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—JOYPAC System—

Part 1

The Concept of Code System,
the Simplified Calculation Subsystem
Predicting the Core Characteristics,
and the Recording Subsystem of JOYO
—SMART and MASTOR Codes—

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The Off-Line Computation System for Supervising Performance of JOYO —JOYPAC System—†

Part 1

The Concept of Code System, the Simplified Calculation Subsystem Predicting the Core Characteristics, and the Recording Subsystem of JOYO ——SMART and MASTOR Codes——

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A code system JOYPAC for monitoring the operation of the fast experimental reactor JOYO has been developed. This is an off-line code system designed for use in making calculation of the nuclear and thermohydraulic characteristics of the reactor core and also to make computation of the history of core irradiation after reactor operation. The use of the code system makes it possible to calculate the various core characteristics with a high degree of accuracy by simplified procedure for the diverse operation patterns of JOYO to confirm its safety. It also enables the details of the history of irradiation of the core to be obtained quickly and accurately after reactor operation. The above include all the operation data and in-pile characteristics that are required for the irradiation test. Furthermore, it is also possible to provide the data for the on-line computer system of JOYO and the data for nuclear material accountability.

The code system consists of the detailed subsystem and the simplified subsystem. The former is used for obtaining the nuclear and thermohydraulic characteristics of the core by use of a detailed calculation model such as three-dimensional hexagonal lattice, for instance, in order to back up the simplified subsystem. On the other hand, the latter is designed to obtain the various core characteristics by use of simple extrapolation and interpolation methods, whose conception is based on the great deal of information obtained by the design calculation of JOYO and the many parameter surveys. The system is used for the normal cycle operation.

Part 1 of this report is devoted to the description of the general conception of the code system, detailed description of the simplified subsystem, the brief application planning for JOYO and its relations with the on-line computer system. The detailed subsystem is described in detail in Part 2.

The computer to be used for the code system is IBM 360/K195.

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† The work was performed under the contracts between Power Reactor and Nuclear Fuel Development Corporation and Japan Atomic Energy Research Institute.

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「常陽」運転監視用オフラインコードシステム
—JOYPAC システム—†

第 1 部

コードシステムの概念，簡易炉心特性解析サブシステム，および記録サブシステム
——SMART, MASTOR コード——

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1976 年 6 月 9 日　受理

高速実験炉「常陽」の運転監視用コードシステム JOYPAC を開発した．これは炉心の核特性および熱水力特性の予測計算を行ない，かつ運転後の炉心照射履歴を計算することを目的とした，オフラインのコードシステムである．このコードシステムを使用することにより，「常陽」の多様な運転パターンに対して，炉心の諸特性を簡単な操作で精度良く計算し，安全性を確認することができる．また運転後には炉心の詳細な照射履歴を短時間に精度良く求めることが出来る．これには照射試験で要求される全ての運転データおよび炉内特性値が含まれる．さらに，「常陽」のオンライン監視システムへのデータの提供，核物質管理のためのデータも提供できる．

本コードシステムは詳細サブシステムと簡易サブシステムにより構成されている．詳細サブシステムは 3 次元六角格子などの詳細な計算モデルと解法により核特性および熱水力特性を求め，簡易サブシステムのバックアップとして使われる．簡易サブシステムは，炉心諸特性を簡単な内外挿方式により求めるもので，その概念は「常陽」の設計計算で得た多数の情報と多数のパラメータサーベイに基づいている．簡易サブシステムは通常のサイクル運転に使用される．

本研究報告 Part 1 は，コードシステム全体の概念を明らかにし，簡易サブシステムの内容を詳述する．また「常陽」における運用計画，オンライン監視システムとの関連等についてその概要を述べる．詳細サブシステムについては Part 2 で詳述される．

本コードシステムの使用計算機種は IBM 360/K195 である．

† 本報告書は日本原子力研究所が動力炉・核燃料開発事業団の委託により行なった研究の成果である．

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1. Introduction

During the operation of the reactor, the parameters of reactor characteristics vary with the burn-up of fuel, the movement of the control rod, the exchange of fuel and various external turbulences. That is to say that the nuclear and thermal characteristics of the core which is the heart of the reactor varies from moment to moment although gradually.

