

and

$I = 9.4 \times 10^{-5}$

$J = (PW + A * R)$

$K = 0.94 - 0.008\ F$

$L = q + h * T_B$

where

h = heat transfer coefficient involving heat transfer from the bulk coolant temperature to the inside surface of the cladding ($W/cm^2\ ^{o}C$)

P_w = mean coolant pressure in psi

R = linear rating in W/cm

$= R_{FS} * W/N_p * F$

R_{FS} = average fuel segment fission rating (W/g)

W = weight of fuel segment

N_p = number of fuel pins in assembly

F = fraction of fission heat deposited in fuel

q = surface heat flux

$= R * 4/D*$ enrichment factor $(^{235}U/^{235}U\ \text{natural})$

D = outside diameter of fuel pin (cm)

T_B = bulk coolant temperature

A = constant depending upon cold radial gap size = 85-7874 G

G = radial gap (cm)

F = percentage of fission gases present in the internal gas mixture

$= 100/(1 + 3.548 \times 10^{10}/(B * R^{4.65}\ (1-2.64 * 10^{-9}\ R^{4.65})))$

where

B = burnup in MWd/T

and

$(T_{FS})_c = (R + (T_{FS})_s/20.3)\ Y + Z$

where

Y and Z have the following values depending on the value of $(R + (T_{FS})_s/20.3)$

if $55 < (R + (T_{FS})_s/20.3) < 85$ $\quad Y = 39.5 \quad Z = 820$

if $35 < (R + (T_{FS})_s/20.3) < 55$ $\quad Y = 32.5 \quad Z = 437.5$

if $0 < (R + (T_{FS})_s/20.3) < 35 \quad Y = 20 \quad Z = 0$

The fuel element absorption cross sections are also adjusted according to the estimate of the critical poison concentration by adding the additional cross section (Σ_p) based on

$$\Sigma_p = \delta_p * P$$

where

δ_p = poison addition absorption cross section per unit poison concentration (Latticed cell averaged)

P = critical poison concentration in units consistent with δ_p.

The Σ_{FS}'s of each channel type in a radial zone are averaged together to obtain Σ_{RZE}'s used in the diffusion calculation. An average based on the channel type volume fractions is normally used but this volume average can be combined with approximate flux weighting factors (Φ_c/Φ_s) stored in the cross section library. These Φ_c/Φ_s's are in terms of the average flux in a homogenised lattice cell, c, divided by the average flux in a supercell, s, made of typical channel types. The (Φ_c/Φ_s)'s are derived from

$$\frac{\Phi_c}{\Phi_s} = A(k_{\infty c} - k_{\infty s})^2 + B(k_{\infty c} - k_{\infty s}) + 1 \qquad \ldots(4)$$

where A and B are library coefficients interpolated for supercell radius.

The code determines the k_∞ ratio from the absorption and μ-fission cross sections, derived for the fuel segment and radial zone element (used in the previous diffusion calculation), flux weighted by the Φ_{RZE}'s (no flux weighting factors are used in the first iteration). The supercell size is determined from the number of channel types in that radial zone and the lattice cell radius. The number of fuel channels making up the supercell size can also be specified to overcome radial zones having too many fuel channel types. The $(\Sigma_{FS})_f$'s are multiplied by the flux weighting factors to enable the subsequent channel type power distribution calculation (see Section 5.2.2 (iii)) to include the flux peaking effect. The cross sections of both groups are equally weighted, this being the simplest approximation for heavy water moderated SGHW reactors.

When the Σ_{RZE}'s for the whole active core have been determined, the reflector absorption cross sections are adjusted for the new critical poison concentration using the poison cross sections. If specified, the additional radial zone element ν-fission and absorption cross sections are added to Σ_{RZE}'s before entering the diffusion calculation section.

(ii) Using the Σ_{RZE}'s and reflector cross sections determined in (i) the flux distribution and effective core multiplication constant (k_{eff}) are obtained in DIFSEC. A precalculation of the Liebman acceleration factors is normally only made on the very first entry to DIFSEC (LIEB). Except for an improvement to the Liebman factor calculation no changes have been made in this section since EQUICORE.

(iii) The fluxes obtained in (ii) are normalised in EDISEC using the $(\Sigma_{RZE})_f$'s derived in (i) to give the specified core power. From this the normalised Φ_{RZE}'s are obtained and in turn the power distribution of each channel type can be determined using the $(\Sigma_{FS})_f$'s. This calculation is similar to that used in EQUICORE except that each $(\Sigma_{FS})_f$ is used instead of the average Σ_{RZE}'s.

(iv) If a hydraulic calculation is specified the coolant density distributions of each channel type in all radial zones are calculated in HYDSEC by determining the coolant flow along the feeder, channel and riser sections to satisfy the specified pressure drop between the feeder inlet and riser exit. This involves an iteration process which uses an estimate of flow to calculate the pressure drop and then to readjust the flow to obtain the correct pressure drop. (See HYDRO (Green 1973)). In the process a consistent distribution of void, slip and two phase multipliers is obtained along the channel. Having established the required flow, the average coolant density in each fuel segment can be determined.

In EQUICORE II the hydraulic calculation uses the previous iteration results of pressure drop, void, slip, two phase multiplier distribution and channel coolant flow to speed up convergence.

It is also possible to specify that the calculation need not be done at every EQUICORE II iteration. The following options are available.

1. Every iteration (i.e. cross sections coolant dependent).
2. Every iteration during the first burnup step and then for following burnup steps when the flux has converged (i.e. for cases where the cross sections are not strongly coolant dependent).
3. Only on first sweep of calculation (i.e. to set nominal coolant

densities).

4. Only when flux converged for every burnup step (i.e. dryout margin purposes).

5. Only when flux converged and only for first and last burnup set of burnup cycle.

(v) A new estimate of the poison cross section is determined in CONTRL from the k_{eff} obtained in (ii) based on the rate of change of k_{eff} with poison concentration (a new rate of change is estimated on each iteration).

(vi) Steps (i) through to (v) are repeated in an iterative manner to obtain a converged critical core and a converged power, coolant density and cross section distribution. In most cases each step uses previous iteration results to achieve a rapid calculation.

(vii) When convergence is achieved (TEST), a print out of results required by the user is given during a final sweep of step (iii), (iv) and (i). During this converged sweep, a dryout margin calculation is performed in the hydraulic section (DRYOUT). The print out can take the following form:

(a) Fuel channel type and radial zone power distribution. Level $\boxed{2}$

(b) Fuel channel type and radial zone total power together with peak to average statistics. Level $\boxed{1}$

(c) Core RZ average power distribution. Level $\boxed{2}$

(d) Core RZ average flux distribution. Level $\boxed{3}$

(e) Hydraulic information for each fuel channel type consisting of pressure, enthalpy, quality voidage, two phase multiplier, critical heat flux and critical heat flux rates distributions. Level $\boxed{3}$

(f) Fuel channel type, coolant flow, minimum critical heat flux ratio, riser exit pressure and quality. Also total core coolant flow. Level $\boxed{2}$

(g) Fuel assembly distributions of irradiation, rating, temperature, xenon concentration together with the associated coolant density normalised fluxes. Level $\boxed{3}$

(h) Fuel assembly average information. Level $\boxed{2}$

The user can choose the detail of print out he requires by specifying the level. Print out occurs of those data equal or below the number specified. The code automatically increases the level by one for the first and last convergence step during the burnup cycle.

5.2.3 Increase of core burnup time

When convergence of a burnup step has been achieved control is passed to subroutine MOVE where

(i) The core burnup time is incremented by the specified burnup time step interval.

(ii) An estimate of the new poison concentration is made based on the rate of change of core k_{eff} with time (a new rate of change is calculated at the completion of each burnup step). If the estimate of poison concentration is less than the minimum specified the burnup time interval is suitably reduced to give the estimated time corresponding to the end of life of the core loading.

(iii) A switch is set so that the irradiation states of each fuel segment will be advanced in CROSEC before determining the new Σ_{FS} and Σ_{RZE}'s . The irradiation will be advanced in CROSEC according to the present $(\Sigma_{FS})_f$'s, Φ_{RZE}'s and the burnup step time.

(iv) Control is returned to CALC where the process outlined in 5.2.2 is repeated to obtain the new core power distribution. If the core burnup time cannot be incremented by a significant amount, or the specified number of burnup time steps has been achieved, control is returned to DATAIN to read further input data.

5.2.4 Reading and interpreting refuelling instructions

On return to DATAIN the code expects either some form of refuelling instruction or an alternative form of calculation (perturbation or supercell type, see Sections 5.4 and 5.5). However the code will, before reading in any data, first dump all relevant information onto disc for possible later restarting at this point. If the read-in data is refuelling information, control is passed to MANDAT where

(i) Fuel assembly moves are interpreted by transferring the fuel segment irradiations and assembly type of the specified assembly to its new specified location.

(ii) Fuel assembly loadings are interpreted by simply resetting the fuel segment irradiations to zero and resetting the fuel assembly type.

(iii) Any data to reset burnup or print instructions is read in.

(iv) The poison concentration is reset to its initial core start value or to a new read-in value.

(v) A summary is printed out of the rejected fuel assembly burnup and fissile concentration distribution. The fissile concentration is obtained from the cross section library and is based on the fuel segment irradiation.

(vi) Control is returned to CALC where the processes outlined in 5.2.2 and 5.2.3 are to be repeated to obtain the next end of core lifetime.

5.3 Equilibrium Burnup Calculation

The equilibrium burnup calculation can be considered to be made up of two parts which are similar to the first two parts of the time dependent burnup calculation.

5.3.1 Reading and preparation of data

In this section the same steps are performed as in 5.2.1 to set up the geometry, hydraulic and cross section data. However the fuel management data in the equilibrium model is set up slightly differently than in (ii) of 5.2.1, in that the paths taken by all the fuel assemblies are converted into a string of fuel assembly core locations. Also the poison concentration, if any, is fixed and an estimate of the average residence time within a core assembly location is calculated from the specified fuel management average core fuel rating and guess of core average burnup. A check is also made that the fuel management is consistent with the core model. The guess of $(\Sigma_{FS})_f$'s are also based on the average of the expected burnup.

5.3.2 Calculation of the equilibrium residence time and critical power distribution

Except for a few minor changes the equilibrium calculation is the same as that in EQUICORE. A similar process to that outlined in 5.2.2 is used to obtain a consistent equilibrium cross section-flux-coolant density distribution for a critical core. However in this case the residence time is varied instead of the poison concentration and the determination of the equilibrium averaged Σ_{FS}'s is more complicated. To determine the Σ_{FS}'s for the time the fuel remains in a location, the following steps are performed in subroutine BURN.

(i) The irradiation of each fuel segment within the first specified assembly core position is set to zero.

(ii) The cross sections are obtained from the library as outlined in (i) of 5.2.2. In this case the temperature used to correct the cross sections is based on the average fuel segment rating and burnup while in that position, and is obtained from the fuel channel type power distribution calculation. The xenon concentration correction is similarly based on the previous iteration average rating and is only made after the average Σ_{FS} has been determined.

(iii) The irradiation of each fuel axial segment is advanced for a fraction of the estimated critical residence time. The burnup rate is based on the $(\Sigma_{FS})_f$ and Σ_{RZE}. On the first iteration control is first passed to EDISEC to convert the guessed flux distribution into Φ_{RZE}'s (see Section 5.2.2 (ii)).

(iv) Steps (ii) and (iii) are repeated for the full residence time and the average cross section of each fuel axial segment is determined.

(v) The fuel assembly is moved to its next core location and steps (ii) through to (iv) are repeated, increasing the fuel irradiation until the assembly has completed its specified path through the core and is rejected.

(vi) Process (i) through to (v) is repeated for each fuel assembly path specified. During this calculation the average Σ_{FS}'s of each channel type in a radial zone are combined to obtain the Σ_{RZE}'s in the same manner as in Section 5.2.2 (i).

Although the channel power and hydraulic calculations are exactly the same as in Section 5.2.2 they give in this case results of a time averaged fuel channel. A similar, but improved technique as described in EQUICORE is used to converge the residence time and Φ_{RZE}'s. Tighter control on the fluxes is necessary when calculating for SGHW reactors having large burnups and non-axisymetric reflector geometries.

After convergence an extra sweep is made to print results (see Section 5.2.2 (vii)) before control is returned to DATAIN where a summary of the burnup and fissile concentration is given. Data for a modified equilibrium calculation or a time dependent calculation is then read in.

5.4 Supercell Calculation

This calculation can be requested at any stage in a time dependent burnup calculation or after convergence of an equilibrium calculation. Control would have been returned to DATAIN and a dump on disc of information existing at that time will have been made. The procedure is again similar to the first two parts of the time dependent calculation.

5.4.1 Reading and setting up data for supercell model

(i) The new supercell geometry (2 radial zones, volumes, etc.) is automatically set up in SUPDAT from the existing data of the fuel channel type and radial zone requested. The inner radial zone radius is made equal to the lattice cell radius and the outer supercell radius is based on the number of channel fractions in the specified channel type. The mesh spacing is set according to the number of mesh intervals specified for the two zones. The axial geometry is left unaltered from the previous main calculation.

(ii) The specified fuel channel type is set up as the first radial zone and the remaining channel types from the specified radial zone are set up in the surrounding radial zone. For a supercell calculation following an equilibrium calculation no outer channels are set and the outer radial zone cross sections are set permanently to those used in the equilibrium calculation for that radial zone. Also the irradiation distribution of the assemblies in the channel are set according to the irradiation obtained by the equilibrium model assembly on first reaching that location.

(iii) For the supercell following a time dependent calculation the total power is set to give the same average power as existed in the radial zone. For the supercell following an equilibrium calculation the power is set such that the flux will be normalised to give the same power at the edge of the supercell as that of the average power in that radial zone.

5.4.2 Calculation of supercell distribution for consistent cross sections-flux-coolant density distribution

A procedure similar to that in Section 5.2.2 is used to obtain a consistent cross section-flux-coolant density distribution, except that the poison cross section is not varied to obtain a critical supercell. Also in the case of an equilibrium type supercell, only the new conditions of the centre fuel channel type are calculated and the flux in the supercell is normalised to the set supercell edge power rather than to the total supercell power.

On return to DATAIN from a supercell calculation the code reads in the previous dump of information to restore data conditions back to the way they were before the supercell calculation started.

5.5 Perturbation Calculation

As with the supercell calculation this option can be requested at any time when control is returned to DATAIN. When a perturbation calculation is requested (key word PERT) control is passed to the special purpose PERT subroutine instead of CALC. The existing PERT subroutine (see Appendix C for typical PERT coding) will have been either modified or rewritten by the user and the typical sequence of events which would take place is

(i) Some changes to the core data are made, e.g. reactor power is altered to determine the changes on fuel temperature, xenon concentration and coolant distribution.

(ii) Depending on the user's wishes, processes outlined in Section 5.2.2 can be requested in terms of simple call statements to bring in the various sections of the code, to enable the new perturbed power distribution to be calculated. The cross sections calculated in CROSEC will be based on the existing irradiation but all other conditions (i.e. coolant density, fuel temperature and equilibrium xenon concentration) will depend on the results of the new core power distribution. In the case of an equilibrium calculation th average Σ_{FS}'s are determined as in Section 5.3.2 except that the fuel segment irradiation steps are not recalculated but are taken from the results of the previous equilibrium burnup calculation.

