BURNUP -- A CODE TO CALCULATE BURNUP OF EBR-II FUEL

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ABSTRACT

The EBR-II BURNUP code was developed to provide a record of subassembly disposition and to calculate fuel burnup and irradiation time to a certain burnup limit for each subassembly in the reactor during a given run. A historical record is kept of each partially irradiated subassembly that has not yet been transferred from the reactor storage basket. The nuclear center of the reactor is found from the calibrated worths of the control rods for each run. This report explains the code, its purposes, how it functions, and its input and output.

I. INTRODUCTION

The BURNUP code was written originally to provide reactor-operations personnel with data related to burnup in subassemblies in EBR-II, but is now providing data for other EBR-II groups as well. Its main function is to routinely calculate incremental and total burnup for each subassembly in the first eight rows of the reactor. Periodically, it is used to calculate the burnup in the remaining blanket subassemblies in the reactor. Burnup is defined as the ratio of the number of heavy atoms fissioned to the number of heavy atoms available when the subassembly first was inserted into the reactor. It is usually expressed in atom percent (at. %) and is directly related to swelling of the fuel. A burnup limit based on this relationship has been established for EBR-II fuel. When this limit is reached or approached so closely that the subassembly should not be left in the reactor for another run, the subassembly containing the fuel is moved.

The BURNUP code was written originally for the IBM 1620 computer. It was run in two passes on that machine, first to calculate the fuel burnup and then to arrange the subassemblies in descending order according to the

number of megawatt-days left until the burnup limit will be reached. All input data were punched on cards, and all data relating to subassembly changes for a given core loading were inserted into the load deck by hand.

When the larger CDC 1604 computer was added to the facilities of the National Reactor Testing Station (NRTS) in January 1967, it enabled the BURNUP code to be run in one pass and the output to be printed as well as punched. Later in the year, a subroutine was added to store historical data on magnetic tape for subassemblies which previously had been in the reactor but had not attained the specified burnup limit. The method for changing the data for the core loading of a new run was modified so that hand insertion was no longer required. By then, the code output was also being used by the group in charge of irradiation of subassemblies, by the Fuel Cycle Facility in its surveillance program, by the special materials group in accounting for fuel, and for comparing calculated data with measurements made by the analytical section. When flux-wire measurements in run 29C revealed the presence of core asymmetry, subroutines were added to the code to determine a "nuclear center" for each run, based on the calibrated worths of control rods.

In January 1969, the IBM 360 computer was put into operation at the NRTS, and the BURNUP code subsequently was converted to the new machine. Recent modifications of the code have provided the operations group with a list of subassemblies and their burnups at the beginning of a run, enabling those concerned to check the current core loading just before startup. To ensure that all subassemblies will be able to stay in for the whole reactor run, an attempt is made to run the code for the projected length of the run as soon after the reactor is loaded as possible.

II. DESCRIPTION OF CODE

In its present form, the BURNUP code consists of a main program and seven subroutines, comprising approximately 1500 cards. Written in FORTRAN IV language, the program requires 124 K bytes of core for compilation and 130 K bytes for execution on the IBM 360/75 computer. The main program handles all input except the historical and loading-change data. Since

large amounts of information are transferred among the program subroutines, nearly all variables are accessed in blocks of COMMON storage.

Input, which is described fully in Section III, is divided into six categories:

1. Run parameters such as run number, megawatt-day length, input/output options, etc.
2. Axial maximum-to-average fission and capture ratios and reactivity worths of materials.
3. Coefficients and calibrated worths of control rods for calculating the nuclear center and fission distribution.
4. Load deck consisting of two cards for each subassembly in the first eight rows of the reactor. These data are punched after each run of the code, for use as input for the next run.
5. Historical record, on magnetic tape, of subassemblies available for reinstallation.
6. Loading changes for the current run.

The subroutines used in the BURNUP code are discussed in the remainder of this section.

A. HISTREAD

This subroutine handles the input, output, and updating of data for subassemblies that have been in the reactor but have not attained the maximum allowed burnup. Since these subassemblies may be removed and reinstalled, the data for each are stored on magnetic tape (FORTRAN Unit 1). It is then possible to retrieve the data for a subassembly and continue computing its additional burnup for a later reactor run. The tape record contains subassembly number, type, installation date, fuel weights, and burnup when the subassembly was removed from the core as well as the initial fuel weights. A subassembly appears only once in the history, regardless of the number of times it has been in the reactor. Coding for updating or correcting historical data is discussed in Section III.E. The updated history is printed out for each BURNUP run.

Changes in the core loading for the current run then are read. If data for a replacement subassembly are stored as history, the historical data are searched until the subassembly number of interest is found. The value of KC (the number of times the subassembly has been in the reactor) on the change card must be one greater than that shown on the history print-out. If not, a warning statement is printed, and the problem aborts. The KC value then is incremented by one, and data for this subassembly are transferred by the computer into the array being built for the current core loading. The weights of irradiated fuel and the burnup values which were computed for the last time the subassembly was in the reactor become the initial values used for the current run. The BURNUP code computes incremental changes in weights and burnup, building on the previous step.

The subassembly being replaced may be either moved to a new core position for the current run or removed from the reactor. In the latter case, data for the subassembly at the end of the last run (both initial and final values) are transferred to the history. If the subassembly is to be left in the reactor for the current run, all its data are stored in a temporary array and then reinserted in the proper position.

As each subassembly becomes history, its data are printed out in a list of subassemblies replaced for the current run. Subassemblies switched from one core position to another between runs do not appear in the list of those replaced. This method of recording the past activity of each subassembly ensures that the correct weights and burnup will be available when the subassembly is installed again. Historical data are then written on Unit 2.

B. WORTH

This subroutine calculates the change in reactivity resulting from the changes in core loading. The subroutine is called each time a subassembly is removed or added. By distinguishing the type and position of the subassembly in question, the correct reactivity worth of the material is found. The worth (in lh/g) is multiplied by the material weight to give the reactivity that is due to the subassembly. Weights of sodium and

stainless steel in standard driver, control-rod, safety-rod, and blanket subassemblies are provided in a table. Weights of sodium and stainless steel in each experimental subassembly are specified separately.

The reactivity resulting from each change is added to or subtracted from a running total. After all changes have been made, the resultant total change in reactivity for the loading is printed.

If one desires to make a preliminary calculation of the reactivity change resulting from a new core loading but has not yet decided on the number of the new subassemblies to be installed, he may read in the last load deck and specify only type and position on the replacement cards. In this case, no history changes or burnup calculations are made.

C. FLXCTR

This subroutine determines the nuclear center of the reactor from calibrated worths of control rods by an iterative technique. The worth of each control rod is determined at the start of each reactor run by the reactor operating crew. The even- and odd-numbered control rods in each sector may not have the same measured worths. This implies that there is flux asymmetry in the core owing to nonsymmetrical placement of structural experiments and experiments with a nonstandard fuel content. The assumption is made that the curve of radial fission-rate distribution which is obtained using simple radial symmetry such as would be calculated with MACH-1¹ shifts from the physical center of the reactor to a new center but does not change in shape. Once the nuclear center has been determined, the distance from the center to any point in the reactor is calculated by a simple transformation of coordinates. The fission rate is then the fission rate calculated for that distance.