It is important for the safety operation of a reactor to grasp correctly such changes of the reactor core behavior and to predict the changes of its various characteristics for the following cycle plan. It is possible to make a proper reactor operation program only when the core properties are correctly estimated.

The fast experimental reactor JOYO is presently being subjected to the performance test and thereafter it will go into operation for the purpose of performing irradiation tests for the fuels and materials of fast reactors. During the initial stage of its operation, the driver fuel assembly will be taken out of the core after each cycle of operation to test and analyze its irradiation behavior. If the irradiation test and analysis of the fuel and materials are to be made accurately, it is essential to have accurate information concerning the history of irradiation and the values of the various characteristics of the core. On the other hand, calculations of density changes of the nuclear material due to burn-up of the fuel are also necessary from the standpoint of nuclear fuel management. The proper operation definitely requires a code system by which all such information can be obtained easily.

It does not seem that any practical code system which can be used extensively to service such purposes has ever been developed for fast reactors, although the on-line and off-line code systems developed on the basis of many years of experience have been put to practical use for light-water power reactors.

The JOYO operation monitoring code JOYPAC*[1] has been developed as an off-line code system capable of making comprehensive analysis of the nuclear and hydrothermal characteristics of the reactor core in order to meet the above-mentioned requirements. For the sake of application convenience, this code system has been divided into two subsystems, that is, the subsystems of simplified and detailed methods. The former is used to approximate the core characteristics by simple formulas, not very strict calculations, on the basis of experiences acquired through the design calculation for JOYO. The simplified method is composed of the subsystem SMART*[2] for performing main computations and the subsystem MASTOR*[3] for controlling the input and output data. The code system is used for each cycle of operation.

The detailed method is used for determining the various core characteristics by use of strict calculations and it is composed of the subsystem HONEYCOMB*[4] for calculation of nuclear characteristics and the subsystem FDCAL–3*[5] for calculation of the thermohydraulic characteristics of fuel assemblies. The detailed method is used as a back-up calculation system for the simplified method or used independently as required.

The code system was planned by the Power Reactor and Nuclear Fuel Development Corpo-

*1 JOYPAC: JOYO Performance Analysis Code System
*2 SMART: Simplified Method to Analize Reactor Technical Performance
*3 MASTOR: Monitoring And Supervising Tool on Operation of Reactor
*4 HONEYCOMB: Three dimensional hexagonal lattice like honeycomb
*5 FDCAL-3: Flow and Temperature Distribution Calculation Code

ration and the overall plan and technical problems were studied by the working group* of the Nuclear Code Committee of the Japan Atomic Energy Research Institute. Of the simplified subsystems, the SMART was made up by Tokyo Shibaura Electric Co., Ltd. and the detailed subsystems were all made up by the Japan Atomic Energy Research Institute, and both were made under the contracts with the Power Reactor and Nuclear Fuel Development Corporation[1].

The report on the system consists of two parts: Part 1 deals with the general description of this system, explanation of the simplified system and the application for JOYO and Part 2 is devoted to the description of the detailed system.

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2. Conception of Code System

2. 1 Objectives

The objectives of the JOYPAC system are as follows.

(1) To make core performance calculation before each cycle of reactor operation to see whether the core characteristics satisfy the safety requirements or not.

(2) To provide the data relating to the core, which are necessary for the monitoring of reactor operation by means of the on-line computer system.

(3) To calculate the values of core characteristics on the basis of actual performance of reactor after each cycle of operation and keep the performance records.

(4) To provide out of the performance records such data that are required for the test and analysis of fuels and materials after irradiation.

(5) To calculate the amounts of changes of nuclear material due to burn-up so that thus obtained data may be effectively used in the nuclear material control.

This code system is so designed all such operations can be performed with a minimum amount of effort within a short period of time, thereby to increase the efficiciency of reactor operation and irradiation test.

2. 2 Functions

2. 2. 1 Prediction calculation

The prediction calculation is made before a cycle of operation to confirm the safety of the core and also to provide information necessary for safe reactor operation such as the data to the on-line computer system. In this case the main input consists of the mode of the specific cycle of operation (reactor power, duration of reactor operation) and the core layout. The main outputs are shown in TABLE 2. 1.