(iii) Alternatively the user can access a built-in subroutine (PXEN) which will calculate the time and flux dependent variation of xenon concentration and at each time interval go through the process outlined in Section 5.2.2. The user can force changes to the additional radial zone element absorption and μ-fission cross sections (see Section 5.2.2 (i)) to maintain criticality or prevent excessive power peaks, via the variable BOOST. (See PERT coding Appendix C.)

When the PXEN subroutine is used the equilibrium full power xenon concentration (XE_p) and library concentration (XE_r) are first determined in each fuel segment position for time t = 0 using equation (1) and (2). The equilibrium iodine concentration (IO_p) is obtained from:

$$IO_p = \gamma_I * \Phi/\lambda_I$$

where

γ_I = iodine fractional yield = 0.003

λ_I = iodine decay constant = 2.9×10^{-5} .

The iodine and xenon in concentrations after a small time interval (DT) is calculated from:

$$IO_p = IO_p + [-IO_p * \lambda_I + \gamma_I * \Phi]\ DT$$

$$XE_p = XE_p + [IO_p * \lambda_I + \gamma_{Xe} * \Phi - XE_p\ (\lambda_{Xe} + \delta_{Xe} * (\Phi_{RZE})_2]\ DT$$

$$\Phi = (\Phi_{RZE})_1 * (\Sigma_{FS})_{f1} + (\Phi_{RZE})_2 * (\Sigma_{FS})_{f2}$$

where

γ_{Xe} = xenon fractional yield

$= \gamma_{XI} - \gamma_I$.

The calculation is performed for INT intervals where

INT = XADT/XSTIM

XADT = time advanced between flux calculations (minutes)

XSTIM = specified time between xenon calculational intervals (minutes).

While using the perturbed (XE_p)'s the (XE_r) is held fixed. The PXEN subroutine is automatically brought in if IXEN is set to 1 and the (XE_r)'s are calculated when XTIM is set to 0. The time dependent xenon concentration is advanced when XADT is set positive. To allow for xenon concentration change during shutdown or low power XPOW can be set to the fraction of the full power required. To alter some radial zone element xenon concentration to check on instability XCTOP, XCBOT or XSRAD can be set to some fraction of the equilibrium value at time t = 0. (See PERT coding, Appendix C.)

As with the supercell option, on return to DATAIN the previous data conditions are automatically re-established.

A computing facility MODIT (E.W. Hesse - MODIT. A computing facility for simplifying the task of developing large FORTRAN computer facilities. AAEC unpublished report) is available which allows the user to alter and extend PERT, PXEN and PCONT subroutines.

6. SPECIFICATION OF INPUT DATA

6.1 General

The cross section and fuel fissile concentration library is separated from the problem input data and is read in from a separate data set (FORTRAN unit 7) normally residing on disc. This library data is described in (6.7).

The data for a problem are supplied in free format following a key word indicating the type of data. Blocks of data are separated only by key words and not by cards. Any special characters (e.g. (,),, etc.) within the data are ignored and treated as blanks, and thus can be inserted for ease of data checking. Numbers are terminated by a blank or letter and key words are terminated by blanks or numbers. Any zeros at the end of a data block need not be specified. The blocks of data for a problem are grouped into 5 types:

(1) Title and cross section library specification data.

(2) Diffusion geometry layout data.

(3) Fuel management data.

(4) Hydraulic data.

(5) General burnup and code running data.

Data types 1 to 4 must be specified in the above order. Data blocks within each type can be specified in any order except for the block of data indicating storage requirements which must preceed each data type. Data belonging to type 5 need not be grouped together and can be placed between data types. Data type 4 need not be specified if no hydraulic calculation is required. No calculation takes place until the key word ENTER is read and it must therefore be supplied after each refuelling. If a case is to start from a dump of a previous case the key word REST must follow the title card. In this case only blocks of data which are to be altered need be specified.

During a continuation of a burnup calculation (i.e. after refuelling) extra data of type 3 will be specified. Also most data belonging to data type 5 may be altered during the calculation.

A listing of input data for thee typical EQUICORE II calculations are given in Appendix A. Appendix A1 is for a CANDU type reactor where the calculation has been arranged so that only half the symetrical core need be calculated. This is achieved by tracing a fuel assembly half way along one channel and then inverting the fuel to trace the fuel back in its partner channel. Appendix A2 is for a very detailed calculation of a typical SGHW reactor where an equilibrium calculation is used to obtain a guess of the core conditions once an equilibrium cycle is reached, before a time dependent calculation is started. This calculation allows for variation of the number of fuel channel types near enrichment and reflector boundaries. Selected extracts of the output of this case is given in Appendix B to demonstrate the information available from the code. Appendix A3 is for a simple calculation of an SGHW reactor requiring little computer time. It is used for scoping type calculations.

6.2 Title and Cross Section Library Specification

Data for each new case must be preceded by a title card having an * in column 1. The code will hunt for the next * card if a case fails for reasons of input data errors, time exceeded or failed conveyance.

The first key following the title card of a new case must be:

LIB, Name of cross section library to be used (4 characters, see library data specification (6.7)), whether any of the following library interpolation and correction factors should not be used (not used $\neq$ 0 use 0): for coolant density (set equal to coolant density number in library), temperature, xenon and flux weighting factors.

Note:

(1) The code automatically sets indicators off if data is not

included in library.

(2) If a hydraulic calculation is not requested and no particular coolant density is requested the code will use the library mid range density.

In the case of a restart of a previously run job the following keyword, REST must follow the title card and the LIB data need not be specified.

REST, Number of output dumps to be skipped over.

Note: *The last dump (on FORTRAN unit 8) will be used if the number of dumps specified exceeds that on the dump file.*

6.3 Diffusion Geometry Layout Data

This data is concerned with setting up data required by the diffusion calculation section. Most of the data is similar to that used in EQUICORE.

The first key word of this data must be:

DIF, Active core height (cm), number of axial slices (NCHANX), radial (NRAD), and axial (NAX) mesh intervals, reflectors (NREF), whether additional radial zone element cross sections are added (1 = yes, -1 = yes but only to be used in PERT calculation, 0 = no), followed by a series of 3 numbers A, I, and B allocating width of radial zones such that A fuel channels are distributed over I radial zones in manner indicated by B

where

B = 0, the channels will be equally divided over the I radial zones, or

B = positive, the radial zones will be set up such that the inner radial zone will be B times the width of the outer.

Note:

(1) NCHANX must be divisible by NPOS in the MAN data (see Section 6.3).

(2) The sum of A's is equal the number of fuel channels in the true core and determines the active core radius.

(3) The sum of I's is equal the number of radial zones in calculation (NCHAN) and must be less than 10.

This data is followed by any of the following keywords.

RM, Radial mesh width specification from centre of core,

i.e. RM O FILL 17 (17) 19 (28) 20

specifies that the first 17 mesh intervals are to be allocated to the active core followed by 2 intervals of 17 cm followed by 1 of 28 cm to the radial reflectors.

<u>*Note:*</u>

(1) The core mesh intervals will be allocated according to the following CA data.

(2) The last number must not be greater than specified in the DIF data (NRAD).

<u>ZM</u>, Axial mesh width specification from bottom of reactor. This data is similar to RM,

i.e. ZM 0 (15) 2 FILL 12 (10) 13 (15) 14

<u>*Note:*</u>

(1) The core mesh intervals will be allocated according to the following CCOR data.

(2) The last number must not be greater than specified in the DIF data (NAX).

(3) Reflectors need not be specified.

<u>CCOR</u>, Arrangement of axial slices over the active core starting from bottom of the reactor,

i.e. CCOR 0 R2 2 Z1 4 Z2 5 Z3 6 Z4 7 Z5 8 Z6 9 Z7 10 Z8 12 R3 14

specifies that the first two axial mesh intervals are to be filled with data of reflector data number 2, the next two intervals with data of core region 1, the next one by core region 2 etc. If the number of mesh intervals for the next and subsequent slices are the same as the preceding slice the word FILL can be used,

i.e. CCOR 0 R2 2 Z1 4 Z2 5 FILL 10 Z8 12 R3 14

<u>*Note:*</u>

(1) The data must be consistent with the ZM data.

(2) The active core slices must be equal to that specified in the DIF data. (NCHANX).

<u>C</u>, Radial reflector number followed by axial layout of radial reflectors starting from bottom of reactor,

i.e. C 2 0 R4 4 R1 10 R4 14

specifies that the second radial reflector arrangement has 4 mesh intervals of reflector material 4, followed by 6 intervals of reflector material 1, etc. C data for each radial reflector specified in the following CA data must be given.

<u>*Note:*</u> *The data must be constant with the ZM data.*

<u>CA</u>, Arrangement of radial zone and radial reflectors. This data is similar to CCOR,

i.e. CA 0 Z1 4 Z2 7 Z3 9 FILL 17 C1 19 C2 20 specifies that the first 4 radial mesh intervals are to be allocated to the first radial zone, followed by 3 to the next radial zone, etc. The first 17 mesh intervals will be allocated as per the CCOR date, the next 2 as per radial reflector number 1, etc. The use of FILL works, in the same way as in CCOR data and allocates in this example 2 mesh intervals for radial zones 4 to 17.

Note:

(1) The data must be consistent with the RM card.

(2) The number of radial zones must be equal to that specified in the DIF data (NCHAN).

R, Reflector data number, cross sections D_1, D_2, Σ_{a1}, Σ_{a2}, Σ_{rem}, $\mu\Sigma_{f1}$, $\mu\Sigma_{f2}$, Σ_{P1}, Σ_{P2}.

where

D = diffusion coefficient

Σ_a = absorption cross section

Σ_{rem} = removal cross section

$\mu\Sigma_f$ = μ fission cross section

Σ_p = poison addition absorption cross section per unit poison concentration.

1 = group 1

2 = group 2

Note:

(1) R data must be given for each reflector material specified in the C or CCOR data

(2) Reflector data number must not be greater than that specified in the DIF data (NREF).

XTRA, Σ_{a1}, Σ_{a2}, $\mu\Sigma_{f1}$, $\mu\Sigma_{f2}$, radial zone I to J, axial slice k to L. These additional cross sections will be added to radial zone elements specified by the I, J, K and L locations. More than one lot of XTRA data can be given.

Note: This data can only be given if specified in the DIF data.

6.4 Fuel Management Data

This data is concerned with the setting up of the layout of fuel channel types and fuel assembly locations, together with the interpreting of the fuel management instructions. The first key word in this section of data in a new case must be:

MAN, Maximum number of fuel channel types in any radial zone (NMAX), number of fuel assembly locations in a channel (NPOS), followed by data required only for an equilibrium burnup calculation consisting of the number of fuel assemblies to be traced through the core (number of assemblies specified in ASS data blocks (see Section 6.4), the maximum number of core locations any fuel assembly visits on its way through the core (see Section 3.4.2), whether a perturbation calculation will follow this equilibrium burnup calculation (yes = 1, no = 0) (see Section 5.5)).

Note: The number of core axial slices (NCHANX) specified in DIF data, must be equally divided by NPOS.

This data is followed by any of the following keywords.

CHAN, Radial zone number, fraction of control poison to be applied to radial zone (PRZF), number of fuel channel types (N) in radial zone, followed by N channel type volume fractions (CTV) (see Section 3.1), N ratios of fuel residence time to control residence time in each channel type (CTRT) (see Section 3.4.2) (required only for equilibrium burnup calculations), number of fuel channels making up supercell (NS).

Note:

(1) If CHAN data for a radial zone is not specified, PRZF = 1.0, N = 1, CTV = 1.0 and CTRT = 1.0 is assumed.

(2) If CTV are not specified or set equal to zero, CTV's are set to 1.0/N. If partly specified, the remaining CTV's are set equally to bring the total to 1.0.

(3) If CTRT's are not specified, CRT's are set to 1.0. If they are only partly specified, the remaining will take the value of the last CTRT specified.

(4) If NS is not specified NS is set to N. If NS = 1 no supercell or flux weighting is possible.

The following key words are concerned with fuel loading and movement in the core. These data would be supplied at various stages of the burnup calculation. Fuel assemblies are loaded into or moved to assembly locations designated by a 3 or 4 digit figure, where the first digit signifies the radial zone, the middle 1 or 2 digits signify the fuel channel type and the last digit the fuel assembly location, numbered from the bottom of the core. Fuel can be moved from any location of the core to any other. If the location number is set negative (i.e. -113) the fuel assembly will be first inverted before being placed in that position. Although the code checks that all locations in the core are loaded and that all locations specified exist, no other check is made on the validity of the management. Therefore the user should check that the channel type volume fractions and equilibrium residence time ratios do not invalidate continuity of material.

LOAD, Library fuel assembly type (see Section 3.4), followed by a string of assembly location numbers indicating where this fuel is to be loaded. Used only for time dependent burnup calculations,

i.e. LOAD 2 121 122 1121 1122

specifies that fuel of library type 2 should be loaded in axial locations 1 and 2 of channel type 2 and of channel type 12 of radial zone 1.

A series of LOAD data should be specified to cover the entire core. If the instruction of some of the following radial zones are the same, the use of the key word REP can be used.

REP, Repeat instruction for next N radial zones. Used with LOAD, MOVE ASS instructions and also with FEED and RISE in hydraulic section

i.e. LOAD 1 221 222 2121 2122 REP 2

will specify loadings to radial zones 2, 3 and 4.

Note:

(1) The use of the word LOAD signals that a time dependent burnup calculation is required.

(2) LOAD 0 specifies the loading of fuel based on results from a previous equilibrium burnup calculation in a time dependent calculation. (See later) following ASS key word.

MOVE, Followed by pairs of assembly location numbers indicating that the fuel assemblies should be moved from the first location to the next. Used only for time dependent burnup calculations,

i.e. MOVE 112 to 113, 111 to 112

specifies that fuel assemblies in axial location 1 and 2 should be moved up 1 location.

A series of MOVE data is normally required to transfer all fuel. The REP key word can be used for radial zones having the same instructions,

i.e. MOVE 112 to 121, 112 to -122, 111 to 112 REP 4

LOAD 3 111 REP 4

specifies that the movement and loading of radial zones 1 to 5. The fuel is inverted on its way down the channel type 2.

Note:

(1) The fuel assembly existing in a location, to which some other

fuel is being transferred, is lost and therefore must be moved first, if it is required. The 'lost' fuel is considered to be unloaded from the core.

(2) The word TO need not be specified.

ASS, Library fuel assembly type, followed by a string of assembly location numbers indicating the path the assembly takes during its burnup in the core. ASS is used only for equilibrium burnup calculations,

i.e. ASS 3 111, 112, -122, 121

specifies that assembly of fuel of library type 3 should be loaded in axial location 1 of channel type 1 in radial zone 1 then moved to location 2, then inverted and moved to location 2 of channel type 2, then moved to location 1 and then finally rejected.

A series of ASS data should be specified to cover the entire core. If the instructions of some of the following radial zones are the same, the use of the key word REP (see back) can be used,

i.e. ASS 3 111, 112, -112, 121 REP 2

ASS 2 411, 412, -422, 421 REP 3

will specify loadings to 7 radial zones.