The technique used to determine the flux center is to assume that the measured worth, ρ_1 , of a control rod is equal to a constant A multiplied by some function I of r' , the distance from the flux center to the control rod, or

$$\rho_1' = AI(r_1'). \quad (1)$$

The best nuclear center, based on N control rods, is that point at which

$$\text{SUMSQ} = \sum_{i=2}^N [\rho_i - AI(r'_i)]^2 = \text{minimum.} \quad (2)$$

A is given by

$$A = \frac{\sum_{i=2}^N \rho_i}{\sum_{i=2}^N I(r'_i)}. \quad (3)$$

The distance r' from the nuclear center at $(R,0)$ to a point at (r, θ) measured from the geometric center is given by

$$r' = [R^2 + r^2 - 2rR \cos(\theta-0)]^{1/2}. \quad (4)$$

The term R is the distance from the geometric center to the nuclear center at an angle 0 to the line dividing sectors C and D, and r is the distance from the geometric center to any other point between sectors C and D.

It has been found that a fourth-order fit of the MACH-1 $\Delta k/k$ -cc curve for ^{235}U works well for the worth function $I(r)$.

The measured control-rod worths are corrected for any differences in worth due to differences in the weight of ^{235}U resulting from burnup. Thus, the corrected control-rod worth (ρ'_i) is given by

$$\rho'_i = \rho_i (U_{\text{max}}/U_i), \quad (5)$$

where U_{max} is the weight of ^{235}U in the heaviest rod and U_i is the weight of the rod whose worth is being determined.

The angle of the nuclear center relative to the line dividing sectors C and D of the reactor, and going in a counter-clockwise direction, is first determined by symmetry, i.e., a radial iteration to find the line through the nuclear center that will make the sum of the control-rod worths on one side of the line equal to the sum on the other side. The nuclear center will be on the side with the greater total control-rod worth. The program is set to sweep around the reactor core, checking the sums of the worths on both sides of lines at angles of multiples of 30° . The sweep

stops when the center is passed, and preliminary limits PHI1 and PHI2 for THETA, the value of the flux center angle, are set.

The first radial iteration tries values of R_2 , the distance between the nuclear and geometric centers, at 0.2-cm intervals between 0 and 3 cm.

For each radial increment, the distance from each control-rod center to the estimated nuclear center is calculated using the law of cosines (Eq. 4). The worth of each control rod at this distance then is calculated from a fourth-order polynomial based on the $\Delta k/k$ -cc curve. The sums of the calibrated control-rod worths (SUMT) and of the calculated control-rod worths (SUMP) are computed. A new constant, A, then is obtained by

$$A = \text{SUMT}/\text{SUMP}. \quad (6)$$

From this, a new value of SUMSQ (Eq. 2) is obtained. After all increments have been considered, the distance at which the minimum value of SUMSQ occurs is determined. A new R_2 then is calculated by a second-order extrapolation considering the point at which SUMSQ is a minimum and at points on either side.

The mathematical method for this extrapolation considers a second-order equation of the form $y = ax^2 + bx + c$.

The minimum value of y will occur when $dy/dx = 0$, or $2ax + b = 0$, and when $x = -b/2a$.

If the coordinates of three points are known, the values of b and a may be calculated. Using these values, the minimum value ($x_{\min}$) of x is determined by linear extrapolation.

Angular iterations are made in the same manner as above, but for increments between PHI1 and PHI2. When a new angle is found, radial iterations are then made for $R_2 \pm 0.5$ cm at 0.1-cm intervals. It has been found that convergence on THETA and R_2 occur within a maximum of six trials, yielding the nuclear center. The results then are printed (Fig. 1), giving the distance (in centimeters) and angle (in degrees and radians) of the nuclear center in relation to the geometric center.

The accuracy of this calculational method has been demonstrated by the fact that the flux distribution with the nuclear center calculated for run 29D was the same as the distribution measured by flux wires during that run.

Control rods 2 through 12 are used to determine which sector contains the nuclear center. Rod 1, in position 5D1, is a stainless steel dummy. If other dummy rods are present, an average value for their worth must be read into the code. The average worth for an odd-numbered dummy rod is the average of the adjacent odd-numbered rods on either side of it. An even-numbered rod is treated analogously.

Although experiments have not been made to determine the nuclear center of the reactor, we believe that our calculations predict the sector in which the center would be found. L. Miller, using the DOT code,² predicts a larger shift of the nuclear center than our results indicate and also predicts many local perturbations and global asymmetries which we cannot obtain with this code.

D. FIS RTE

This subroutine uses the values of R and THETA calculated in FLXCTR to compute the relative ^{235}U and ^{238}U fission-rate distributions and the relative ^{238}U capture rate for each subassembly. It has been found that the relative distributions follow a fourth-order polynomial. Thus, by reading in five coefficients--A, B, C, D, E--obtained from a polynomial curve-fitting routine, the relative fission rate F_{av} at any distance r from the geometric center of the subassembly to the nuclear center of the reactor is determined by

$$F_{av}(r) = A + Br + Cr^2 + Dr^3 + Er^4. \quad (7)$$

The coefficients need be recalculated only when a major change occurs in the reactor configuration. This would require reevaluation of the fission-rate distribution, using the results of both the measurements and the calculations.

The ^{235}U fission-rate coefficients and ^{238}U capture coefficients are those calculated directly. For the fourth-order ^{238}U fission rate, data for the first six rows of the reactor are fitted separately from those for rows 7 and 8. The ^{238}U fission-rate coefficients are multiplied by the $^{238}\text{U}/^{235}\text{U}$ central-fission-rate ratio, currently 0.069, obtained from MACH-1 results. Before being read into the code, the coefficients for the ^{238}U distribution in rows 7 and 8 must be multiplied by the first coefficient from the six-row fit.

FLUX CENTER CALCULATION RUN NUMBER 37 A

THE FLUX CENTER IS 1.244 CM FROM THE CORE CENTER
AT AN ANGLE OF 283.05 DEGREES OR 4.9401 RADIAN

ROD NO.	MEASURED WORTH	MASS CORRECTED WORTH	CALCULATED WORTH	DIFFERENCE
2	178.95	178.95	175.87	3.08
3	149.50	150.25	153.56	-3.30
4	175.81	177.86	178.83	-0.97
5	151.71	153.47	152.32	1.14
6	174.24	176.36	171.95	4.41
7	140.83	142.18	141.84	0.35
8	154.46	155.17	162.41	-7.24
9	132.60	133.26	135.45	-2.19
10	174.24	175.40	159.63	15.76
11	131.17	131.64	135.31	-3.67
12	152.90	154.65	161.95	-7.30

THE SUM OF THE DIFFERENCE SQUARED IS 414.67
OR 6.14 PER ROD DIFFERENCE

Fig. 1. Printout of Calculation of Nuclear Center by FLXCTR Subroutine

The radial maximum-to-average fission-rate ratio also is calculated for each subassembly position in the following manner. For all subassemblies in the first six rows, this number is based on the ^{235}U fission-rate distribution. For each of these subassemblies, 2.83 cm is subtracted from r , the distance from the geometric center of the subassembly to the nuclear center of the reactor, thereby giving the location, r_c , of the pin closest to the nuclear center and therefore receiving the maximum burnup. The fission rate $F_{\text{max}}(r_c)$ at this distance is computed, and the radial max/avg fission-rate ratio $RX(r)$ is calculated from the equation

$$RX(r) = F_{\text{max}}(r_c) \text{ XAV5} / F(r), \quad (8)$$

where XAV5 is the axial max/avg fission-rate ratio for ^{235}U in the core height of 13.5 in., as obtained from measured axial distribution.

The radial max/avg fission-rate ratio for a blanket subassembly is obtained with a similar calculation over a length of 55 in., with the ^{238}U distribution. For these subassemblies, 2.60 cm is subtracted from r . The max/avg fission-rate ratios for experimental subassemblies in rows 7 and 8 are based on ^{235}U data for these positions.