Note: *The use of the word ASS signals that an equilibrium burnup calculation is required.*

LOAD 0, Following an equilibrium burnup calculation the use of LOAD 0 will automatically convert the equilibrium fuel management to a time dependent burnup calculation,

i.e. ASS 1 111 121 131 141 151 161 171 181 191

followed after the equilibrium calculation by

LOAD 0

will load fuel of type 1 having:

(i) zero irradiation into location 111

(ii) irradiation achieved during the equilibrium calculation in 111 into location 121

(iii) irradiation achieved in 121 into location 131 etc.

6.5 Hydraulic Data

This data is concerned with supplying the necessary remaining hydraulic model (i.e. feeder and riser layout), including the hydraulic constants and calculation requirements.

The first key word in this section must be:

HYD, Number of sections in feeder (NF) and in riser (NR), type of calculation (NT) (0 = fixed inlet feeder pressure, 1# = constant riser exit quality), indicator when calculation should be done (see Section 5.2.2 (iv)) i.e. 1 every iteration, 4 only when converged, etc.), whether light (0) or heavy (1)# water coolant, options for slip correlation (1 = homogeneous, 2 = Bankoff-Armand, 3 = Bankoff-Jones), for grid two phase multiplier (1 = homogeneous, 2 = equal to friction coefficient), for dryout correlation (1 = CISE, 2 = Macbeth, 3 = Barnett, 4 = Janssen, 5 = W3, 6 = WAPD-92), for flow two phase multiplier (1 = homogeneous, 2 = Martinelli Nelson, 3 = Levy, 5 = 1.0 + quality* (quality * XFMN + XFMN2), XFMN for option 5.

#<u>Note</u>: *Constant riser exit quality and heavy steam water properties are yet to be incorporated.*

This is followed by any of the following keywords.

<u>CIRC</u>, Fraction of fission heat deposited in coolant, riser exit pressure (psi), feeder inlet enthalpy (Btu/lb), if NT = 1 feeder inlet pressure (psi) or NT = 2 riser exit quality (3), whether fuel channels vertical (0) or horizontal (1), active core channel hydraulic equivalent diameter (ft), flow area (ft^2), total discontinuity pressure loss coefficient smeared over the active length of the channel, dryout correlation equivalent heated diameter, guess of an average mass coolant flow (lbs/hr).

<u>FEED</u>, Radial zone number, feeder section number (numbered from pump), feeder section height (ft), length (ft), equivalent hydraulic diameter (ft), flow area (ft^2), discontinuity pressure loss coefficient.

A series of FEED data should be specified for each section and radial zone. If the following radial zones are similar the key word REP (see Section 6.4) can be used,

i.e. FEED 1,1 17.0, 20.0, 0.21, 0.33, 3.3 REP 3
FEED 1,2 1.0, 8.0, 0.4, 0.13, 0.6
FEED 2,2 0., 6.5, 0.4, 0.13, 0.6 REP 2

will specify that data for feeder section will be set for radial zones 1 to 4. Similarly for section 2 in radial zones 2 to 4, feeder section 2 in radial zone 1 being different.

<u>Note</u>: *NF x NCHAN feeder sections must be specified.*

<u>RISE</u>, As per FEED data (sections number from active core)

<u>Note</u>: *NR x NCHAN riser sections must be specified.*

6.6 General Burnup and Running Data

This data consists of general information for controlling the running of the calculation. Most of this data may be changed during a burnup calculation.

BURN, Core thermal power (MW), desired core k_{eff}, guess of core average burnup (MWd/tU), lower limit of poison concentration (library units), initial guess of poison concentration (library units), time dependent burnup time steps (days).

Note: For an equilibrium burnup calculation the guess of poison concentration is taken as a fixed amount of core poison.

RUN, Maximum allowable code executing time (minutes) between ENTER steps, maximum number of burnup steps to be done before returning to read further instructions, degree of print out (see Section 5.2.2 (vii)).

FLUX, Last radial mesh point for the flattened region to be used in flux guess calculation, flux extrapolation distances (cm from reflector edge) radial, bottom and top, diffusion calculation dominance ratio, Liebman acceleration factors group 1, group 2.

Note:

(1) The code will use the previous calculated flux if the flux guess radius is set to 0.

(2) If Liebman factors are not specified the code will calculate its own.

(3) For dominance ratio see EQUICORE 5.3.5.

CONS, Guess of rate of change of core k_{eff} with change in irradiation (k/MWd/tU), guess of rate of change of core k_{eff} with change in poison concentration (units of library), fuel temperature calculation data (see Section 5.2.2 (i) (d)) consisting of: fraction of heat of fission being deposited in fuel, fuel pin diameter (cm), number of pins in assembly, cold fuel/can gap (cm), enrichment factor, system pressure (psi), heat transfer coefficient (W/cm^2 $^{\circ}$C) average coolant temperature ($^{\circ}$C). Guess of average coolant density (g / cm^3) at core inlet, at core outlet (only required for coolant dependent cross section calculations).

Note:

(1) If temperature data not specified temperature corrections are not made to cross sections.

(2) Coolant density will be linearly interpolated along fuel channel.

ENTER, To start any calculation going.

ACC, Accuracy specification. This data need not be specified unless values are to be altered. Built in values are indicated in brackets.

Accuracy in core k_{eff} (K = 0.0005), in power distribution (fraction = 0.005), in hydraulic pressure calculation (fraction = 0.01), in source in diffusion calculation at start (0.001), at finish (0.000075), maximum number of diffusion calculation iterations (40), maximum number of flux iterations per burnup step or equilibrium calculation (15), equilibrium flux combination factor modifier (1.0).

PERT, Indicator of type of calculation, followed by a string of numbers to be used in the special purpose perturbation calculation.

Note:

(1) This data is only required if a perturbation calculation is required and the data will be dependent on the users wishes. (See Appendix C) for testing of existing PERT routines which includes a dictionary of important variables likely to be altered.

(2) A PERT calculation must follow a normal burnup calculation.

SUPER, Channel type number, radial zone number, number of mesh intervals in centre fuel channel cell, number of mesh intervals in rest of supercell,centre channel volume fraction of supercell (only required for supercell following equilibrium burnup calculation supercell).

6.7 Cross Section Library Data Specification

This data is normally read from disc (using FORTRAN unit number 7) and consists of two group cross sections as a function of irradiation and coolant density, together with correction factors for temperature and xenon. In addition the library can include fuel fissile concentration as a function of irradiation and flux weighting factors. All this data is in fixed format. A library preparation code exists to help produce an EQUICORE library from the METHUSELAH (Alpiar 1964) code.

Item 1 (1A4,19A4) Library identification name, other reference information.

Item 2 (6I6) Number of fuel assembly types in library (NMAT), maximum number of coolant densities in any material, maximum number of irradiations in any coolant density set, whether fuel temperature correction data available (LTEMP, yes = 1, no = 0), whether fissile concentration data available (LCONC) whether flux weighting factors available (LFLUX).

Item 3 (4E 12.5) Lattice cell cross sectional area (cm^2), weight of uranium metal (kg/cm length), effective xenon yield at equilibrium.

Items 4 to 16 required for NMAT fuel assemblies in turn.

Item 4 (I6,6X,3E 12.5) Number of coolant densities (NCOOL), datum fuel temperature (^{o}C), datum fuel rating (W/g), cell averaged microscopic xenon thermal absorption cross section.

Items 5 to 14 required for NCOOL coolant density in turn.

Item 5 (I6,6X,3E 12.5) Number of irradiations (NIRAD), coolant density (g/cm), cell average poison macroscopic absorption cross section fast, thermal

Items 6 and 7 for NIRAD irradiations.

Item 6 (6E 12.5) irradiation (MWd/tU) D_1, Σ_{a1}, Σ_{rem}, $\mu\Sigma_{f1}$, Σ_{f1}

Item 7 (4E 12.5) D_2, Σ_{a2}, $\mu\Sigma_{f2}$, Σ_{f2}

where

D = cell averaged diffusion coefficient

Σ_a = cell averaged absorption cross section

Σ_{rem} = cell averaged removal cross section

$\mu\Sigma_f$ = cell averaged μ fission cross section

Σ_f = cell averaged fission cross section

and suffix

1 = fast neutron group

2 = thermal neutron group

Item 8 required only if LCONC = 1

Item 8 (3E,12.5) Fissile concentration (g/kg initial U) ^{235}U, total plutonium, fissile plutonium.

Items 9 to 14 required only if LTEMP = 1

Item 9 (3E 13.6) Irradiation start, middle, end (MWd/tU)

Item 10 (6E 13.6) As, Am, Ae, Bs, Bm, Be for Σ_{a1}

Item 11 (6E 13.6) As, Am, Ae, Bs, Bm, Be for Σ_{a2}

Item 12 (6E 13.6) As, Am, Ae, Bs, Bm, Be for Σ_{rem}

Item 13 (6E,13.6) As, Am, Ae, Bs, Bm, Be for $\mu\Sigma_{f1}$

Item 14 (6E 13.6) As, Am, Ae, Bs, Bm, Be for $\mu\Sigma_{f2}$

where A and B are coefficients of Equation (3) and s, m, e indicates values for irradiation near start, middle and end of life of assembly.

Note: If A set to -1. for any cross section the temperature correction for that cross section is bypassed.

Items 15 and 16 required only if LFLUX = 1

Item 15 (3E 12.5) Supercell radius small, average, large (CM).

Item 16 (6E 12.5) As, Aa, Aℓ, Bs, Ba, Bℓ for (Φ_C/Φ_A)
where A and B are coefficients of Equation (4) and s, a, ℓ, indicates values for a small, average and large supercell.

7. COMMONLY USED NOMENCLATURE

Φ_{RZE}	radial zone element neutron flux
$(\Phi_{RZE})_1$	radial zone element neutron fast flux
$(\Phi_{RZE})_2$	radial zone element neutron thermal flux
Φ_{RZE}'s	all radial zone element neutron fluxes in core
Σ_{FS}	fuel segment macroscopic cross section
$(\Sigma_{FS})_f$	fuel segment macroscopic fission cross section
Σ_{FS}'s	all fuel segment macroscopic cross section used in the core
Σ_{RZE}	radial zone element macroscopic cross section
$(\Sigma_{RZE})_f$	radial zone element macroscopic fission cross section
Σ_{RZE}'s	all radial zone element macroscopic cross section used in the core
γ_I	iodine fractional yield
γ_{Xe}	xenon fractional yield
γ_{XI}	effective equilibrium xenon fractional yield = $(\gamma_I + \gamma_{Xe})$
λ_I	iodine decay constant
λ_{Xe}	xenon decay constant
Xe	xenon thermal cross section

8. ACKNOWLEDGEMENTS

The programming assistance of Mrs. J. Faulkner is gratefully acknowledged.

9. REFERENCES

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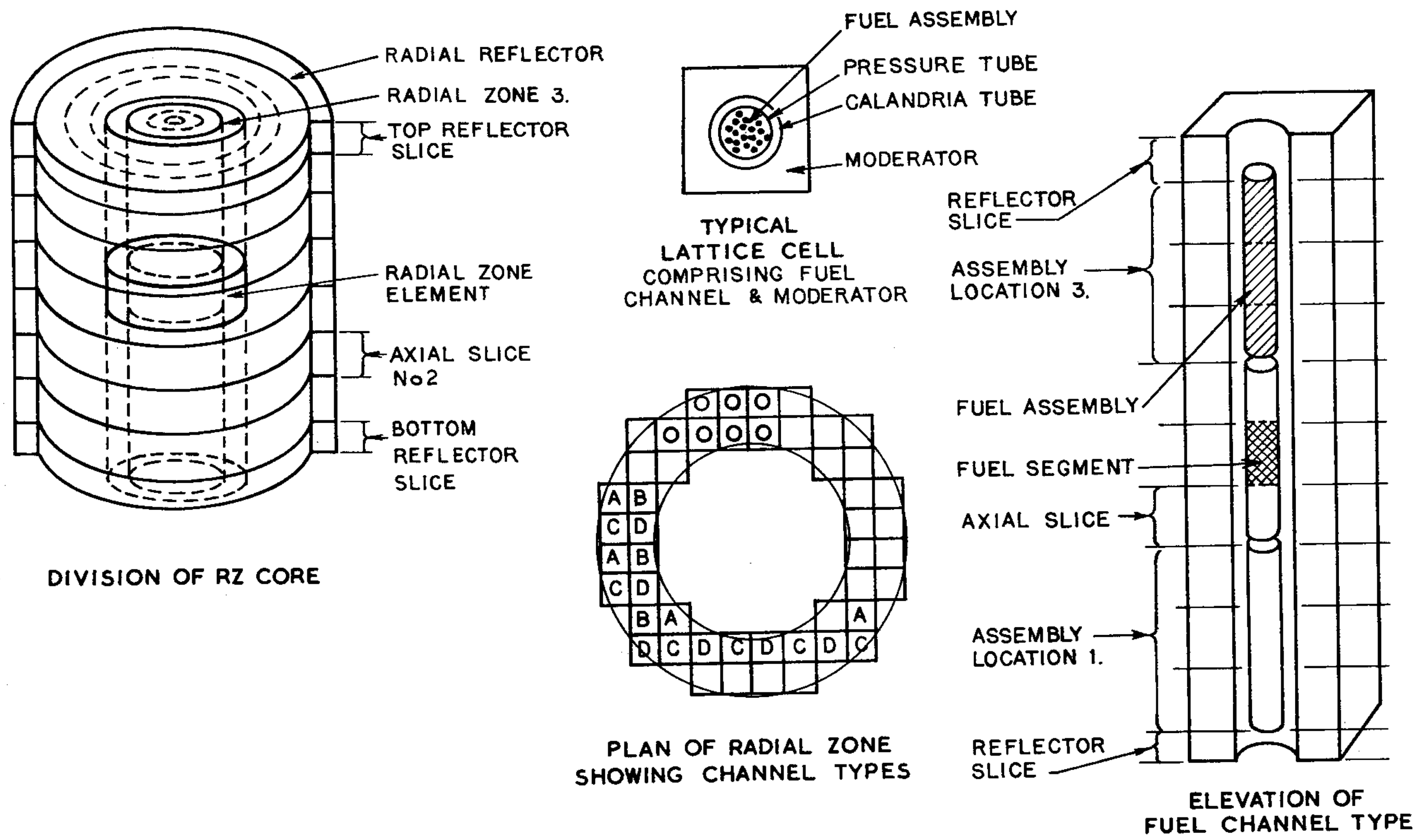

FIGURE 1. EQUICORE CORE MODEL

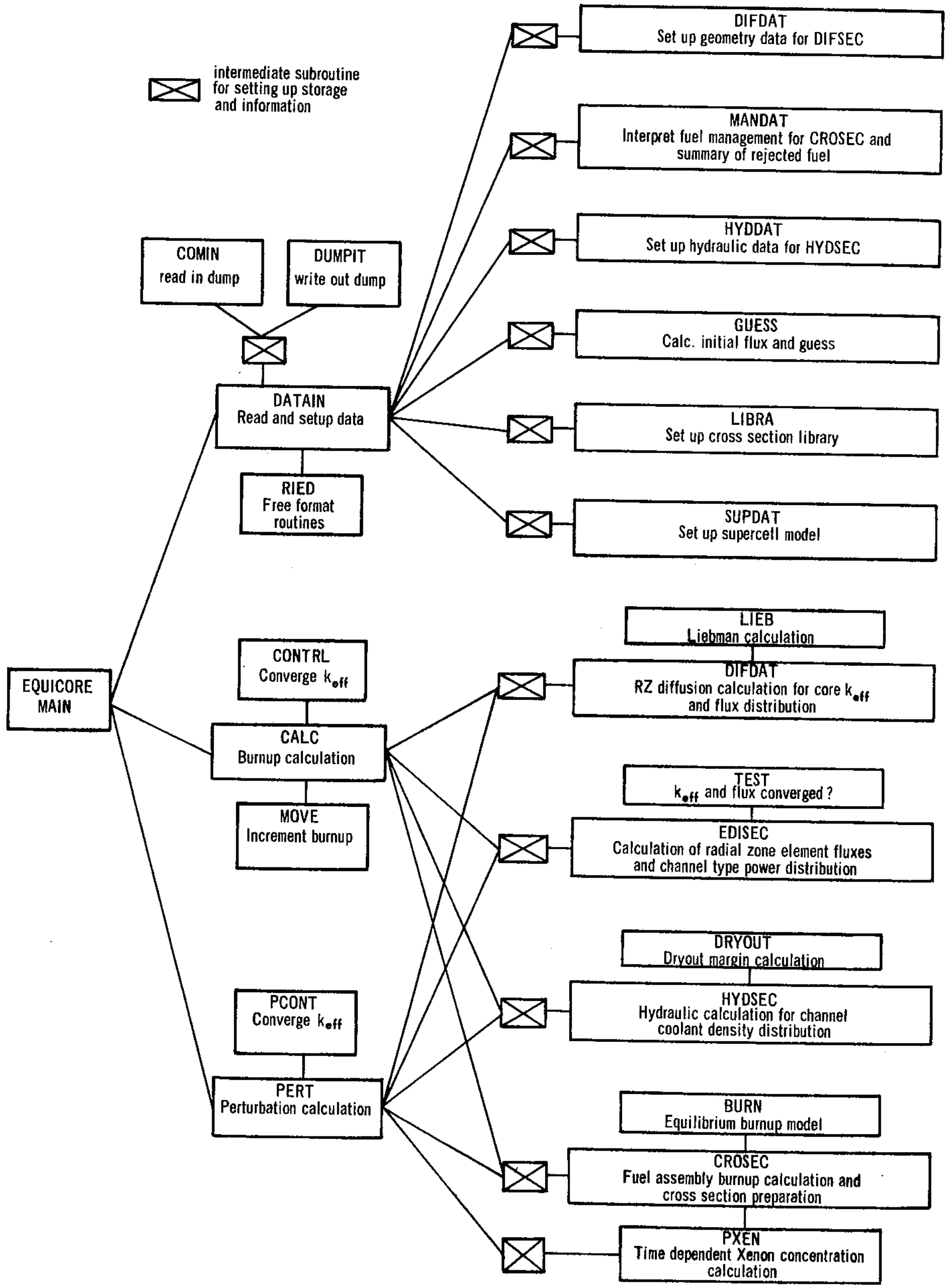

FIGURE 2. ARRANGEMENT OF EQUICORE CODE

APPENDIX A

EXAMPLE OF INPUT DATA FOR EQUICORE

A1. Sample of Equilibrium Burnup Calculation

The following input data is for a CANDU type on load refueled reactor having two fuel channel types to achieve the bi-directional once through refueling scheme. Because the core and fuel management are axially symetric only half the core is specified with the fuel being inverted to move down its neighbouring channel.

The core has been divided into 3 flattened (140 channels) and 6 unflattened radial zones (228) with 3 fuel assembly positions in the half core. The residence time fractions have been arranged to give a uniform burnup in the unflattened region. The notched reflector (used to save D_2O) is achieved by using an absorbing material as reflector R2. No hydraulic calculation is required.

A supercell calculation is set up for the 3rd radial zone using an arbitrary supercell radius equivalent to the centre cell volume. A PERT booster rod type calculation has also been requested for which XTRA data has been supplied.

```
* CANDU TYPE EQUILIB. UNIFORM. UNFLAT. ZONE BURNUP. HALF HEIGHT
LIB KCAN
DIF 297.2  9 24  9 2 -1 (140 3 3.0),(228 6 0.)
RM 0 FILL 19(6.14)20(7)21(10)22(16)23(25)24
ZM 0 FILL 9
R1 1.3152 0.8840 0.000091 0.00007 0.01157 0.000094 0
R2 1.3152 0.8840 10. 10.
C1 0 R1 9
C2 0 R2 2 R1 9
CCOR 0 Z1 1 FILL 9
CA 0 Z1 3 Z2 5 FILL 19 C1 22 C2 24
XTRA 0  0.0003  0  0.00065 4  1  3  8
MAN 2 3 9 6 1
CHAN 1 1.0 2
CHAN 2 1.0 2
CHAN 3 1.0 2
CHAN 4 1.0 2 0    .632
CHAN 5 1.0 2 0    .671
CHAN 6 1.0 2 0    0.740
CHAN 7 1.0 2 0   .843
CHAN 8 1.0 2 0   .987
CHAN 9 1.0 2 0   1.147
ASS  1  111  112  113 -123  122  121  REP 8
BURN  901.7  1.004   9000.
FLUX   7  0.  3.0  1.E+36  1.045
CONS 1.0E-5 0
     .96 1.518 28 0.0038 1.0 1426.0 2.1 272.
RUN  10  20  2
ENTER
RUN  30 20 1
PERT 2 30 10 1. 36.2 120 .63
ENTER
RUN  5  20  3
SUPER  1  3  3  7  0.02
ENTER
```

A2. Sample of Time Dependent Burnup (Complex Case)

The following input data is for a SGHW type reactor having a 9 batch off-load refueling scheme. To examine the operating conditions during the equilibium cycle, use is made of the equilibrium method to provide a reasonable starting fuel irradiation distribution.

The core is divided into 6 radial zones having the 9 channel types to represent the 9 batch refueling scheme. Radial zone 4 represents the region near the fuel enrichment boundary and therefore has 18 channel types to allow for both enrichments.

The 9 batch refueling scheme also necessitates different channel type volume fractions in zone 4 and 6 to achieve the correct number of fuel assemblies being loaded at each refueling. To achieve the equilibrium irradiation distribution, fuel assemblies are moved from one channel type to the other. This equilibrium fuel management is not entirely correct because of the different channel type volume fractions in zone 4 and 6.

Hydraulic data is provided for this case. A supercell calculation for channel type 9 radial zone 4 is requested after the core power distribution has been obtained for the second refueling. A check on the effect of a 10% change in power is also obtained via a PERT calculation.

```
*SGHW TYPE TIME DEPENDENT WITH COOLANT AND EQUILIBRIUM START
LIB SGHT
DIF  366. 8  16  12  2  Ø  (2Ø7 3 3.Ø), (144 2 Ø), (65 1 Ø)
R1  1.33Ø7  .92142  7.-6  7.578-5  1.Ø5Ø7-2  8.Ø-4 Ø.  5.23-6  1.88-4
R2  1.4Ø9  Ø.9Ø7  3.Ø-4  1.22-3  .Ø151  1.12-4 Ø  3.Ø-6  1.31-4
RM  Ø  FILL  13  (8.Ø) 14 (12) 15 (2Ø) 16
ZM Ø(2Ø)1(1Ø)2 FILL 12
C1  Ø  R1  12
CA  Ø  Z1  3  Z2  5  FILL 13  C1  16
CCOR Ø R2 2 Z1 4 Z2 5 FILL 1Ø Z8 12
MAN  18  1  7  18
CHAN  1 1.Ø 9
CHAN  2 1.Ø 9
CHAN  3 1.Ø 9
CHAN  4 1.Ø 18         .Ø69444  .Ø97222  .Ø69444
      .Ø97222  .Ø69444  .Ø97222  .Ø13889  .Ø97222
      .Ø69444  .Ø41667  .Ø13889  .Ø41667
      .Ø13889  .Ø41667  .Ø13889  .Ø97222  .Ø13889
      .Ø41667  Ø  9
CHAN  5 1.Ø 9
CHAN  6  1.Ø  9 .13846  .Ø9231  .13846  .Ø9231
      .13846  .Ø9231  .Ø7692  .Ø9231  .13846
ASS  1  111  121  131  141  151  161  171  181  191  REP 3
ASS  2  41Ø1  4111  4121  4131  4141  4151  4161  4171  4181
ASS 2 511 521 531 541 551 561 571 581 591 REP 1
FLUX  9  Ø.  Ø.  3.  1.Ø4
CONS 8.-6  1.-2  .986  1.  36  .ØØ3  .97  841.  4.48  272.  .6  .4
HYD  2  4  Ø  1  Ø  3  1  1  5  59.Ø
CIRC  Ø.96  8ØØ  495.2  881  Ø
      Ø.Ø34Ø5  Ø.Ø6338  Ø.9  Ø.Ø4173  Ø.95+5
FEED  1  1  17.Ø  2Ø.Ø  Ø.2Ø5  Ø.33  3.28  REP 5
FEED  1  2  Ø.Ø  6.5  Ø.41  Ø.132  Ø.57  REP 5
RISE  1  1  1Ø.25  23.Ø  Ø.2Ø5  Ø.Ø688  1.428  REP 5
RISE  1  2  Ø.  16.Ø  Ø.41  Ø.Ø688  2.5  REP 5
RISE  1  3  Ø  Ø  2Ø.Ø  Ø.Ø688  2.4  REP 5
RISE  1  4  Ø  6.5  6.5  Ø.Ø688  11.Ø  REP 5
BURN  1643.Ø  1.ØØ2  18ØØØ.  Ø.  1.2  5Ø.
RUN  25  2Ø  1
ENTER
RUN 2Ø 2Ø 1
BURN  1643.  1.ØØ2  18ØØØ.  Ø.  2.4  5Ø.
LOAD Ø
ENTER
LOAD 1 191 REP 3
LOAD  2 4181
LOAD 2 591  REP 1
RUN  2Ø  1  2
ENTER
RUN  1Ø  Ø  3
PERT  1  1  1.1
ENTER
SUPER 9 4 4 8
ENTER
RUN 25 2Ø 1
ENTER
```

A3. Sample of Simple EQUICORE Calculation

The following input data is for a simple representation of a typical SGHW reactor. This case is a simple version of Appendix Ab having only three radial zones and few mesh points. In addition to reduce running times the bottom reflector has been removed and the hydraulic calculation is only requested at the end of each step of the calculation. A check on the axial stability is requested via the PERT card.

```
*SGHWR SIMPLE CASE NO REFLECTOR & COOLANT AT END
LIB SGHT
DIF 366.8 4 9 6 2 0 (207 1 0 ) ( 90 1 0 ) ( 117 1 0)
R1  1.3307  .92142  7.-6  7.578-5  1.0507-2  8.8-4 0.  5.23-6 1.88-4
R2  1.409  0.907  3.0-4  1.22-3  .0151  1.12-4 0  3.06-6  1.31-4
RM  0  FILL  7  (15) 8 (25) 9
ZM 0 FILL 6
CCOR 0Z1 2 Z2 3 Z3 4 Z4 6
C1  0  R1  6
CA 0 Z1 3 Z2 5 FILL 7 C1 9
CONS 8.-6  1.-2  .986  1.  36  .003  .97  841.  4.48  272.  .6  .4
MAN 18 1 4 18
CHAN  1  1  9
CHAN  2  1  18  0  0  9
CHAN  3  1  9
ASS  1  111  121  131  141  151  161  171  181  191  REP 1
ASS2 2101 2111 2121 2131 2141 2151 2161 2171 2181
ASS2 311 321 331 341 351 361 371 381 391
HYD  2  4  0  5  0  3  1  1  5  59.0
CIRC  0.96  800  495.2  881  0
      0.03405  0.06338  10.8  0.04173  0.95+5
FEED  1  1  17.0  20.0  0.205  0.33  3.28  REP 2
FEED  1  2  0.0  6.5  0.41  0.132  0.57  REP 2
RISE  1  1  10.25  23.0  0.205  0.0688  1.428  REP 2
RISE  1  2  0.  16.0  0.41  0.0688  2.5  REP 2
RISE  1  3  0  0  20.0  0.0688  2.4  REP 2
RISE  1  4  0  6.5  6.5  0.0688  11.0  REP 2
FLUX 5 0 0 0 1.045
BURN  1653.  1.008  18500.  0.  0.9  50
RUN  10  20  1
ENTER
LOAD 0
BURN  1653.  1.008  18500  0.  1.8  50
ENTER
LOAD  1  191  REP 1
LOAD  2  2181
LOAD  2  391
RUN  10  1  2
ENTER
RUN  30  20  1
PERT  3  0  200  20  .2  1400
ENTER
RUN  10  20  1
ENTER
LOAD 1 181 REP1
LOAD 2 2171
LOAD 2 381
ENTER
```

APPENDIX B

EXTRACTS OF OUTPUT DATA FOR A TYPICAL EQUICORE CALCULATION

This output is extracted from the SGHW reactor calculation described in Appendix A2.

MAIN DIFFUSION CALCULATION SECTION PARAMETERS.....