If requested, the relative fission and capture rates and the max/avg fission-rate ratios for each subassembly position are printed.

The code also has the option of reading in individual fission rates and capture rates for any or all subassembly positions.

E. FISCOMP

To compute fuel burnup, the absolute number of fissions per gram per megawatt day for each isotope must be obtained. First, the total number of atoms of ^{235}U fissioned per megawatt-day, TFMD, is computed by

$$\text{TFMD} = \text{RMEV} \times 0.005 \times 0.26948 \times 10^{22} \times \text{POWF}, \quad (9)$$

where RMEV is the number of millions of electron volts per fission of a ^{235}U atom. This number is 200 MeV, the sum of 192 MeV due to beta contribution, fissioning, capture, etc., and 8 MeV due to neutron-gamma reactions. POWF is the fraction of the total 50-MWt power which is produced in the first eight rows of the reactor. A burnup calculation on the reactor with depleted uranium in rows 7-16 has shown this number to be 0.965. The constant 0.005 is a normalization factor, and the constant 0.26948×10^{22}

is the number of atoms of ^{235}U fissioned per megawatt-day at 200 MeV per fission.

The central fission rate C is determined from the equation

$$\text{TFMD} = C \times \sum_{j=1}^{169} \sum_{i=1}^3 (F_{i,j}) \times (G_{i,j}) \times (R_i)/M_i, \quad (10)$$

where $F_{i,j}$ = relative fission rate of isotope i in position j (the $F_{238,i}$ values are already normalized to the central $^{238}\text{U}/^{235}\text{U}$ fission ratio),

$G_{i,j}$ = weight in grams of isotope i in position j ,

R_i = normalization factor for isotope i relative to ^{235}U (equal to 1 for ^{235}U and ^{238}U and 1.21 for ^{239}Pu), and

M_i = axial max/avg fission-rate ratio for isotope i .

A correction factor also is put in to account for length of control-rod insertion, or banking. During operation, all control rods except the controlling rod are banked at the same level. The fraction of the total power produced in the control rods due to partial insertion, RV , is given by the polynomial equation

$$RV = 0.376127 + 0.046642(P) + 0.0028145(P^2) - 0.00021167(P^3) \quad (11)$$

where P is the control-rod bank position in inches.

C is computed on a row-by-row basis to retain the fraction of the total fissions due to each row (Fig. 2). At the same time, the weight of each isotope at the start of the run is logged. By using the number of grams fissioned in a subassembly, the power due to fissioning (based on 200 MeV/fission) is made to equal $\text{PWF} \times 50 \text{ MWt}$.

F. BUCOMP

This subroutine computes the change in weight of fissionable material and the atom-percent burnup incurred by each subassembly during a reactor run of specified length in megawatt-days. These are incremental changes.

The total initial weight* of fissionable material in a subassembly is obtained by summing the weights of the fuel isotopes present:

$$\text{FUELO} = G50 + G80 + G90, \quad (12)$$

* All material weights are in grams throughout this report.

RUN 37 A (1200. MWD) 27474. TO 28674. MWD

TOTAL FISSIONS/MWD= 0.26053E 22 AT 200. MFV/FISSION

	CORE	BLANKET	PCT OF TOTAL
235	0.21695E 22	0.20138E 20	84.05
238	0.10721E 21	0.20640E 21	12.04
239	0.87198E 20	0.14844E 20	3.92
TOT	0.23639E 22	0.24139E 21	
PCT	0.90735E 02	0.92650E 01	

MAX FISSION RATES FISSION/(GM-MWD)

235	0.15488E 17
238	0.10432E 16
239	0.18740E 17

TOTAL LOADING

235	0.20228E 06	0.68766E 04
238	0.17255E 06	0.31889E 07
239	0.67820E 04	0.42789E 04

ROW	PCT.
1	1.56
2	4.52
3	15.63
4	18.51
5	22.12
6	26.40
7	4.82
8	3.13
9	0.0
10	0.0
11	0.0
12	0.0
13	0.0
14	0.0
15	0.0
16	0.0

PCT TOTAL POWER 96.68

KW PRODUCED IN DRIVERS 43111.7 KW PRODUCED IN EXPERIMENTS 5228.6
TOTAL POWER 48340.2 KW

Fig. 2. Printout of Calculation of Central Fission Rate and
Fuel Loading by FISCMP Subroutine

where G50, G80, and G90 are the weights of ^{235}U , ^{238}U , and ^{239}Pu before the subassembly was ever irradiated.

The weight of ^{239}Pu formed by ^{238}U capture, CT9, is

$$\text{CT9} = \text{G8} \times \text{C8} \times \text{S} \times \text{CAPC8} / (0.2519 \times 10^{22} \times \text{XACP8}), \quad (13)$$

where G8 = weight of ^{238}U at start of run,
 C8 = relative radial ^{238}U capture rate,
 S = central ^{235}U fission/g rate over the number of megawatt-days (AMWD) the run lasted,
 CAPC8 = (captures/g ^{238}U) / (fission/g ^{235}U),
 0.2519×10^{22} = equivalent atoms of ^{239}Pu per g ^{235}U , and
 XACP8 = ^{238}U axial max/avg capture ratio in EBR-II core.

The weight of ^{239}Pu fissioned, FT9, is

$$\text{FT9} = (\text{G9} + \text{CT9}/2) \times 1.21 \times \text{F5} \times \text{S} / (\text{XAV5} \times 0.2519 \times 10^{22}), \quad (14)$$

where G9 = weight of ^{239}U at start of run,
 1.21 = factor expressing ^{239}Pu fission rates in terms of the ^{235}U fission rate,
 F5 = relative radial ^{235}U fission rate, and
 XAV5 = ^{235}U axial max/avg fission ratio for the row.

Thus, the resultant weight of ^{239}Pu at the end of run G9 is

$$\text{G9}' = \text{G9} + \text{CT9} - \text{FT9}. \quad (15)$$

A similar calculation is made to compute the number of atoms of ^{235}U fissioned, FT5, as follows:

$$\text{FT5} = \text{S} \times \text{G5} \times \text{F5} / \text{XAV5}, \quad (16)$$

where G5 is the weight of ^{235}U at start of run. Some ^{236}U may be formed as a result of ^{235}U capture. The weight of this isotope at the end of the run, G6', is calculated using the following equations:

$$\text{CT5} = \text{CAPC5} \times \text{FT5} / 0.2551 \times 10^{22}, \text{ and} \quad (17)$$

$$\text{G6}' = \text{G6} + \text{CT5}, \quad (18)$$

where

CAPC5 = (captures/g ^{235}U) / (fissions/g ^{235}U), and
 0.2551×10^{22} = equivalent atoms of ^{236}U per g of ^{235}U .