CORE	CHANNELS	RADIUS(CM)	HEIGHT(CM)	VOLUME(CC)
	416.0000	211.5584	366.0000	1.0293E+08

MESH POINT COORDINATES

R MESH POINT	POSITION (CM)	Z MESH POINT	POSITION (CM)
1	17.587	1	10.000
2	52.762	2	25.000
3	87.937	3	41.438
4	123.112	4	64.313
5	158.287	5	98.625
6	184.668	6	144.375
7	202.255	7	190.125
8	219.542	8	235.875
9	236.527	9	281.625
10	252.470	10	327.375
11	267.371	11	361.688
12	280.914	12	384.563
13	293.097		
14	303.188		
15	313.188		
16	329.188		

RADIAL ZONE SPECIFICATIONS

ZONE	WIDTH (CM)	END POINT (CM)	VOLUMF FRACTION
1	105.525	105.525	0.12440
2	70.350	175.874	0.22115
3	35.175	211.049	0.15204
4	33.970	245.019	0.17308
5	29.803	274.822	0.17308
6	24.366	299.189	0.15625

REFLECTOR DATA

REFL.	CROSS SECTIONS D1	D2	A1	A2	R1	UF1	UF2	BORON SIG 1	AND 2
1	1.331E+00	9.214E-01	7.000E-06	7.578E-05	1.051E-02	8.800E-04	0.0	5.230E-06	1.880E-04
2	1.409E+00	9.070E-01	3.000E-04	1.220E-03	1.510E-02	1.120E-04	0.0	3.000E-06	1.310E-04

MESH LAYOUT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14
12	8	8	8	16	16	24	24	32	32	40	40	48	48	49
11	8	8	8	16	16	24	24	32	32	40	40	48	48	49
10	7	7	7	15	15	23	23	31	31	39	39	47	47	49
9	6	6	6	14	14	22	22	30	30	38	38	46	46	49
8	5	5	5	13	13	21	21	29	29	37	37	45	45	49
7	4	4	4	12	12	20	20	28	28	36	36	44	44	49
6	3	3	3	11	11	19	19	27	27	35	35	43	43	49
5	2	2	2	10	10	18	18	26	26	34	34	42	42	49
4	1	1	1	9	9	17	17	25	25	33	33	41	41	49
3	1	1	1	9	9	17	17	25	25	33	33	41	41	49
2	50	50	50	50	50	50	50	50	50	50	50	50	50	49
1	50	50	50	50	50	50	50	50	50	50	50	50	50	49

	15	16
12	49	49
11	49	49
10	49	49
9	49	49
8	49	49
7	49	49
6	49	49
5	49	49
4	49	49
3	49	49
2	49	49
1	49	49

CHANNEL TYPE ARRANGEMENTS

RADIAL ZONE	NUMBER OF CHAN. TYPES	CHAN. TYPE	VOLUME FRACTION	NO. OF CHANNELS	RESIDENCE FRACTION	POISON FRACTION	SUPERCELL RADIUS CM
1	9	1	0.1111	5.7500	1.0000	1.0000	44.0068
		2	0.1111	5.7500	1.0000		
		3	0.1111	5.7500	1.0000		
		4	0.1111	5.7500	1.0000		
		5	0.1111	5.7500	1.0000		
		6	0.1111	5.7500	1.0000		
		7	0.1111	5.7500	1.0000		
		8	0.1111	5.7500	1.0000		
		9	0.1111	5.7500	1.0000		
2	9	1	0.1111	10.2222	1.0000	1.0000	44.0068
		2	0.1111	10.2222	1.0000		
		3	0.1111	10.2222	1.0000		
		4	0.1111	10.2222	1.0000		
		5	0.1111	10.2222	1.0000		
		6	0.1111	10.2222	1.0000		
		7	0.1111	10.2222	1.0000		
		8	0.1111	10.2222	1.0000		
		9	0.1111	10.2222	1.0000		

ETC. FOR RADIAL ZONES 3 TO 6

EQUILIBRIUM FUEL MANAGEMENT CALCULATION

ASSEMBLY	MATERIAL	POSITION OF REST								
1	1	1 1 1,	1 2 1,	1 3 1,	1 4 1,	1 5 1,	1 6 1,	1 7 1,	1 8 1,	1 9 1,
2	1	2 1 1,	2 2 1,	2 3 1,	2 4 1,	2 5 1,	2 6 1,	2 7 1,	2 8 1,	2 9 1,
3	1	3 1 1,	3 2 1,	3 3 1,	3 4 1,	3 5 1,	3 6 1,	3 7 1,	3 8 1,	3 9 1,
4	1	4 1 1,	4 2 1,	4 3 1,	4 4 1,	4 5 1,	4 6 1,	4 7 1,	4 8 1,	4 9 1,
5	2	410 1,	411 1,	412 1,	413 1,	414 1,	415 1,	416 1,	417 1,	418 1,
6	2	5 1 1,	5 2 1,	5 3 1,	5 4 1,	5 5 1,	5 6 1,	5 7 1,	5 8 1,	5 9 1,
7	2	6 1 1,	6 2 1,	6 3 1,	6 4 1,	6 5 1,	6 6 1,	6 7 1,	6 8 1,	6 9 1,

**** CHANNEL HYDRAULIC INPUT DATA FOR CONSTANT FEEDER INLET PRESSURE ****

CHANNEL FEEDER INLET PRESSURE (PSIA) = 8.000E+02
CHANNEL FEEDER INLET ENTHALPY (BTU/LB) = 4.952E+02

CHANNEL RISER EXIT PRESSURE (PSIA) = 8.810E+02

INITIAL CHANNEL FLOW GUESS (LB/HR) = 9.500E+04

DATA FOR SELECTION OF CORRELATIONS:-

TWOPHASE MULTIPLIER OPTION = 5
SLIP OPTION = 3
TWOPHASE FRICTION MULTIPLIER OPTION = 1
DRYOUT CORRELATION SELECTED IS NUMBER 1

GEOMETRICAL DATA FOR FEEDER SECTION:-

RADIAL ZONE	SECTION NO.	VERTICAL HEIGHT (FT)	LENGTH (FT)	EQUIVALENT HYDRAULIC DIAMETER (FT)	FLOW AREA (FT2)	PRESSURE LOSS COEFFICIENT
1	1	1.700E+01	2.000E+01	2.050E-01	3.300E-01	3.280E+00
	2	0.0	6.500E+00	4.100E-01	1.320E-01	5.700E-01
2	1	1.700E+01	2.000E+01	2.050E-01	3.300E-01	3.280E+00
	2	0.0	6.500E+00	4.100E-01	1.320E-01	5.700E-01
3	1	1.700E+01	2.000E+01	2.050E-01	3.300E-01	3.280E+00
	2	0.0	6.500E+00	4.100E-01	1.320E-01	5.700E-01
4	1	1.700E+01	2.000E+01	2.050E-01	3.300E-01	3.280E+00
	2	0.0	6.500E+00	4.100E-01	1.320E-01	5.700E-01
5	1	1.700E+01	2.000E+01	2.050E-01	3.300E-01	3.280E+00
	2	0.0	6.500E+00	4.100E-01	1.320E-01	5.700E-01
6	1	1.700E+01	2.000E+01	2.050E-01	3.300E-01	3.280E+00
	2	0.0	6.500E+00	4.100E-01	1.320E-01	5.700E-01

GEOMETRICAL DATA FOR RISER SECTION:-

RADIAL ZONE	SECTION NO.	VERTICAL HEIGHT (FT)	LENGTH (FT)	EQUIVALENT HYDRAULIC DIAMETER (FT)	FLOW AREA (FT2)	PRESSURE LOSS COEFFICIENT
1	1	1.025E+01	2.300E+01	2.050E-01	6.880E-02	1.428E+00
	2	0.0	1.600E+01	4.100E-01	6.880E-02	2.500E+00
	3	0.0	0.0	2.000E+01	6.880E-02	2.400E+00
	4	0.0	6.500E+00	6.500E+00	6.880E-02	1.100E+01

ETC. FOR RADIAL ZONES 2 TO 6

```
BURN  1643.0  1.002  18000.  0.  1.2  50.

RUN 25 20 1

ENTER

GUESS OF FLUX, CROSS SECTION AND COOLANT DENSITY DISTRIBUTIONS GENERATED

INITIAL RESIDENCE TIME  103.05
RATE OF CHANGE OF K WITH TIME   1.397E-03
VOLUME/TIME FRACTION   0.115
REQUIRED STORAGE FOR IRRADIATION STEPS   864
NUMBER OF STEPS      5 CORE RATING    20.155


BOUNDARY R,B,T,   0.0            0.0            3.00000E+00

DATA STORAGE REQUIREMENTS:

CROSS SECTION PREPARATION SECTION TOTAL 9180
MADE UP AS FOLLOWS:-
LMCRS,LC8,LCRSE,LCRSR,LHYPH,LMAN,LXNDA 1810  350 1920   22 2748 2329    1

DIFFUSION CALCULATION SECTION TOTAL 2338
MADE UP AS FOLLOWS:-
LCRSG,LGEN,LSSG  374  620 1344

FLUX EDITING SECTION TOTAL 5384
MADE UP AS FOLLOWS:-
LCRSE,LGEN,LHYPH,LPH 1920  620 2748   96

ELAPSED TIME(MIN)   0.33

NEW POISON CONC.ESTIMATE 1.200E+00


     LIEBMAN CALCULATION REQUIRED
OMEGZ    1 1.69412D-01 0.0        -4.47214D+01 1.74637D+00
OMEGZ    2 2.27540D-01 0.0        -9.06848D+01 1.76445D+00
OMEGZ    2 2.78169D-01 0.0        -1.60323D+01 1.08131D+00
OMEGZ    2 2.77699D-01 0.0        -6.05702D+00 1.08115D+00
OMEGZ    2 2.65459D-01 0.0        -1.20445D+00 1.07697D+00

FLAPSED TIME(MIN)   1.03

     EQUILIBRIUM RESIDENCE TIME    103.05

FORM FACTORS:  RADIAL  1.2796 PEAK AXIAL  1.2612 CORE  1.6138 SMEARED CORE  1.3650
    PEAK RADIAL ZONE ELEMENT RATING IN ZONE  7 IS  27.501
    PEAK CHANNEL RATING IN AXIAL SLICE  7 CHANNEL TYPE 10 OF RADIAL ZONE  4 IS  32.526

ELAPSED TIME(MIN)   1.11
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ETC. DURING ITERATION PRINTOUT

EQUILIBRIUM RESIDENCE TIME 103.38

FORM FACTORS: RADIAL 1.3787 PEAK AXIAL 1.2198 CORE 1.6818 SMEARED CORE 1.2271
PEAK RADIAL ZONE ELEMENT RATING IN ZONE 28 IS 24.305
PEAK CHANNEL RATING IN AXIAL SLICE 4 CHANNEL TYPE 10 OF RADIAL ZONE 4 IS 33.896

CURRENT KEFF 1.001949E+00 GOG FINAL ACCURACY 9.1076E-05 PREV. KEFF 1.001945E+00
CURRENT RESIDENCE TIME 103.36 PREVIOUS RESIDENCE TIME102.78
NEW OVERALL RESIDENCE TIME 103.33 RATE OF CHANGE OF K WITH TIME 9.157E-04

ELAPSED TIME(MIN) 19.33

ELAPSED TIME(MIN) 19.70

RADIAL ZONE AND CHANNEL TYPE RATING DISTRIBUTIONS(W/GRM)

EQUILIBRIUM RESIDENCE TIME 103.33

RADIAL ZONE 1

AXIAL SLICE	ZONE AVERAGE	CHANNEL TYPE NOS, 1 TO 9 1	2	3	4	5	6	7	8	9
8	10.501	11.495	11.296	11.071	10.807	10.530	10.248	9.955	9.683	9.420
7	20.606	24.921	23.910	22.728	21.535	20.398	19.347	18.358	17.510	16.751
6	23.485	29.402	27.907	26.227	24.589	23.077	21.717	20.467	19.429	18.550
5	24.217	30.867	29.127	27.210	25.363	23.671	22.176	20.871	19.800	18.870
4	24.086	31.025	29.120	27.080	25.192	23.468	21.935	20.639	19.603	18.713
3	23.465	30.020	28.197	26.272	24.452	22.813	21.376	20.240	19.305	18.510
2	21.894	27.298	25.836	24.270	22.780	21.435	20.254	19.229	18.348	17.593
1	16.440	19.283	18.614	17.850	17.072	16.322	15.619	14.972	14.382	13.847
TOT.MW.	4.234	5.005	4.752	4.475	4.208	3.961	3.740	3.545	3.382	3.240

RADIAL ZONE 2

AXIAL SLICE	ZONE AVERAGE	CHANNEL TYPE NOS, 1 TO 9 1	2	3	4	5	6	7	8	9

ETC. FOR RADIAL ZONES 2 TO 6

FORM FACTORS: RADIAL 1.3785 PEAK AXIAL 1.2198 CORE 1.6814 SMEARED CORE 1.2269
PEAK RADIAL ZONE ELEMENT RATING IN ZONE 29 IS 24.309
PEAK CHANNEL RATING IN AXIAL SLICE 4 CHANNEL TYPE 10 OF RADIAL ZONE 4 IS 33.889

POINT RATING MAP (W/GR) AND CHANNEL DISTRIBUTION (MW)'

	1	2	3	4	5	6	7	8	9	10	11	12	13	14
12	6.27	6.27	6.27	6.27	6.25	6.24	6.21	6.31	6.15	6.23	5.79	5.35	5.04	0.0
11	14.74	14.73	14.73	14.72	14.69	14.65	14.58	14.82	14.47	14.67	13.65	12.65	11.96	0.0
10	20.61	20.61	20.60	20.60	20.56	20.52	20.44	20.77	20.30	20.69	19.34	18.24	17.43	0.0
9	23.49	23.49	23.48	23.48	23.45	23.42	23.34	23.73	23.24	23.74	22.25	21.11	20.19	0.0
8	24.22	24.22	24.22	24.21	24.20	24.18	24.12	24.55	24.08	24.65	23.13	21.98	20.90	0.0
7	24.09	24.09	24.09	24.08	24.09	24.09	24.06	24.52	24.10	24.73	23.20	22.04	20.77	0.0
6	23.46	23.46	23.47	23.47	23.48	23.50	23.49	23.98	23.60	24.27	22.73	21.56	20.13	0.0
5	21.89	21.89	21.90	21.90	21.92	21.94	21.95	22.43	22.08	22.71	21.20	19.99	18.51	0.0
4	18.90	18.90	18.91	18.91	18.93	18.95	18.96	19.40	19.08	19.58	18.20	16.92	15.49	0.0
3	13.97	13.97	13.98	13.98	13.99	14.00	14.00	14.31	14.05	14.38	13.31	12.29	11.12	0.0
2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
M.W.	4.034	4.034	4.034	4.034	4.032	4.031	4.024	4.100	4.024	4.121	3.854	3.638	3.423	0.0

	15	16
12	0.0	0.0
11	0.0	0.0
10	0.0	0.0
9	0.0	0.0
8	0.0	0.0
7	0.0	0.0
6	0.0	0.0
5	0.0	0.0
4	0.0	0.0
3	0.0	0.0
2	0.0	0.0
1	0.0	0.0
M.W.	0.0	0.0

ELAPSED TIME(MIN) 19.81

**** CHANNEL HYDRAULIC RESULTS FOR CONSTANT FEEDER INLET PRESSURE ****

C.I.S.E. CORRELATION USED

RADIAL ZONE	CHANNEL TYPE	FLOW (LBS/HR)	RISER EXIT PRESSURE (PSI)	QUALITY	MINIMUM CRIT. HEAT FLUX
1	1	1.365E+05	799.994	0.153	3.813E+00
	2	1.447E+05	799.904	0.135	4.120E+00
	3	1.507E+05	800.041	0.120	4.488E+00
	4	1.572E+05	800.026	0.106	4.881E+00
	5	1.637E+05	800.021	0.094	5.282E+00
	6	1.700E+05	800.016	0.084	5.680E+00
	7	1.786E+05	800.059	0.073	6.070E+00
	8	1.837E+05	800.030	0.066	6.435E+00
		WARNING..FLOW LIMITS EXCEEDED			
	9	1.884E+05	800.024	0.061	6.772E+00
2	1	1.365E+05	799.993	0.153	3.818E+00
	2	1.447E+05	799.901	0.135	4.125E+00
	3	1.507E+05	800.038	0.120	4.492E+00
	4	1.572E+05	800.021	0.106	4.886E+00
	5	1.637E+05	800.016	0.094	5.287E+00
	6	1.700E+05	800.013	0.083	5.685E+00
	7	1.786E+05	800.053	0.073	6.075E+00
	8	1.838E+05	800.023	0.066	6.440E+00
		WARNING..FLOW LIMITS EXCEEDED			
	9	1.884E+05	800.019	0.061	6.778E+00
3	1	1.366E+05	800.002	0.153	3.835E+00
	2	1.448E+05	799.908	0.135	4.142E+00
	3	1.508E+05	800.030	0.120	4.509E+00
	4	1.573E+05	800.028	0.106	4.903E+00
	5	1.638E+05	800.021	0.094	5.304E+00
	6	1.701E+05	800.017	0.083	5.703E+00
	7	1.787E+05	800.057	0.073	6.094E+00
	8	1.839E+05	800.029	0.066	6.459E+00
		WARNING..FLOW LIMITS EXCEEDED			
	9	1.885E+05	800.024	0.060	6.798E+00
4	1	1.420E+05	799.866	0.142	3.998E+00
	2	1.469E+05	800.025	0.129	4.292E+00
	3	1.531E+05	800.012	0.115	4.657E+00
	4	1.595E+05	800.011	0.102	5.049E+00

ETC. FOR RADIAL ZONES 4 TO 6

EQUILIBRIUM BURNUP CONVERGED

FILE NUMBER 1 DUMPED

SUMMARY FOLLOWS

*SGHW TYPE TIME DEPENDENT WITH COOLANT AND EQUILIBRIUM START

ASSM.	MAT.	FUEL SUPPLY/DAY WT(TU)	NO.ASSM	TOT.RESID. TIME(DAYS)	FINAL BURNUP (MWD/TU)	U235	PU.FIS GRM/KLOGRAM U	PU.TOT
1	1	0.0109	0.056	929.97	19241.	5.291E+00	3.891E+00	6.128E+00
2	1	0.0194	0.099	929.97	19236.	5.293E+00	3.891E+00	6.126E+00
3	1	0.0133	0.068	929.97	19211.	5.304E+00	3.889E+00	6.121E+00
4	1	0.0095	0.048	929.97	18814.	5.461E+00	3.862E+00	6.039E+00
5	2	0.0057	0.029	929.97	20628.	6.767E+00	4.000E+00	6.197E+00
6	2	0.0152	0.077	929.97	19001.	7.518E+00	3.890E+00	5.872E+00
7	2	0.0171	0.087	929.97	16706.	8.700E+00	3.703E+00	5.381E+00

MAT.	FUEL SUPPLY/DAY WT(TU)	NO.ASSM	AVE.RESID. TIME(DAYS)	AVRGE BURNUP (MWD/TU)	U235	PU.FIS GRM/KLOGRAM U	PU.TOT
1	0.0531	0.271	929.97	19155.	5.326E+00	3.886E+00	6.110E+00
2	0.0379	0.194	929.97	18212.	7.937E+00	3.823E+00	5.700E+00
TOTALS	0.0910	0.465	929.97	18762.	6.414E+00	3.859E+00	5.939E+00

CORE FORM FACTOR 1.6814 CHAN. RADIAL F.F. 1.3785 AVE. AXIAL F.F. 1.2198

KEFF	CORE BURNUP	CORRECTED CORE BURNUP	TOTAL FUEL SUPPLY/DAY WT(TU)	NO.ASSM
1.00203	18049.1	18054.4	0.091	0.465

RUN 20 20 1

BURN 1643. 1.002 18070. 0. 2.4 50.

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 FUEL LOADING INPUT DATA

LOAD Ø


 CHANGE FUEL LOADING FROM EQILIBRIUM TO APPROACH TYPE

 LOADING OF FUEL AS PER FIRST PART OF IRRADIATION CYCLE


         TIME DEPENDENT BURNUP CALCULATION

REACTOR TIME AT STAGE  1      Ø.Ø  DAYS

FUEL LOADING  IS


RADIAL ZONE   1


ASSEMBLY     CHANNEL TYPE NOS, 1 TO  9
POSITION      1     2     3     4     5     6     7     8     9

    1         1  1Ø11  1Ø21  1Ø31  1Ø41  1Ø51  1Ø61  1Ø71  1Ø81


RADIAL ZONE   2


ASSEMBLY     CHANNEL TYPE NOS, 1 TO  9
POSITION      1     2     3     4     5     6     7     8     9

    1         1  2Ø11  2Ø21  2Ø31  2Ø41  2Ø51  2Ø61  2Ø71  2Ø81


RADIAL ZONE   3


ASSEMBLY     CHANNEL TYPE NOS, 1 TO  9
POSITION      1     2     3     4     5     6     7     8     9

    1         1  3Ø11  3Ø21  3Ø31  3Ø41  3Ø51  3Ø61  3Ø71  3Ø81


RADIAL ZONE   4


ASSEMBLY     CHANNEL TYPE NOS, 1 TO 18
POSITION      1     2     3     4     5     6     7     8     9    1Ø
               11    12    13    14    15    16    17    18

    1         1  4Ø11  4Ø21  4Ø31  4Ø41  4Ø51  4Ø61  4Ø71  4Ø81     2
             41Ø1  4111  4121  4131  4141  4151  4161  4171
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ETC. FOR RADIAL ZONES 5 TO 6

BURN UP STAGE COMPLETED. REACTOR TIME 100.7 DAYS

FILE NUMBER 2 DUMPED

FUEL LOADING INPUT DATA

LOAD 1 191 REP 3

LOAD 2 4181

LOAD 2 591 REP 1

TIME DEPENDENT BURNUP CALCULATION

REACTOR TIME AT STAGE 1 100.72 DAYS

FUEL LOADING IS

RADIAL ZONE 1

ASSEMBLY POSITION	CHANNEL TYPE NOS, 1 TO 9: 1	2	3	4	5	6	7	8	9
1	1010	1020	1030	1040	1050	1060	1070	1080	1

ETC. FOR RADIAL ZONES 2 TO 6

SUMMARY OF FUEL REMOVED

RADIAL ZONE	CHAN. TYPE	ASSEMBLY POSITION	FUEL TYPE	NO.OF ASSEMBLIES	TOTAL WEIGHT(TU)	BURNUP MWD/TU	U235 GRM/KLOGRAM U	PU.FIS GRM/KLOGRAM U	PU.TOT GRM/KLOGRAM U
1	9	1	1	5.75	1.127	19206.8	5.292E+00	3.869E+00	6.298E+00
2	9	1	1	10.22	2.003	19196.8	5.302E+00	3.876E+00	6.105E+00
3	9	1	1	7.03	1.377	19163.9	5.322E+00	3.883E+00	6.108E+00
4	9	1	1	5.00	0.980	18758.4	5.487E+00	3.863E+00	6.233E+00
4	18	1	2	3.00	0.588	20569.6	6.797E+00	4.001E+00	6.190E+00
5	9	1	2	8.00	1.568	18935.1	7.561E+00	3.899E+00	5.873E+00
6	9	1	2	9.00	1.764	16627.2	8.760E+00	3.722E+00	5.392E+00

AVERAGE FOR EACH FUEL MATERIAL

ZONE	TYPE	POSITION	FUEL TYPE	NO.OF ASSEMBLIES	TOTAL WEIGHT(TU)	BURNUP MWD/TU	U235 GRM/KLOGRAM U	PU.FIS GRM/KLOGRAM U	PU.TOT GRM/KLOGRAM U
			1	28.00	5.487	19112.3	5.338E+00	3.874E+00	6.092E+00
			2	20.00	3.919	18141.7	7.986E+00	3.834E+00	5.704E+00
			TOTALS	48.00	9.406	18707.9	6.441E+00	3.858E+00	5.930E+00

REACTOR TIME 100.72 DAYS, BURNUP STAGE 1

FORM FACTORS: RADIAL 1.3762 PEAK AXIAL 1.2440 CORE 1.7122 SMEARED CORE 1.2466
PEAK RADIAL ZONE ELEMENT RATING IN ZONE 5 IS 24.882
PEAK CHANNEL RATING IN AXIAL SLICE 4 CHANNEL TYPE 18 OF RADIAL ZONE 4 IS 34.504

CURRENT KEFF 1.002102E+00 GOG FINAL ACCURACY 5.4598E-05 PREV. KEFF 1.004313E+00

REACTOR TIME 100.72 DAYS, BURNUP STAGE 1
CURRENT POISON CONC. 2.6503E+00 PREVIOUS 2.4619E+00
RATE OF CHANGE OF K WITH TIME 3.004E-04 WITH BORON 1.173E-02

NEW POISON CONC.ESTIMATE 2.659E+00

ELAPSED TIME(MIN) 44.03

ELAPSED TIME(MIN) 44.37

RADIAL ZONE AND CHANNEL TYPE RATING DISTRIBUTIONS(W/GRM)

REACTOR TIME 100.72 DAYS, BURNUP STAGE 1

RADIAL ZONE 1

AXIAL SLICE	ZONE AVERAGE	CHANNEL TYPE NOS, 1 TO 9 1	2	3	4	5	6	7	8	9
8	10.178	10.916	10.730	10.482	10.227	9.950	9.667	9.401	9.144	11.088
7	20.520	23.828	22.706	21.525	20.385	19.305	18.289	17.410	16.626	24.605
6	23.833	28.388	26.748	25.075	23.514	22.073	20.752	19.646	18.709	29.591
5	24.833	29.964	28.064	26.151	24.385	22.759	21.364	20.225	19.208	31.396
4	24.785	30.092	28.039	26.061	24.255	22.594	21.190	20.061	19.104	31.669
3	24.058	29.027	27.104	25.206	23.482	21.933	20.692	19.684	18.829	30.567
2	22.115	26.172	24.635	23.116	21.717	20.473	19.389	18.455	17.656	27.422
1	16.124	18.229	17.531	16.784	16.048	15.349	14.698	14.103	13.566	18.811
TOT.MW.	4.077	4.816	4.545	4.272	4.018	3.783	3.577	3.404	3.254	5.025

RADIAL ZONE 2

ETC. FOR RADIAL ZONES 2 TO 6

**** CHANNEL HYDRAULIC RESULTS FOR CONSTANT FEEDEP INLET PRESSURE ****

C.I.S.E. CORRELATION USED

EXPLANATION OF WARNING ASTERISKS : *P MEANS PRESSURE LIMITS EXCEEDED
*Q MEANS QUALITY LIMITS EXCEEDED
*B MEANS CORRELATION HAS BLOWN UP

CHANNEL TYPE 1 OF RADIAL ZONE 1 : COOLANT FLOW(LBS/HR) IS 1.435E+05

POS'N FT	PRESSURE PSI	ENTHALPY BTU/LB	TEMP. F	QUALITY PERCENT	VOIDAGE FRAC.	SLIP	TFM	STEP POWER MW(TH)	CRIT. HEAT FLUX CHU/FT2/HR.C	CRIT. HEAT FLUX RATIO	
0.0	874.95	495.20	506.24	-4.053	0.0			NO VALUES	FOR 0 FT		
1.501	873.76	505.39	514.71	-2.510	0.0	1.000	1.000	4.287E-01	1.394E+06	8.689E+00	*Q
3.002	872.56	520.03	526.66	-0.309	0.0	1.000	1.000	6.155E-01	1.323E+06	5.745E+00	*Q
4.503	871.39	536.26	528.17	2.127	0.27579	1.408	1.000	6.826E-01	1.245E+06	4.874E+00	
6.004	868.09	553.09	527.73	4.698	0.43740	1.563	3.772	7.076E-01	1.163E+06	4.392E+00	
7.505	864.01	569.84	527.18	7.272	0.53308	1.703	5.290	7.046E-01	1.081E+06	4.101E+00	
9.006	859.17	585.72	526.52	9.726	0.59565	1.827	6.738	6.676E-01	1.004E+06	4.018E+00	
10.507	853.72	599.04	525.78	11.811	0.63639	1.927	7.968	5.603E-01	9.383E+05	4.475E+00	
12.008	848.14	605.15	525.01	12.829	0.65392	1.977	8.568	2.567E-01	9.072E+05	9.443E+00	
RISER EXIT	799.74	605.15	518.22	13.850					MINIMUM	4.018E+00	

CHANNEL TYPE 2 OF RADIAL ZONE 1 : COOLANT FLOW(LBS/HR) IS 1.495E+05

POS'N FT	PRESSURE PSI	ENTHALPY BTU/LB	TEMP. F	QUALITY PERCENT	VOIDAGE FRAC.	SLIP	TFM	STEP POWER MW(TH)	CRIT. HEAT FLUX CHU/FT2/HR.C	CRIT. HEAT FLUX RATIO	
0.0	874.93	495.20	506.24	-4.052	0.0			NO VALUES	FOR 0 FT		
1.501	873.69	504.61	514.06	-2.623	0.0	1.000	1.000	4.122E-01	1.380E+06	8.944E+00	*Q
3.002	872.44	517.84	524.89	-0.630	0.0	1.000	1.000	5.793E-01	1.316E+06	6.070E+00	*Q
4.503	871.21	532.39	528.15	1.558	0.22135	1.371	1.000	6.374E-01	1.246E+06	5.223E+00	
6.004	868.01	547.45	527.72	3.865	0.39560	1.515	3.280	6.593E-01	1.172E+06	4.750E+00	
7.505	864.06	562.52	527.18	6.188	0.49836	1.647	4.650	6.599E-01	1.098E+06	4.447E+00	
9.006	859.38	576.88	526.55	8.417	0.56545	1.765	5.965	6.290E-01	1.028E+06	4.367E+00	
10.507	854.09	589.07	525.83	10.332	0.60934	1.860	7.095	5.339E-01	9.678E+05	4.843E+00	
12.008	848.66	594.83	525.08	11.299	0.62867	1.910	7.666	2.523E-01	9.381E+05	9.935E+00	
RISER EXIT	799.75	594.83	518.22	12.353					MINIMUM	4.367E+00	

ETC. FOR RADIAL ZONES 3 TO 6

DISTANCE CM	IRRAD. MWD/TU	RATING MW/TU	COOLANT DENS. (GM/CC)	FUEL C	FAST FLUX	THERMAL FLUX	FLUX PEAKING FACTOR
ASSEMBLY 1 MATERIAL 1							
POSITION 1 11							
343.1	1146.	10.917	0.289	435.9	5.823E+13	3.661E+13	1.018E+00
297.4	2512.	23.829	0.310	708.8	1.235E+14	8.030E+13	1.033E+00
251.6	2991.	28.388	0.347	837.3	1.425E+14	9.710E+13	1.037E+00
205.9	3157.	29.962	0.403	892.5	1.440E+14	1.044E+14	1.039E+00
160.1	3181.	30.089	0.492	897.1	1.370E+14	1.075E+14	1.040E+00
114.4	3076.	29.026	0.643	859.8	1.263E+14	1.068E+14	1.039E+00
68.6	2782.	26.172	0.766	773.4	1.121E+14	9.797E+13	1.035E+00
22.9	1939.	18.229	0.776	569.0	7.594E+13	6.826E+13	1.027E+00
AVERAGE	2598.	24.576	0.503	746.7	1.149E+14	8.736E+13	
ASSEMBLY 2 MATERIAL 1							
POSITION 1 21							
343.1	2315.	10.731	0.308	433.2	5.823E+13	3.661E+13	1.013E+00
297.4	5001.	22.708	0.331	693.0	1.235E+14	8.030E+13	1.021E+00
251.6	5900.	26.750	0.370	815.0	1.425E+14	9.710E+13	1.023E+00
205.9	6195.	28.065	0.430	857.9	1.440E+14	1.044E+14	1.024E+00
160.1	6220.	28.038	0.527	857.2	1.370E+14	1.075E+14	1.024E+00
114.4	6020.	27.103	0.670	826.3	1.263E+14	1.068E+14	1.023E+00
68.6	5477.	24.635	0.767	750.2	1.121E+14	9.797E+13	1.021E+00
22.9	3875.	17.531	0.776	556.0	7.594E+13	6.826E+13	1.017E+00
AVERAGE	5125.	23.195	0.522	723.6	1.149E+14	8.736E+13	
ASSEMBLY 3 MATERIAL 1							
POSITION 1 31							
343.1	3463.	10.483	0.330	429.6	5.823E+13	3.661E+13	1.007E+00
297.4	7365.	21.529	0.354	669.4	1.235E+14	8.030E+13	1.009E+00
251.6	8630.	25.079	0.397	779.4	1.425E+14	9.710E+13	1.009E+00
205.9	9028.	26.151	0.462	814.0	1.440E+14	1.044E+14	1.009E+00
160.1	9042.	26.059	0.566	811.3	1.370E+14	1.075E+14	1.009E+00
114.4	8759.	25.204	0.698	783.9	1.263E+14	1.068E+14	1.008E+00
68.6	8007.	23.116	0.768	717.9	1.121E+14	9.797E+13	1.008E+00
22.9	5734.	16.784	0.777	540.4	7.594E+13	6.826E+13	1.008E+00
AVERAGE	7503.	21.801	0.544	693.2	1.149E+14	8.736E+13	

ETC. FOR ALL POSITIONS IN CORE

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BURN UP STAGE COMPLETED.  REACTOR TIME 100.7 DAYS

FILE NUMBER     3 DUMPED

RUN  10  0  3

PERT  1  1  1.1

ADDITIONAL STORAGE REQUIRED FOR XENON CALC 2592

ENTER

POWER COEF. PERT CALCULATION

  REACTOR TIME    100.72 DAYS,  BURNUP STAGE   1

FORM FACTORS:  RADIAL  1.3764 PEAK AXIAL  1.2428 CORE  1.7106 SMEARED CORE  1.2455
    PEAK RADIAL ZONE ELEMENT RATING IN ZONE  5 IS  27.322
    PEAK CHANNEL RATING IN AXIAL SLICE  4 CHANNEL TYPE 18 OF RADIAL ZONE  4 IS  37.924

MINIMUM REACTOR CHANNEL CRITICAL HEAT FLUX RATIO =  2.834E+00
AVERAGE CHANNEL FLOW (LBS/HR) =  1.541E+05
TOTAL REACTOR FLOW =  6.412E+07

CURRENT KEFF. 1.001079E+00  GOG FINAL ACCURACY 7.3969E-05  PREVIOUS KEFF. 1.002102E+00

POISON CONCENTRATION ESTIMATE 2.581E+00  PREVIOUS ESTIMATE 2.659E+00

  REACTOR TIME    100.72 DAYS,  BURNUP STAGE   1

FORM FACTORS:  RADIAL  1.3670 PEAK AXIAL  1.2477 CORE  1.7056 SMEARED CORE  1.2524
    PEAK RADIAL ZONE ELEMENT RATING IN ZONE  4 IS  27.717
    PEAK CHANNEL RATING IN AXIAL SLICE  4 CHANNEL TYPE 18 OF RADIAL ZONE  4 IS  37.813

MINIMUM REACTOR CHANNEL CRITICAL HEAT FLUX RATIO =  2.906E+00
AVERAGE CHANNEL FLOW (LBS/HR) =  1.548E+05
TOTAL REACTOR FLOW =  6.439E+07

CURRENT KEFF. 1.001956E+00  GOG FINAL ACCURACY 5.1081E-05  PREVIOUS KEFF. 1.001079E+00

POISON CONCENTRATION ESTIMATE 2.577E+00  PREVIOUS ESTIMATE 2.581E+00

  REACTOR TIME    100.72 DAYS,  BURNUP STAGE   1
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ETC.DURING PERT ITERATION

SUPERCELL CALCULATION REQUESTED

SUPERCELL RADIAL MESH LAYOUT

MESH NO.	1	2	3	4	5	6	7	8
	9	10	11	12				
RADIUS(CM)	1.83	5.50	9.17	12.84	16.50	20.17	23.84	27.50
	31.17	34.84	38.51	42.17				
ZONE NO.	1	1	1	1	2	2	2	2
	2	2	2	2				

MESH LAYOUT

	1	2	3	4	5	6	7	8	9	10	11	12
12	8	8	8	8	16	16	16	16	16	16	16	16
11	8	8	8	8	16	16	16	16	16	16	16	16
10	7	7	7	7	15	15	15	15	15	15	15	15
9	6	6	6	6	14	14	14	14	14	14	14	14
8	5	5	5	5	13	13	13	13	13	13	13	13
7	4	4	4	4	12	12	12	12	12	12	12	12
6	3	3	3	3	11	11	11	11	11	11	11	11
5	2	2	2	2	10	10	10	10	10	10	10	10
4	1	1	1	1	9	9	9	9	9	9	9	9
3	1	1	1	1	9	9	9	9	9	9	9	9
2	18	18	18	18	18	18	18	18	18	18	18	18
1	18	18	18	18	18	18	18	18	18	18	18	18

SUPERCELL POWER 36.609 MW NUMBER OF FUEL CHANNELS 9.000

CENTRE FUEL MATERIAL TYPE AND IRRADIATION DISTRIBUTION

FUEL AXIAL SLICE	MATERIAL TYPE	IRRADIATION MWD/T
8	4091	0.0
7		0.0
6		0.0
5		0.0
4		0.0
3		0.0
2		0.0
1		0.0

ENTER

ELAPSED TIME(MIN) 51.53

NEW POISON CONC.ESTIMATE 2.659E+00

RADIAL ZONE AND CHANNEL TYPE RATING DISTRIBUTIONS(W/GRM)

REACTOR TIME 100.72 DAYS, BURNUP STAGE 1

RADIAL ZONE 1

AXIAL SLICE	ZONE AVERAGE	CHANNEL TYPE NOS, 1 TO 1 1
8	10.499	10.499
7	23.296	23.296
6	28.064	28.064
5	29.689	29.689
4	29.786	29.786
3	28.895	28.895
2	26.096	26.096
1	17.843	17.843
TOT.MW.	4.756	4.756

RADIAL ZONE 2

AXIAL SLICE	ZONE AVERAGE	CHANNEL TYPE NOS, 1 TO 17 1 10	2 11	3 12	4 13	5 14	6 15	7 16	8 17	9
8	9.996	10.382	10.207	9.981	9.748	9.481	9.232	8.984	8.742	11.590
		11.354	11.068	10.779	10.487	10.191	9.889	9.609	11.815	
7	20.153	22.875	21.840	20.729	19.661	18.607	17.676	16.832	16.071	25.407
		24.124	22.826	21.565	20.400	19.310	18.290	17.406	26.437	
6	23.452	27.425	25.881	24.286	22.800	21.372	20.153	19.089	18.195	30.448
		28.582	26.728	24.979	23.410	21.979	20.669	19.565	32.007	
5	24.405	28.970	27.165	25.324	23.633	22.069	20.770	19.638	18.656	32.212
		30.047	27.915	25.923	24.166	22.584	21.210	20.062	34.045	
4	24.241	28.961	27.046	25.148	23.410	21.819	20.542	19.447	18.513	32.305
		29.972	27.690	25.656	23.868	22.244	20.908	19.803	34.297	
3	23.371	27.783	25.963	24.155	22.513	21.091	19.952	18.981	18.156	30.879
		28.671	26.556	24.624	22.929	21.431	20.279	19.304	32.861	
2	21.362	24.892	23.452	22.026	20.708	19.532	18.505	17.618	16.857	27.542
		25.764	24.084	22.546	21.177	19.975	18.928	18.024	29.184	
1	15.570	17.205	16.561	15.868	15.183	14.530	13.919	13.364	12.858	19.100
		18.276	17.439	16.630	15.867	15.158	14.506	13.908	19.808	
TOT.MW.	3.982	4.617	4.363	4.103	3.862	3.637	3.448	3.281	3.137	5.131
		4.820	4.515	4.230	3.976	3.745	3.544	3.373	5.400	

FORM FACTORS: RADIAL 1.3275 PEAK AXIAL 1.2446 CORE 1.6523 SMEARED CORE 1.4460
PEAK RADIAL ZONE ELEMENT RATING IN ZONE 4 IS 29.786
PEAK CHANNEL RATING IN AXIAL SLICE 4 CHANNEL TYPE 17 OF RADIAL ZONE 2 IS 34.297

POINT RATING MAP (W/GR) AND CHANNEL DISTRIBUTION (MW)'

	1	2	3	4	5	6	7	8	9	10	11	12
12	6.25	6.24	6.23	6.21	5.98	5.97	5.95	5.94	5.93	5.92	5.92	5.92
11	14.84	14.82	14.79	14.74	14.19	14.14	14.10	14.07	14.04	14.03	14.01	14.01
10	23.47	23.42	23.33	23.19	20.49	20.37	20.26	20.18	20.12	20.07	20.04	20.03
9	28.29	28.23	28.10	27.93	23.90	23.74	23.60	23.49	23.41	23.34	23.30	23.28
8	29.92	29.85	29.73	29.56	24.88	24.72	24.57	24.45	24.36	24.29	24.24	24.22
7	30.02	29.95	29.82	29.66	24.73	24.57	24.42	24.29	24.19	24.12	24.07	24.05
6	29.19	29.11	28.95	28.72	23.93	23.73	23.56	23.42	23.31	23.24	23.19	23.17
5	26.42	26.33	26.16	25.90	21.93	21.71	21.53	21.40	21.30	21.23	21.19	21.17
4	20.91	20.85	20.73	20.56	18.42	18.26	18.14	18.05	17.96	17.93	17.91	17.89
3	15.14	15.10	15.03	14.91	13.37	13.27	13.19	13.13	13.09	13.06	13.04	13.03
2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
M.W.	4.798	4.786	4.764	4.732	4.062	4.033	4.008	3.989	3.973	3.962	3.955	3.952

FAST FLUX

1	2.174E+01	2.171E+01	2.166E+01	2.159E+01	2.151E+01	2.143E+01	2.135E+01	2.129E+01	2.123E+01	2.119E+01	2.117E+01	2.116E+01
2	1.264E+02	1.262E+02	1.257E+02	1.250E+02	1.243E+02	1.235E+02	1.229E+02	1.224E+02	1.221E+02	1.218E+02	1.216E+02	1.215E+02
3	4.450E+02	4.437E+02	4.413E+02	4.376E+02	4.327E+02	4.291E+02	4.264E+02	4.244E+02	4.229E+02	4.219E+02	4.213E+02	4.210E+02
4	6.591E+02	6.572E+02	6.534E+02	6.477E+02	6.402E+02	6.346E+02	6.303E+02	6.270E+02	6.246E+02	6.230E+02	6.219E+02	6.214E+02
5	8.388E+02	8.353E+02	8.283E+02	8.178E+02	8.035E+02	7.933E+02	7.859E+02	7.803E+02	7.763E+02	7.736E+02	7.718E+02	7.710E+02
6	9.645E+02	9.598E+02	9.502E+02	9.355E+02	9.150E+02	9.009E+02	8.908E+02	8.835E+02	8.784E+02	8.748E+02	8.726E+02	8.716E+02
7	1.055E+03	1.049E+03	1.038E+03	1.022E+03	9.981E+02	9.820E+02	9.708E+02	9.627E+02	9.570E+02	9.532E+02	9.508E+02	9.496E+02
8	1.104E+03	1.099E+03	1.088E+03	1.072E+03	1.050E+03	1.035E+03	1.024E+03	1.016E+03	1.010E+03	1.007E+03	1.004E+03	1.003E+03
9	1.081E+03	1.076E+03	1.068E+03	1.054E+03	1.035E+03	1.022E+03	1.013E+03	1.006E+03	1.001E+03	9.975E+02	9.955E+02	9.945E+02
10	9.190E+02	9.160E+02	9.101E+02	9.009E+02	8.884E+02	8.795E+02	8.731E+02	8.684E+02	8.650E+02	8.627E+02	8.612E+02	8.605E+02
11	5.877E+02	5.867E+02	5.849E+02	5.821E+02	5.784E+02	5.756E+02	5.734E+02	5.718E+02	5.706E+02	5.697E+02	5.692E+02	5.689E+02
12	2.478E+02	2.475E+02	2.468E+02	2.458E+02	2.444E+02	2.433E+02	2.426E+02	2.420E+02	2.415E+02	2.412E+02	2.411E+02	2.410E+02

THERMAL FLUX

1	1.639E+02	1.638E+02	1.636E+02	1.633E+02	1.630E+02	1.627E+02	1.624E+02	1.621E+02	1.619E+02	1.617E+02	1.616E+02	1.615E+02
2	3.906E+02	3.902E+02	3.892E+02	3.880E+02	3.866E+02	3.852E+02	3.839E+02	3.828E+02	3.820E+02	3.813E+02	3.809E+02	3.807E+02
3	4.132E+02	4.122E+02	4.101E+02	4.070E+02	4.029E+02	3.998E+02	3.975E+02	3.957E+02	3.944E+02	3.935E+02	3.930E+02	3.927E+02
4	5.648E+02	5.632E+02	5.600E+02	5.554E+02	5.492E+02	5.446E+02	5.410E+02	5.383E+02	5.363E+02	5.349E+02	5.340E+02	5.336E+02
5	7.126E+02	7.103E+02	7.057E+02	6.991E+02	6.906E+02	6.839E+02	6.787E+02	6.746E+02	6.715E+02	6.693E+02	6.679E+02	6.673E+02
6	7.725E+02	7.706E+02	7.669E+02	7.619E+02	7.563E+02	7.508E+02	7.459E+02	7.418E+02	7.386E+02	7.363E+02	7.348E+02	7.341E+02
7	7.729E+02	7.716E+02	7.692E+02	7.662E+02	7.637E+02	7.599E+02	7.560E+02	7.525E+02	7.497E+02	7.476E+02	7.462E+02	7.455E+02
8	7.537E+02	7.524E+02	7.499E+02	7.467E+02	7.437E+02	7.399E+02	7.361E+02	7.328E+02	7.302E+02	7.282E+02	7.269E+02	7.263E+02
9	7.023E+02	7.009E+02	6.984E+02	6.949E+02	6.911E+02	6.872E+02	6.837E+02	6.807E+02	6.783E+02	6.766E+02	6.755E+02	6.749E+02
10	5.767E+02	5.756E+02	5.736E+02	5.707E+02	5.674E+02	5.643E+02	5.616E+02	5.595E+02	5.578E+02	5.566E+02	5.558E+02	5.554E+02
11	3.625E+02	3.621E+02	3.613E+02	3.603E+02	3.590E+02	3.578E+02	3.568E+02	3.560E+02	3.554E+02	3.549E+02	3.546E+02	3.545E+02
12	1.525E+02	1.524E+02	1.521E+02	1.517E+02	1.513E+02	1.508E+02	1.505E+02	1.502E+02	1.500E+02	1.498E+02	1.497E+02	1.497E+02

ELAPSED TIME(MIN) 54.35

APPENDIX C

EXAMPLE OF PERT SUBROUTINE

The following is a listing of the existing PERT subroutine with its auxiliary subroutine PCONT. It contains the following 3 options:

(i) Simple change in power calculation.

(ii) CANDU type booster restart after xenon build up calculation.

(iii) Xenon axial instability calculation.