The new weight of ^{235}U is thus

$$G5' = G5 - (FT5 + CT5 \times 0.2551 \times 10^{22}) / (0.2562 \times 10^{22}), \quad (19)$$

where

$$(0.2562 \times 10^{22} \text{ atoms } ^{235}\text{U} / \text{g } ^{235}\text{U}) = (0.2696 \times 10^{22} \text{ atoms } ^{235}\text{U} / \text{MWd}) / (1.050 \text{ g } ^{235}\text{U} / \text{MWd}).$$

The weight of ^{238}U at the end of the run, $G8'$, is calculated using the following two equations:

$$FT8 = S \times F8 \times G8 / \text{XAV8}, \text{ and} \quad (20)$$

$$G8' = G8 - (FT8 + CT9 \times 0.2519 \times 10^{22}) / (0.2530 \times 10^{22}), \quad (21)$$

where

$$\begin{aligned} F8 &= \text{relative fission rate of } ^{238}\text{U}, \\ \text{XAV8} &= ^{238}\text{U axial max/avg fission ratio, and} \\ 0.2530 \times 10^{22} &= \text{equivalent atoms of } ^{238}\text{U per g of } ^{235}\text{U}. \end{aligned}$$

The fission products resulting from burning the fissionable material are labeled "fissium" and are added to that already present. The resultant fissium is thus

$$FZ = \text{FUELO} + G60 + FZ0 - (G5' + G8' + G6' + G9'), \quad (22)$$

where $FZ0$ is weight of fissium when the subassembly was last inserted into the reactor.

Thus, the incremental average burnup, BUL , is

$$BUL = [FT5 / (0.2562 \times 10^{22}) + FT8 / (0.2530 \times 10^{22}) + FT9] / \text{FUELO} \quad (23)$$

and the average burnup at the end of this run, BU' , is

$$BU' = BU + BUL \times 100, \quad (24)$$

where BU is the at. % burnup on the subassembly prior to the run.

The incremental maximum burnup is

$$\text{DELB} = +(BU' - BU) \text{ RX}, \quad (25)$$

and the total maximum burnup is

$$BX' = BX + \text{DELB}, \quad (26)$$

where

$$\begin{aligned} \text{RX} &= \text{radial max/avg fission-rate ratio (from Eq. 8), and} \\ \text{BX} &= \text{previous maximum burnup, in at. \%}. \end{aligned}$$

Note that the maximum burnup is correct for a subassembly only if it has not been rotated (i.e., placed in a different sector). Otherwise, the same pin would not have continually received the maximum irradiation.

The number of megawatt-days left (DL) until the maximum burnup is equal to the burnup limit is

$$DL = (AL - BX) (AMWD/DELB), \quad (27)$$

where AL is the maximum burnup limit, and AMWD is the length of the current run.

This procedure is followed for each subassembly in the reactor loading. These data are printed (Fig. 3) and also punched on cards, the punched deck serving as input for the next run. The subassemblies then are listed in order of increasing megawatt-days left (Fig. 4). The "residence megawatt-days" also is computed. This is the length of time (TD) a similar but unirradiated subassembly may remain in each location and is given by

$$TD = AMWD (AL/DELB). \quad (28)$$

G. CHART

This subroutine lists the average and maximum burnup on each subassembly, beginning with the center subassembly, prior to startup for the current run. Subassemblies are put in ascending numerical order by means of an exchange sort³ and are listed by type and position. These lists (Figs. 5 and 6), which are needed for operations reports, provide a concise picture of the loading as seen by the BURNUP code.

III. INPUT

As mentioned at the beginning of Section II, the input data are divided into six categories. Figure 7 shows sample input. The data for these six categories are identified here.

A. Run Parameters

The first three cards contain data pertinent to the current (this) run.

RUN 37 A (1200. MWD) 27474. TO 28674. MWD

TV	NO	POS	235	238	236	239	FZ	B-AV	B-MK	MWD	MWDL	DATE	LIMIT	INST
1	2102	10101	2863.8	2651.0	12.4	3.0	332.	0.7148	0.8047	2400.	1340.	63069	1.250	1
KW =		780.3	2909.0	2656.3	4.7	0.0	292.	0.0	0.0					
1	2197	20101	2811.3	2631.6	15.9	4.5	384.	1.0494	1.1962	3602.	317.	53169	1.300	1
KW =		753.3	2877.0	2639.3	4.7	0.0	326.	0.0	0.0					
1	9057	20201	-0.0	-0.0	0.0	0.0	-0.	0.0	0.0	5910.	84090.	32369	9.990	1
KW =		0.0	0.0	0.0	0.0	0.0	0.	0.0	0.0					
1	2188	20301	2786.2	2631.1	19.3	5.8	388.	1.3737	1.5674	4734.	255.	42469	1.650	1
KW =		744.8	2872.0	2641.3	4.7	0.0	312.	0.0	0.0					
1	9021	20401	-0.0	-0.0	0.0	0.0	-0.	0.0	0.0	14857.	75143.	41067	9.900	3
KW =		0.0	0.0	0.0	0.0	0.0	0.	0.0	0.0					
1	2148	20501	2823.4	2637.2	13.5	3.5	372.	0.8264	0.9375	2802.	1094.	53169	1.300	2
KW =		762.0	2852.6	2640.7	8.5	1.5	346.	0.3571	0.4051					
1	9000	20601	-0.0	-0.0	0.0	0.0	-0.	-0.0	-0.0	4586.	85414.	91868	1.200	4
KW =		0.0	-0.0	-0.0	0.0	0.0	0.	-0.0	-0.0					
3	618	30101	1887.3	1767.0	10.3	2.8	231.	0.9944	1.1541	3602.	1251.	53169	1.550	1
KW =		479.7	1929.0	1771.8	3.2	0.0	194.	0.0	0.0					
1	9051	30102	13.7	1966.7	0.0	557.5	28.	1.0820	1.2504	6455.	-2.	121768	1.250	1
KW =		196.8	14.3	1976.6	-0.0	574.7	-0.	0.0	0.0					
1	2087	30201	2849.3	2667.1	15.4	4.2	315.	0.9850	1.1471	3602.	327.	53169	1.250	1
KW =		720.7	2932.0	2674.3	4.7	0.0	260.	0.0	0.0					
1	2203	30202	2990.7	1690.1	12.2	1.8	295.	0.8063	0.9332	2400.	2765.	63069	2.000	1
KW =		749.3	3035.0	1693.3	4.7	0.0	257.	0.0	0.0					
1	2173	30301	2808.2	2609.1	15.1	4.2	381.	0.9910	1.1522	3602.	1260.	53169	1.550	1
KW =		707.9	2870.0	2616.3	4.7	0.0	327.	0.0	0.0					
1	2199	30302	2828.1	2626.5	13.2	3.4	375.	0.7997	0.9201	2802.	1946.	53169	1.550	2
KW =		730.9	2856.2	2629.8	8.4	1.5	350.	0.3490	0.4002					
3	617	30401	1872.6	1869.8	12.6	4.0	241.	1.2888	1.4908	4734.	192.	42469	1.550	1
KW =		481.3	1928.0	1876.8	3.2	0.0	192.	0.0	0.0					
1	2201	30402	2988.7	1689.1	12.4	1.8	293.	0.8273	0.9498	2400.	2683.	63069	2.000	1
KW =		764.7	3034.0	1692.3	4.7	0.0	254.	0.0	0.0					
1	2211	30501	2865.5	2639.8	8.9	1.4	331.	0.3382	0.3901	1200.	184.	80169	0.450	1
KW =		739.9	2886.7	2642.3	4.7	0.0	312.	0.0	0.0					
1	2099	30502	2874.3	2657.2	12.1	2.9	329.	0.6902	0.7919	2400.	1402.	63069	1.250	1
KW =		754.4	2918.0	2662.3	4.7	0.0	290.	0.0	0.0					

Fig. 3. Printout of Calculation of Burnup by BUCOMP Subroutine

RUN 37 A (1200. MWD) 27474. TO 28674. MWD MAXIMUM BURNUP							
NO	POS	BU AVE	BU MAX	LIMIT	MWDL	RES MWD	MWD
9033	50502	5.0089	6.0556	5.1000	-1929.	10294.	11922.
9012	40202	8.6069	10.2162	10.0000	-423.	19571.	18260.
9050	40302	2.7024	3.1827	3.0000	-347.	5705.	5910.
9051	30102	1.0820	1.2504	1.2500	-2.	6591.	6455.
2132	40601	0.7436	0.8808	0.9000	63.	2980.	2921.
2168	50404	1.4944	1.7820	1.8000	64.	6368.	6051.
2211	30501	0.3382	0.3901	0.4500	184.	1384.	1200.
617	30401	1.2888	1.4908	1.5500	192.	5026.	4734.
3051	60503	1.2186	1.4997	1.5500	201.	6187.	5910.
2188	20301	1.3737	1.5674	1.6500	255.	5084.	4734.
2210	40603	0.3155	0.3707	0.4500	257.	1457.	1200.
2084	50504	1.2662	1.4780	1.5500	257.	5539.	4734.
3048	60604	1.3837	1.7364	1.8000	259.	7337.	7335.
3038	60201	1.5515	1.9464	2.0000	267.	9946.	8135.
3039	60103	1.3881	1.7361	1.8000	272.	7661.	7328.
2176	40403	1.4506	1.7074	1.8000	298.	5801.	5547.
2197	20101	1.0494	1.1962	1.3000	317.	3968.	3602.
387	60105	1.3788	1.7291	1.8000	321.	8143.	7307.
2087	30201	0.9850	1.1471	1.2500	327.	3969.	3602.
3054	60602	1.1792	1.4696	1.5500	340.	6549.	6134.
3047	60101	1.3759	1.7281	1.8000	344.	8602.	7335.
3042	60305	1.3639	1.7157	1.8000	362.	7727.	7335.
474	50601	1.1703	1.4247	1.5500	475.	5877.	5310.
478	50503	1.1552	1.3793	1.5500	598.	5431.	4734.
2097	50604	0.8904	1.0576	1.2500	699.	4541.	3602.
479	50603	1.1233	1.3486	1.5500	718.	5528.	4734.
766	70504	0.0507	0.2483	0.2700	772.	9630.	10502.
475	50303	1.1080	1.3338	1.5500	783.	5614.	4734.
2095	50202	0.8670	1.0350	1.2500	805.	4679.	3602.
476	50203	1.0782	1.3053	1.5500	901.	5708.	4734.
3057	60202	1.0618	1.3374	1.5500	961.	7007.	5910.
2198	20501	0.8264	0.9375	1.3000	1094.	3922.	2802.
618	30101	0.9944	1.1541	1.5500	1251.	4897.	3602.
480	50101	1.0012	1.2282	1.5500	1256.	6047.	4734.
2173	30301	0.9910	1.1522	1.5500	1260.	4909.	3602.
3061	60403	0.7446	0.9171	1.2500	1334.	5009.	3602.
2102	10101	0.7148	0.8047	1.2500	1340.	3762.	2400.
2009	40502	0.9569	1.1257	1.5500	1357.	4958.	3480.
2099	30502	0.6902	0.7919	1.2500	1402.	3826.	2400.
477	50201	0.9682	1.1941	1.5500	1428.	6217.	4734.
3062	60405	0.7188	0.8904	1.2500	1486.	5165.	3602.
2100	40303	0.9258	1.0937	1.5500	1499.	5091.	3602.
786	70604	0.0468	0.2320	0.2700	1519.	10778.	10502.
2117	40102	0.6195	0.7314	1.2500	1708.	4116.	2400.
2179	50402	0.8509	1.0231	1.5500	1889.	5556.	3602.
2199	30302	0.7997	0.9201	1.5500	1946.	4790.	2802.
785	70104	0.0621	0.2272	0.2700	1984.	12515.	11203.
481	50103	0.8257	0.9983	1.5500	2004.	5630.	3602.

Fig. 4. Listing of Subassemblies in Order of Increasing
Megawatt-Days Remaining

SUBASSEMBLIES LISTED IN NUMERICAL ORDER BY TYPE

S/A	POS	S/A	POS	S/A	POS
2009	40502	481	50103	787	70304
2012	40103	482	50301	788	70601
2084	50504	483	50501	789	70403
2087	30201	484	50403	790	70306
2095	50202	617	30401	791	70406
2097	50604	618	30101	792	70401
2098	50102	3002	60402	793	70205
2099	30502	3038	60201	794	70204
2100	40303	3039	60103	795	70404
2102	10101	3042	60305	796	70505
2117	40102	3047	60101	799	70506
2118	40401	3048	60604	837	70503
2122	40301	3051	60503	896	60301
2123	40503	3053	60302	897	70606
2125	50204	3054	60602	9000	20601
2126	50104	3055	60204	9012	40202
2139	50304	3057	60202	9020	60205
2153	40201	3058	60303	9021	20401
2168	50404	3060	60401	9027	40203
2173	30301	3061	60403	9032	60505
2176	40403	3062	60405	9033	50502
2179	50402	3063	60502	9035	70203
2180	50602	3064	60605	9036	70501
2182	40601	3065	60504	9038	70305
2188	20301	3066	60404	9041	70103
2194	30602	3070	60102	9042	70405
2197	20101	3071	60203	9043	40402
2198	20501	3074	60304	9044	70101
2199	30302	735	70301	9050	40302
2201	30402	737	60501	9051	30102
2203	30202	766	70504	9054	40501
2205	30601	773	70603	9055	60104
2210	40603	777	70102	9056	50302
2211	30501	778	70302	9057	20201
387	60105	779	70402	9058	60601
474	50601	780	70502	9059	40101
475	50303	781	70602	9060	70303
476	50203	782	70106	9061	70105
477	50201	783	70202	9062	60603
478	50503	784	70206	9063	70605
479	50603	785	70104	9064	40602
480	50101	786	70604	9704	70201
				2	50401