```
      SUBROUTINE PERT(WORD)
C
      DIMENSION WORD(40)
      COMMON /DESC/
     1   ACC(10), AKEF(10), BCONC(10), TTFF(10), FLWT(10), RVOL(10)
     9 , NOSTR(10), CONC(10)
     2 , TITLE(15),  ACH, ASTPLS, AXFORM, BET, BLF, BURNUP, CAYT, CELLAR
     3 , EIGEN, HSX, IB, IBURN, ISUPER, IEQUIL, IFILE, IFILER, IPAGE
     4 , IPERT, IPOINT, IPRIN, ISAGE, LFLWT, LXENN, LCOOL, ITEST, IXTRA
     5 , KBOT, KFLUX, LRING, LTEMP, MIRI, NLIEB, NSTEPS, NTOT2, N3T
     6 , PERIOD, POWER, PRECIS, RATPFR, RDFORM, RESTME, RTIME, SUB
     7 , SUM, SVOL, THERMF, TIMLIM, UD, UGCC, URAN, XB, XDEC, XFACT1
     8 , XFACT2, XR, XT, YIELD, ANST, KDATA, MDATA, IDATA, IMAT, IMOVE
     9 , NAX, NCHAN, NCHANX, NSEGM, NRAD, NSTRE,NSGTOT,LUMPS
     9 , IFLOW, THERMK, PRDLIM, HEIGHT,TOTFLO, PDRUM, PPUMP, TPUMP
     9 , HPUMP, CEQXH, CEQXL, CEQDE, CEQA, CKOBS, CDHE, ICOOLH, SVMPUM
     9 , JS,JGR,JTP,JFR,INTOT,NRISE,NFEED,VOILIM,XFMN,PD,JDRY,XFMN2
     9 , PSTPAT,BOOST,IXEN,XTIM,XADT,XPOW,XSTIM,XCTOP,XCBOT,XSRAD,IXCON
      EQUIVALENCE(PSTPAT,PFSPOW)
C
C   PERTURBATION SUBROUTINE FOR SOME USER CONTROL
C
C     DATA FOR CONTROL OF PERT COMES VIA WORD HOLDING STRING
C   OF NUMBERS FOLLOWING FIRST CONTROL INTEGER(IPERT) IN PERT DATA BLOCK
C
C   THE FOLLOWING IS A LIST OF THE MAIN VARIABLES
C   LIKELY TO BE CHANGED BY USER
C
C     GENERAL CONTROL
C
C     ACC(1)  ACCURACY ON POWER DISTRIBUTION
C     ACC(8)  PRINT OUT SPECIFICATION
C     ......ETC.        FULL LIST IN PROGRAM.
C
C   TEST FOR TYPE OF EXISTING PERT CALC.
      GO TO (1,2,3,4),IPERT
C
    1 CONTINUE
      WRITE(3,1001)
 1001 FORMAT(/,10X,'POWER COEF. PERT CALCULATION',/)
C   SIMPLE POWER COEFICIENT TYPE CALC.
C   RECALCULATE NEW NORMALISED FLUX AND CHANNEL POWER
C     WORD(1) TYPE OF CONTROL 1=POISON 0=NONE
C     WORD(2) FRACTIONAL CHANGE IN CORE POWER
      IXCON=WORD(1)+.1
      POWER=POWER*WORD(2)
   10 CALL EDISEC
C   RECALCULATE COOLANT DISTRIBUTION (IF REQUIRED) FOR CROSS SECTIONS
      IF(LCOOL.LT.3) CALL HYDSEC
C   RECALCULATE CROSS SECTIONS
      CALL CROSEC
C   RECALCULATE FLUX DISTRIBUTION
C   TEST IF CONVERGED(ITEST=0),OR RUN OUT OF TIME (ITEST=-1)
      IF(ITEST)11,12,13
   13 CALL DIFSEC
C     ADJUST REACTIVITY IF NECESSARY
      CALL PCONT
      GO TO 10
   11 RETURN
   12 WRITE(3,101)
```