Fig. 5. Listing of Subassemblies in Ascending Numerical Order by Type

EBR II REACTOR
REACTOR LOADING FOR POWER RUN 37 A
27474. MWD TO 28574. MWD

SUBASSEMBLY LISTING BY GRID POSITION

GRID POS	S/A NO	AVG BU	MAX BU	GRID POS	S/A NO	AVG BU	MAX BU
10101	2102	0.3605	0.4060	60103	3039	1.1617	1.4541
20101	2197	0.7043	0.8031	60104	9055	1.1147	1.4099
20201	9057	0.0	0.0	60105	387	1.1682	1.4638
20301	2188	1.0323	1.1779	60201	3038	1.3636	1.7051
20401	9021	0.0	0.0	60202	3057	0.8513	1.0719
20501	2198	0.4759	0.5397	60203	3071	0.2256	0.2825
20601	9000	-0.0	-0.0	60204	3055	0.4531	0.5667
30101	618	0.6669	0.7743	60205	9020	5.0994	6.2345
30102	9051	0.8850	1.0228	60301	896	0.0011	0.0039
30201	2087	0.6602	0.7692	60302	3053	0.2223	0.2789
30202	2203	0.4060	0.4702	60303	3058	0.4726	0.5874
30301	2173	0.6650	0.7733	60304	3074	0.0	0.0
30302	2199	0.4631	0.5318	60305	3042	1.1403	1.4362
30401	617	0.9691	1.1208	60401	3060	0.4185	0.5304
30402	2201	0.4186	0.4801	60402	3002	0.1450	0.1834
30501	2211	0.0	0.0	60403	3061	0.5018	0.6176
30502	2099	0.3486	0.3998	60404	3066	0.0853	0.1047
30601	2205	0.4096	0.4731	60405	3062	0.4848	0.6000
30602	2194	0.0871	0.1012	60501	737	0.0169	0.0886
40101	9059	1.0219	1.2155	60502	3063	0.2448	0.3026
40102	2117	0.3104	0.3670	60503	3051	0.9741	1.1991
40103	2012	0.3564	0.4251	60504	3065	0.5084	0.6243
40201	2153	0.0	0.0	60505	9032	4.1050	5.0304
40202	9012	8.0880	9.6030	60601	9058	1.0328	1.3245
40203	9027	5.4515	6.4712	60602	3054	0.9509	1.1856
40301	2122	0.0999	0.1191	60603	9062	0.8742	1.0803
40302	9050	2.1669	2.5517	60604	3048	1.1458	1.4420
40303	2100	0.6152	0.7283	60605	3064	0.0229	0.2797
40401	2118	0.0	0.0	70101	9044	-0.0	-0.0
40402	9043	2.2291	2.6135	70102	777	0.0274	0.1350
40403	2176	1.1326	1.3351	70103	9041	-0.0	-0.0
40501	9054	1.8774	2.2156	70104	785	0.0368	0.2013
40502	2009	0.6361	0.7506	70105	9061	0.0	0.0
40503	2123	0.0	0.0	70106	782	0.0262	0.1301
40601	2182	0.4368	0.5184	70201	9704	4.2308	4.5563
40602	9064	1.1261	1.3240	70202	783	0.0261	0.1293
40603	2210	0.0	0.0	70203	9035	-0.0	-0.0
50101	480	0.7502	0.9206	70204	794	0.0374	0.1856
50102	2098	0.2699	0.3281	70205	793	0.0331	0.1675
50103	481	0.5521	0.6680	70206	784	0.0266	0.1314
50104	2126	0.0	0.0	70301	735	0.0192	0.0904
50201	477	0.7253	0.8949	70302	778	0.0272	0.1340
50202	2095	0.6032	0.7144	70303	9060	0.0	0.0
50203	476	0.8088	0.9794	70304	787	0.0164	0.0790

Fig. 6. Listing of Subassemblies in Ascending Order by Grid Position

```

37 A 1200. 27474. 11.5 200.
  1 0 1 0 91 78 0 0 2 0 0 0 1 0 1
.093 .203 1.2 1.91 1.21 .9668
1.2
1.2
1.2 1.2
1.2 1.2 1.2
1.2 1.2 1.2
1.2 1.2 1.2 1.2 1.2
3. 3.1 3.18 3.2 3.18 3.1
2.72 2.85 2.94 2.98 2.98 2.94 2.85
1.125 1.125 1.125 1.125 1.125 1.125 2.26 2.26
13.314E-2 6.792E-3 23.095E-2 6.808E-3 4.2 E-2
12.769E-2 7.497E-3 22.035E-2 8.637E-3 4.457E-2
11.412E-2 6.574E-3 19.660E-2 7.717E-3 3.975E-2
11.859E-2 6.548E-3 20.476E-2 7.308E-3 3.986E-2
9.339E-2 5.500E-3 15.964E-2 7.195E-3 3.435E-2
10.021E-2 5.771E-3 17.187E-2 7.359E-3 3.627E-2
7.033E-2 4.899E-3 11.734E-2 7.602E-3 2.919E-2
7.879E-2 5.174E-3 13.297E-2 7.448E-3 3.127E-2
8.252E-2 5.290E-3 13.984E-2 7.398E-3 3.212E-2
4.689E-2 2.712E-3 7.316E-2 6.334E-3 2.060E-2
5.766E-2 3.783E-3 9.373E-2 7.107E-3 2.516E-2
6.239E-2 4.198E-3 10.266E-2 7.245E-3 2.666E-2
1.492E-2 .535E-3 2.092E-2 1.705E-3 .441E-2
2.060E-2 .924E-3 3.028E-2 2.639E-3 .767E-2
4.086E-2 2.120E-3 6.213E-2 5.439E-3 1.677E-2
4.280E-2 2.298E-3 6.565E-2 5.729E-3 1.799E-2
2900. 2655. 4.1 1525.5 433.5
1937. 1776. 3.2 2137. 459.
1937. 1776. 3.2 2137. 459.
102. 47715. 0. 1958. 206.
1467. 1344.8 2.4 3034. 423.5
2900. 2655. 4.1 1525.5 433.5
102. 47715. 0. 1958. 206.
0. 0. 0. 7047. 138.9
1.0014E+00-9.758E-04--2.528E-04-1.467E-052.6193E-07
6.7695E-02-3.134E-043.8684E-05-4.602E-067.0162E-08
4.2468E-01-2.356E-023.3388E-041.6429E-06-4.599E-08
1.0016E+00-2.729E-04-4.184E-04-2.951E-069.0747E-08
178.95149.50175.81151.71174.24140.83154.46132.60174.24131.17152.90
3.3194E-00-4.204E-013.0796E-02-1.081E-031.4008E-05
1 2102 10101 2886.2 2653.6 8.6 1.5 312.1 0.3605 0.4060 1200. 2504.
0.0 0.0 2909.0 2656.3 4.7 0.0 292.0 0.0 0.0 63069 1.250 10
1 2197 20101 2832.9 2634.1 12.2 3.0 365.1 0.7043 0.8031 2402. 1499.
0.0 0.0 2877.0 2639.3 4.7 0.0 326.4 0.0 0.0 53169 1.300 10
1 9057 20201 -0.0 -0.0 0.0 0.0 -0.0 0.0 0.0 4710. 85290.
1300.0 300.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 32369 9.990 10
1 2188 20301 2807.5 2633.6 15.7 4.4 369.0 1.0323 1.1779 3534. 219.
0.0 0.0 2872.0 2641.3 4.7 0.0 312.2 0.0 0.0 42469 1.650 10
1 9021 20401 -0.0 -0.0 0.0 0.0 -0.0 0.0 0.0 13657. 76343.
6808.0 157.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 41067 9.900 30
1 2198 20501 2845.2 2639.8 9.8 2.0 352.6 0.4759 0.5397 1602. 2259.
0.0 0.0 2852.6 2640.7 8.5 1.5 346.1 0.3571 0.4051 53169 1.300 20

```

Fig. 7. Sample of Input for BURNUP Code

Card 1 has FORMAT (A8, 4F8.0):

Cols

1-8	RUNO	= Number of this reactor run.
9-16	AMWD	= Megawatt-days duration of this run.
17-24	PRMWD	= Total accumulated megawatt-days by the reactor prior to this run.
25-32	BANK	= Control-rod bank position (in.).
33-40	RMEV	= Millions of electron volts per ^{235}U fission (nominally 200 MeV).

Card 2 contains input/output options and has FORMAT (16I5):

Cols

1-5	LFIS	= 1 Perform calculation of fission rates and radial max/avg ratios. = 0 These data are stored in memory (applies to a sequence of problems).
6-10	LOAD	= 1 Core loading for present run is written on Unit 3. Burnup calculations and printing are repeated. This allows two sets of multi-part printouts without rerunning the code. = 0 Loading is not written.
11-15	LHIST	= 0 Read history from cards. = 1 Read history from Unit FT01FO01.
16-20	LPNCH	= 1 Punch burnup calculations. = 0 Do not punch burnup calculations.
21-25	NOCORE	= Number of subassemblies in first six rows
26-30	NOBL	= Number of subassemblies past the sixth row to be considered in the calculation (up to eight rows may be handled).
31-35	IWORTH	= 2 Perform preliminary worth calculation, read in last core, present type and position changes. = 0 Perform worth and burnup calculations.
36-40		Blank.
41-45	IUP	= Option to update or revise history tape on Unit FT01FO01. IUP is the <u>total number</u> of subassemblies to be updated. A pair of cards is required for each subassembly. These cards

follow the load deck, but go before the loading changes for the present run.