```
  101 FORMAT(/,10X,'CASE CONVERGED')
      RETURN
C
    2 CONTINUE
C   BOOSTER ROD (START UP) SIMULATION CALCULATION
C     BOOSTER RODS REQUIRED TO OVER-RIDE XENON BUILT UP DURING SHUTDOWN
C     POWER IS LIMITED TO MAXIMUM FUEL SEGMENT RATING SPECIFIED
C   CALCULATION REQUIRES TIME DEPENDENT XENON
C   SET SWITCH FOR XENON CALC.
      WRITE(3,200)WORD(1)
  200 FORMAT(//,10X,'BOOSTER ROD RESTART CALCULATION',/,
     110X,'SHUT DOWN TIME (MINS)',F10.1)
      IXEN=1
C   SET XENON TIME (XTIM) TO ZERO
      XTIM=0.
C     SET XENON ADVANCE TIME TO END OF SHUTDOWN
      XADT=WORD(1)
C     SET XENON SMALL TIME STEP INTERVAL
      XSTIM=WORD(3)
C   SET SHUTDOWN POWER FRACTION
      XPOW=0.
C   PUT IN BOOSTER RODS BY MAKING THE RADIAL ZONE ELEMENT
C   CROSS SECTIONS GIVEN IN THE INPUT EFFECTIVE
      IXTRA=1
C
C   SET POWER TO EXPECTED MAXIMUM STARTUP POWER
C   AND STORE REFERENCE FULL POWER
      REFPOW=POWER
      POWER=POWER*WORD(6)
C     SET TYPE OF CONTROL IXCON=2 FOR BOOSTER
      IXCON=2
      JITT=0
C  MUST CALL CROSEC FIRST TO ESTABLISH EQUIL. POWER XENON CONCENTRATION
   25 CONTINUE
C   SET UP NEW CROSS SECTIONS FOR NEW XENON CONDITION
      CALL CROSEC
C   TEST IF CONVERGED(ITEST=0),OR RUN OUT OF TIME (ITEST=-1)
      IF(ITEST)21,22,23
C   CALCULATE NEW FLUX DISTRIBUTION
   23 CALL DIFSEC
C   CALCULATE NEW POWER DISTRIBUTION FOR SET CORE POWER
C     WILL RETURN PEAK FUEL SEGMENT RATING (W/GRM)IN PFSPOW
      CALL EDISEC
      JITT=JITT+1
      IF(JITT.EQ.1)GO TO 25
C   CALCULATE NEW MAXIMUM CORE POWER TO SAFETY PEAK PEAK SEGMENT
      POWER=POWER*WORD(4)/PFSPOW
      IF(POWER.GT.REFPOW)POWER=REFPOW
      WRITE(3,2101)POWER
 2101 FORMAT(10X,'CORE POWER ',1PE10.3)
      IF(ITEST.EQ.0)GO TO 25
C     RE-NORMALISES FLUXES AND POWERS
      CALL EDISEC
C   CALL CONTROL SUBROUTINE TO INCREASE OR REDUCE BOOSTERS
      CALL PCONT
      GO TO 25
   21 RETURN
   22 CONTINUE
C   STEP CONVERGED
      VAB=POWER/REFPOW*100.
```

```
      WRITE(3,201)XTIM,VAB,BOOST
  201 FORMAT(/,10X,'STEP CONVERGED FOR TIME (MINS)',F10.1,
     1'  REACTOR POWER (% FULL POWER)',F10.2,'BOOSTERS',1PE10.3)
C   BEFORE INCREASING TIME CHECK IF FAR ENOUGH
      IF(POWER.GE.0.95*REFPOW) GO TO 26
      IF(XTIM.GE.WORD(5))GO TO 27
C   INCREASE XENON ADVANCE TIME
C     NB XENON ADVANCE TIME(XADT)MUST BE SET EACH TIME
      XADT=WORD(2)
C     SWITCH OFF PRINTING AND CONVERGENCE TEST
      IPRIN=0
      ITEST=1
      GO TO 25
   26 WRITE(3,202)
  202 FORMAT(/,10X,'BACK TO FULL POWER')
      RETURN
   27 WRITE(3,203)
  203 FORMAT(10X,'TAKING TOO LONG TO GET TO FULL POWER')
      RETURN
C
C     SIMPLE TEST OF AXIAL STABILITY WITH XENON
C     PERTURBATION, WITH NO ROD CONTROL
C
C     SEE SECTION 2 FOR XENON REMARKS
    3 CONTINUE
      WRITE(3,1003)
 1003 FORMAT(/,10X,'XENON STABILITY PERT CALCULATION',/)
      IXEN=1
      XTIM=0.
      XPOW=1.0
      XSTIM=WORD(3)
C     SET ADVANCE TIME TO 0. TO GET XENON NOW
      XADT=0.
C     FORCE SOME SUDDEN FRACTIONAL CHANGE TO XENON CONCENTRATION IN ALL
C     BOTTOM FUEL SEGMENTS IN INNER RADIAL ZONE
      XCBOT=WORD(4)
C     AND INVERSE TO TOP
      XCTOP=1.0/WORD(4)
C     SET TYPE OF CONTROL
      IXCON=WORD(1)
C
C     CALCULATION SECTION AS AVOVE
   35 CONTINUE
      CALL CROSEC
      IF(ITEST)21,32,33
   33 CALL DIFSEC
      CALL EDISEC
      IF(LCOOL.LT.3) CALL HYDSEC
      CALL PCONT
      GO TO 35
   32 WRITE(3,301)XTIM
  301 FORMAT(/,10X,'STEP CONVERGED FOR TIME (MINS.)',F10.1)
      IF(XTIM.GT.WORD(5)) GO TO 37
      XADT=WORD(2)
      IPRIN=0
      ITEST=1
      GO TO 35
   37 WRITE(3,302)
  302 FORMAT(/,10X,'END OF PERT CALCULATION')
      RETURN
```

```
    4 CONTINUE
      RETURN
      END
      SUBROUTINE PCONT
      COMMON /DESC/
     1   ACC(10), AKEF(10), BCONC(10), TTFF(10), FLWT(10), RVOL(10)
     9 , NOSTR(10), CONC(10)
     2 , TITLE(15),  ACH, ASTPLS, AXFORM, BET, BLF, BURNUP, CAYT, CELLAR
     3 , EIGEN, HSX, IB, IBURN, ISUPER, IEQUIL, IFILE, IFILER, IPAGE
     4 , IPERT, IPOINT, IPRIN, ISAGE, LFLWT, LXENN, LCOOL, ITEST, IXTRA
     5 , KBOT, KFLUX, LRING, LTEMP, MIRI, NLIEB, NSTEPS, NTOT2, N3T
     6 , PERIOD, POWER, PRECIS, RATPFR, RDFORM, RESTME, RTIME, SUB
     7 , SUM, SVOL, THERMF, TIMLIM, UD, UGCC, URAN, XB, XDEC, XFACT1
     8 , XFACT2, XR, XT, YIELD, ANST, KDATA, MDATA, IDATA, IMAT, IMOVE
     9 , NAX, NCHAN, NCHANX, NSEGM, NRAD, NSTRE,NSGTOT,LUMPS
     9 , IFLOW, THERMK, PRDLIM, HEIGHT,TOTFLO, PDRUM, PPUMP, TPUMP
     9 , HPUMP, CEQXH, CEQXL, CEQDE, CEQA, CKOBS, CDHE, ICOOLH, SVMPUM
     9 , JS,JGR,JTP,JFR,INTOT,NRISE,NFEED,VOILIM,XFMN,PD,JDRY,XFMN2
     9 , PSTPAT,BOOST,IXEN,XTIM,XADT,XPOW,XSTIM,XCTOP,XCBOT,XSRAD,IXCON
      DATA ITER/0/
C
C
C   SUBROUTINE FOR ADJUSTING POISON (IXCON=1) OR
C   BOOSTER RODS (IXCON=2) TO ACHIEVE CRITICALITY
C
      WRITE(3,100)EIGEN,SUB,AKEF(4)
  100 FORMAT(/,10X,'CURRENT KEFF.',1PE13.6,
     1'  GOG FINAL ACCURACY',E11.4,'  PREVIOUS KEFF.',E13.6)
      DIF=EIGEN-AKEF(4)
C     STORE K IN PREVIOUS K BOX
      AKEF(4)=EIGEN
C
C   CHECK TYPE (IXCON=0 NO CONTROL)
      IF(IXCON.EQ.0)GO TO 30
C   FIND REACTIVITY ERROR, DESIRED K IN AKEF(1)
      DELK=EIGEN - AKEF(1)
      IF(IXCON.GT.1) GO TO 20
C
C   POISON CONTROL
C   CALCULATE NEW POISON CONCENTRATION
C   POISON CONCENTRATION STORED IN BCONC(1)
C   RATE OF CHANGE OF K WITH POISON IN BCONC(5)
      BLAST=BCONC(1)
      BCONC(1)=BCONC(1)+DELK/BCONC(5)
      WRITE(3,103) BCONC(1),BLAST
      BCONC(2)=BLAST
  103 FORMAT(/,10X,'POISON CONCENTRATION ESTIMATE',1PE10.3,
     1'  PREVIOUS ESTIMATE',E10.3)
C   RESET IF CONCENTRATION ESTIMATE FALLS BELOW LIMIT
C   LOWEST ALLOWABLE POISON CONC. STORE IN BCONC(4)
      IF(BCONC(1).GT.BCONC(4)) RETURN
C   CHECK IF TRUE RESULT WILL BE BELOW LIMIT
      ITER=ITER+1
      IF(ITER.LT.3)GO TO 13
C   CRITICALITY NOT POSSIBLE
      ITEST=-1
      WRITE(3,102)
  102 FORMAT(/10X,'CASE WONT CONVERGE, REACTIVITY TOO LOW')
      RETURN
   13 BCONC(1)=BCONC(4)
```

```
      WRITE(3,104) BCONC(4)
  104 FORMAT(10X,'RESET TO',1PE10.3)
      RETURN
C
C
   20 CONTINUE
C
C    REACTIVITY CONTROL WITH SIMPLE TYPE BOOSTER RODS
C    CONTROL IS MADE BY INCREASING OR DECREASING BOOST.
C    BOOST BEING THE MULTIPLICATION FACTOR USED TO
C    ADJUST THE FIXED RADIAL ZONE ELEMENT CROSS SECTIONS
C    GIVEN IN INPUT IE. IXTRA EQ.TO -1 OR 1
C
C    TEST IF NOT FIRST ITERATION
      IF(ITER.NE.0) GO TO 21
C
C    CALCULATED ARBITARY ADJUSTMENT
      BOOST=1.1
      IF(DELK.GT.0.)BOOST=0.9
C    RATE OF CHANGE OF K WITH BOOSTER IS RCBOOS
      RCBOOS=DELK/(1.0-BOOST)
      ITER=ITER+1
      PBOOST=1.0
      GO TO 22
   21 CONTINUE
C    CALCULATE RATE OF CHANGE OF K ON PREVIOUS CALCS.
C    DONT CHANGE IF CHANGE IN K TOO SMALL.
      RCBOOS=DIF/(BOOST-PBOOST)
C     STORE AND ESTIMATE ADJUSTMENT
   23 PBOOST=BOOST
      BOOST=BOOST-DELK/RCBOOS
      IF(BOOST.LT.0.)BOOST=0.
   22 CONTINUE
      WRITE(3,201)BOOST,PBOOST
  201 FORMAT(/,10X,'BOOSTER CONTROL ESTIMATE',1PE10.3,
     1'  PREVIOUS ESTIMATE',E10.3)
      RETURN
   30 BCONC(2)=BCONC(1)
      RETURN
      END
```