= 0 No updating to be done.

46-50 LFR = Number of special fission rates to be read (see Section III.C, Case 3).

= 0 All fission rates to be calculated from FISRT routine.

51-55 NFR = 1 Punch fission rates for each subassembly.

= 0 Do not punch.

56-60 LCTR = 1 The flux-center calculation is not performed, and the flux center is read in.

= 0 Control-rod reactivities read in, and flux-center calculation is performed.

61-65 LPRT = 1 Print fission-rate distribution, radial max/avg numbers, and distances for each subassembly.

= 0 Do not print the above data.

66-70 IDEL = Number of subassemblies to be deleted from history. These subassemblies are no longer available for reinstallation, having reached their burnup limits. IDEL cards with FORMAT (2X,I5) are required, each containing the number of the subassembly to be deleted. Final weights and burnups are punched and also printed for each subassembly deleted. The remaining subassemblies are then written on Unit FT02F001, and the program terminates. Although the code can accommodate information on 200 historical subassemblies, the number in the reactor storage basket cannot exceed 75. Thus, deleting should keep the amount of data stored on tape at about this number.

71-75 LIST = 1 Print a listing of core loading and burnup at the beginning of the run, and list

subassemblies in ascending order according to their numbers. This is done in subroutine CHART (Section III.G).

= 0 Do not print list.

Card 3 has FORMAT (6F8.0):

Cols

1-8	CAPC8	= Captures per g ^{238}U /fissions per g ^{235}U .
9-16	CAPC5	= Captures per g ^{235}U /fissions per g ^{235}U .
17-24	XACC	= $\left\{ \begin{array}{l} ^{238}\text{U} \text{ axial max/avg capture ratios in core} \\ \text{and blanket subassemblies, respectively.} \end{array} \right.$
25-32	XALC	
33-40	PUK	= Relative weight of ^{239}Pu fissions to ^{235}U fissions. PUK = 1.21.
41-48	POWF	= Fraction of reactor power produced in the portion of the reactor being considered in this burnup calculation (for example, in the first eight rows).

B. Axial Max/Avg Fission and Capture Ratios and Worths

If LFIS is greater than zero, the following data are required. Axial max/avg values for both ^{235}U and ^{238}U are obtained from MACH-1 calculations and input into BURNUP.

Cards 4-11 have FORMAT (8F10.0):

XA8 = ^{238}U axial max/avg values. Each of the eight rows over which the burnup calculation is performed is allotted a card that has an XA8 value specified for each subassembly position in one sector.

Card 12 has FORMAT (8F10.0):

XA5 = ^{235}U axial max/avg values. Here, only one value is required for each of the eight rows, so only one card is used.

The BURNUP code is capable of calculating the reactivity change resulting from replacing one subassembly with another. To compute these changes, the reactivity worth, in lh/g , of the material in each subassembly

is required. Since a given type of subassembly (driver, control rod, etc.) has a weight which is similar to that of other subassemblies of the same type, these data are also input to enable one to perform preliminary worth calculations. The data for both the worth and material are read according to FORMAT (5E12.4).

The numbers corresponding to the type of subassembly are:

- 1 Enriched uranium driver
- 2 Control rod
- 3 Safety rod
- 4 Outer-blanket depleted uranium
- 5 Enriched uranium, half worth
- 6 B-type enriched
- 7 Inner-blanket depleted uranium
- 8 Stainless steel

Worth data are provided for subassembly positions in one sector for the first seven rows. Thus, 16 cards are required for the 16 different positions. These data are calculated by the MACH-1 reactor code.

Cards 13-28 have FORMAT (5E12.4):

WTH(I,1) = Worth of ^{235}U in lb/g (for position I).

WTH(I,2) = Worth of ^{238}U .

WTH(I,3) = Worth of ^{239}Pu .

WTH(I,4) = Weighted worth of stainless steel (obtained by weighting amounts and worths of iron, nickel, and chromium in stainless steel).

WTH(I,5) = Worth of sodium.

Eight material tables corresponding to the subassembly types are required. The order of the weights, in grams, is ^{235}U , ^{238}U , ^{239}Pu , stainless steel, and sodium.

C. Fission-distribution Parameters

In calculating the fission and capture distributions, the BURNUP code requires the coefficients of a fourth-order polynomial curve fit to these data.

Cards 29-32 and 34 have FORMAT (5E10.6):

Card 29 F25(1) - F25(5) Five polynomial coefficients for ^{235}U .

Card 30 F28(1) - F28(5) Five coefficients for ^{238}U in rows 1-6.

Card 31 F28(6) - F28(10) Five coefficients for ^{238}U in rows 7 and 8. These five coefficients must be multiplied by the first coefficient obtained from fitting data in the first six rows.

Card 32 C28(1) - C28(5) Five coefficients for ^{238}U capture.

The amount of data next required depends on the options specified on Card 2.

Case 1: LCTR = 1. The nuclear center has been calculated prior to this run. Read in R2 and THETA (Section II.C).

Card 33 has FORMAT (2F10.6):

Cols

1-10 R2 = Distance (cm) between the geometric and nuclear center of the reactor.

11-20 THETA = Angle (radians) between R2 and the reference line between sectors C and D, in the counter-clockwise direction.

Case 2: LCTR = 0. Subroutine FLXCTR will be called, and the nuclear center calculated. The calibrated reactivity worths of control rods 2 through 12 must be read.

Card 33 has FORMAT (11F6.0, 2I3):

Cols

1-66 RHOI = Calibrated control-rod worth (1h) for rods 2-12. (Control rod 1 is a stainless steel dummy and is ignored.)

67-69 ID1 = Any other odd-numbered dummy control rod.

70-72 ID2 = Even-numbered dummy control rod. Dummy control rods must be supplied with an average worth. (See Section I.C.) Dummy rods are installed only in special cases.

Card 34 W(1) - W(5) Five coefficients for the polynomial fits of the MACH-1 $(\Delta k/k)/\Delta V$ data.

Case 3: LFR $\neq$ 0. At times, it may be desirable to alter the relative fission rate for a particular subassembly to account for some local perturbation. LFR specifies the number of positions to be altered. LFR cards are read.

Alternate Cards 35...LFR + 35 have FORMAT (I5, 4F10.3):

Cols

1-5	NS	= Number of the subassembly to which this special rate applies.
	F5S	= Special ^{235}U relative fission rate.
	F8S	= Special ^{238}U relative fission rate.
	C8S	= Special ^{238}U relative capture rate.
	RXS	= Special ^{235}U radial max/avg ratio.

D. Load Deck

The next set of data is the load deck, consisting of the following input for each subassembly in the first eight rows of the reactor. The load deck is punched by the computer and is the reactor loading at the end of the last run. It consists of NOCGRE + NOBL = NOS pairs of cards.

Cards 35 and 37 have FORMAT (I2, I5, I7, 5F7.1, 2F7.4, 2F7.0).

Cards 36 and 38 have FORMAT (7F.1, 2F7.4, I7, F7.3, I2).

The first set of numbers below (not in parentheses) denotes data fields on the first card of the pair for each subassembly. The numbers in parentheses denote the field on the second card.

Cols

1-2	KN	= Subassembly type number (see Section III.B).
3-7	NO	= Subassembly number.
8-14	LIP	= Subassembly position in the reactor, e.g., position 4B3 is input as 40203, where 4 = row number and 2 = section number (A=1, B=2, ..., F=6).

- (1-7) GS = Weight of stainless steel within core height (13.5 in.) in the subassembly.
- (8-14) GN = Weight of sodium. The weights of stainless steel and sodium need be specified only for the experimental subassemblies (NO = 9xxx). These data for the remaining subassemblies are supplied by TABM (Section III.B).
- 15-21 G5 = Weight of ^{235}U in the subassembly at the end of the last run (irradiated weight).
- (15-21) G50 = Weight of ^{235}U when it was originally installed in the reactor.
- 22-28 G8 = Present weight of ^{238}U .
- (22-28) 680 = Original weight of ^{238}U .
- 29-35 G6 = Present weight of ^{236}U .
- (29-35) G60 = Original weight of ^{236}U .
- 36-42 G9 = Present weight of ^{239}Pu .
- (36-42) G90 = Original weight of ^{239}Pu .
- 43-49 FZ = Present weight of fissium alloy.
- (43-49) FZD = Original weight of fissium alloy.
- 50-56 BU = Present average (center) atom-percent burnup of subassembly fuel.
- (50-56) BI = Original average (center) atom-percent burnup of subassembly fuel.
- 57-63 BX = Present maximum (pin nearest fission center) atom-percent burnup.
- (57-63) BMI = Original maximum (pin nearest fission center) atom-percent burnup.
- 64-70 DD = Number of megawatt-days the subassembly has been irradiated.
- 71-77 DL = Megawatt-days left until the maximum burnup reaches the atom-percent limit AL.
- (63-70) IN = Original date of installation, i.e., October 25, 1968 reads 102568.
- (71-77) AL = Atom-percent burnup limit.
- (78-79) KC = Number of times the subassembly has been installed in the reactor.

E. History

After the load deck has been read, the history is read. The history consists of subassemblies that have been in the reactor and then transferred back to the storage basket before reaching maximum burnup. Therefore, these subassemblies are available for reinstallation.

If LHIST = 0, the history is read from cards in the same format as the load deck, two cards per subassembly. It is assumed that IUP = 0 in this case. If LHIST = 1, the history is read from a magnetic tape on Unit 1. These data correspond to those which would be read from cards.

If IUP $\neq$ 0, two cards must be supplied for each of the IUP subassemblies that are to be modified on the tape. These cards have the same format as those in the load deck, except that the last card of the pair has the ISTOP in column 80.

ISTOP = 2, LHIST = 0 Last card of history input deck.

ISTOP = 0, LHIST = 0 More history follows.

ISTOP = 1, LHIST = 1 All data for this subassembly are to be removed from history tape. The values of KC and the maximum burnup must equal those for this subassembly in the history.

ISTOP = 0, LHIST = 1 Updating is to be done to correct data in the history.

F. Loading Changes

The last set of input consists of subassemblies which are to be put into the reactor for this run. Each replacement subassembly requires one card.

Replacement cards have FORMAT (I2, I5, I7, 7F7.1, F7.3, I2, I1):

Cols

1-2 KNR = Type number of replacement subassembly.

3-7 NOR = Number of replacement subassembly.

8-14 IJPR = Core position of replacement subassembly.

15-21 G5RO = Initial weight of ²³⁵U, obtained from Fuel Cycle Facility (FCF) transfer sheet.

- 22-28 G8RO = Initial weight of ^{238}U . Note that this weight is equal to the weight of ^{238}U on the transfer sheet minus the weight of ^{236}U in the subassembly.
- 29-35 G6RO = Initial weight of ^{236}U . For some subassemblies, the weight of ^{236}U is available from FCF personnel. If it is not, an initial weight of 4.7 g is assigned arbitrarily to drivers and 3.2 g to control and safety rods.
- 36-42 G9RO = Initial weight of ^{239}Pu , from transfer sheet.
- 43-49 FZRO = Initial weight of fissium alloy in the fuel. In reference to the transfer sheets:

$$\text{FZRO} = \text{Weight of uncanned pins} - (\text{G5RO} + \text{G8RO} + \text{G6RO} + \text{G9RO})$$

$$= \text{weight of uncanned pins} - (\text{weight of U} + \text{Pu}).$$
- 50-56 GSR = Weight of stainless steel in 13.5-in. core height of subassembly.
- 57-63 GNR = Weight of sodium.
- 64-70 INR = Installation date.
- 71-77 ALR = Atom-percent burnup limit.
- 78-79 KCR = Number of times subassembly has been installed in the reactor, including this one.
- 80 NIPT = Option designating source of data for each replacement.
 = 0, 1 Replacement subassembly is new or stored in the history. If KC = 1, all data must be supplied. If KC is greater than 1, weights are found in the history. In the latter case, no weights or installation date need be supplied on the card. However, the value of KCR must be 1 greater than the value of KC contained in the history for that subassembly. If not, a diagnostic is returned. ALR, the burnup limit, must be given.
 = 2 Subassembly being replaced is to be installed in a new core location. It is stored by the computer in a temporary array. The replacement subassembly is new or in the history, and data must be supplied for it accordingly.

- = 3 Same as 2, but the replacement subassembly is found in the temporary array (i.e., it has been switched, just as the one it is replacing has been).
- = 4 The replacement subassembly is found in the temporary array.
- = 5 This is the last change card. The last card must be one where NIPT would = 0 or 1.

IV. OUTPUT

The output from the code consists of the following:

1. A printout of the history at the beginning of the current run.
2. A magnetic tape containing these data, written on FORTRAN Unit 2 (FT02F001).
3. A printout of subassemblies moved into the basket for the current run, appearing as future history, and also written on Unit FT02F001.
4. Punched cards for subassemblies updated within or removed from the history.
5. A printout showing the calculated nuclear center and associated control-rod data.
6. An optional printout of the fission distributions and radial max/avg values if LPRT = 1.
7. A table of central fission rates, total fuel weights, and percentage of power in each row (Fig. 2).
8. A printout of the current core loading (Fig. 3). Two lines are allowed for each subassembly. The first line consists of weights, burnups, total megawatt-days on the subassembly, and megawatt-days remaining at the end of the current run. The second line shows the power produced due to fissioning, the fuel weights and burnups when the subassembly was installed at this KC value, the initial installation date, and the burnup limit.
9. A punched-card deck of the data in 8. These cards will serve as the load deck for the next BURNUP run.
10. A table arranging all subassemblies in order of increasing megawatt-days left (Fig. 4).

11. An optional table showing the core loading and burnups at the beginning of the run (Figs. 5 and 6).

V. CONCLUSION

It is recognized that the BURNUP code, while serving its purpose of providing a record of subassembly irradiation during an EBR-II run, is subject to uncertainties. At the moment, however, it is our only means of calculating burnup.

Periodically, subassemblies are analyzed for burnup by the chemical analysis group. The results calculated by BURNUP and those obtained by analysis agree within 5% on the average.

The figure of 200 MeV per fission the BURNUP code uses is open to some question. Use of 201 or 202 MeV has been suggested. Additional study is also needed to more closely determine the fraction of power produced in the blanket.

It has also been suggested that calculations of fuel burnup should take into account cross sections, the change in fuel, and the variations in fission rate during the course of a run. Since the maximum run is 1200 MWd, we do not consider this. For a large power reactor running for thousands of megawatt-days before fuel handling, such factors should be considered. So far, the BURNUP code has provided us with a simple, reliable method of keeping track of fuel consumption in EBR-II.

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