(1) Checking of safety

In order to ascertain the safety of the core when the reactor is power-generating operation, evaluation will be made as to whether the values of the main core characteristics are within the limits of the design criteria. Evaluation is made of the following items.

1) Items to be evaluated for reactivity

a. Reactivity worth of control rod (regulating rod and safety rod)

b. Reactivity shut-down margin in each operating status of reactor

c. Maximum reactivity insertion rate

d. Total excess reactivity

e. Isothermal temperature coefficient and power coefficient

The safety of these items are to be evaluated, considering the width of uncertainty. If the criterion for evaluation of the property A is "to be below the limit value W" and UW is the width of uncertainty including a margin and UF is the width of uncertainty not including such a margin, the WARNING message will be printed out when

$$A(1+UW)>W,$$

TABLE 2. 1 Characteristic parameters in prediction calculation

1. Nuclear characteristics	2. Thermal characteristics
(1) K_{eff} at each power level	(15) Coolant flow rate of fuel assemblies
(2) Reactivity balance	(16) Pressure drop
(3) Isothermal reactivity coefficients	(17) Temperature of pellet, clad and coolant of fuel subassemblies
(4) Power reactivity coefficients	(18) Temperature of pellet and clad of control rods
(5) Control rod worth	(19) Temperature of reflectors and clad of neutron source
(6) Shut down margin	3. General
(7) Control rods withdraw length	(20) Fuel inventory
(8) Reactivity insertion rate of control rods	(21) Average burn up and composition of each core zone
(9) Neutron flux distribution	(22) Flow distribution
(10) Power distribution (included gamma power distribution)	(23) Thermal neutron flux at neutron detector positions
(11) Fluence	(24) Neutron source intensity
(12) Integrated power output	(25) Kinetic parameters for on-line computer system
(13) Burn up of fuels and control rods	(26) Coolant outlet temperature of fuel assemblies
(14) Weight of nuclear materials	
*(9)-(14) given for each core assemblies.	

and the FATAL ERROR message will be printed out when

$$A(1+UF)>W.$$

2) Evaluation items for the degree of burn-up

a. Maximum degree of burn-up of the core and blanket fuel assembly

b. Maximum integrated neutron flux of reflector

c. ^{10}B burn-up degree of control rod

3) Evaluation items for thermal and hydraulic characteristics

a. Maximum temperature of fuel pellet

b. Maximum temperature of clad

c. Maximum temperature of coolant

The above maximum temperatures were obtained, taking the hot spot factors into consideration. The hot spot factors are obtained by the semi-statistical treatment method of one-point approximation as in the desigh calculation.

(2) Supply of data to on-line computer system

An on-line computer is installed in the JOYO control room so that it will detect any change from normal occurring in the core and plant and issue a warning as required. In this system, the SMART will supply the following data.

a. Various reactivities for monitoring the abnormal reactivity balance

b. Various constants for solving the dynamic characteristics of the plant

c. Power and flow rate of fuel assemblies

Apart from the above, it also supplies such data for reactor operation as those of the neutron flux at the position of the neutron detector when the reactor is started. This is obtained by evaluating the intensity of neutron source, taking into consideration the duration of reactor operation and shut-down.

2. 2. 2 Record calculation

The record calculation is done after completion of a cycle of operation. On the basis of various measured values obtained in such transient states as when the reactor is started and stopped and in the steady state of the reactor, re-calculation is made for the core characteristics to obtain the operation records. Operation records thus obtained are kept not only for the JOYO's own use but also to provide the history of irradiation data of irradiated specimens and the fuel management data.

(1) Editing of operation records

As for the operation records, there is the operation log treated on on-line computer system, in which detailed data concerning the entire plant system including the reactor core are recorded. The operation records in this system place emphasis on the analysis of behaviors of the core and fuel. The data obtained by the instrumentation system are edited for each scanning cycle, and are compiled as the output data at an appropriate frequency when the reactor power is in a transient period and when it is in a steady state. The output data will be the main input data for use in the calculations to be made by SMART. The data computed by SMART and the operation data from the on-line computer system are copied to be used as a master file so that these data may be retrieved if necessary.

(2) Calculation of actual performance values

On the basis of the operation data, the values of various characteristics of the core and fuel are recalculated by use of SMART with regard to the same items as in the case of prediction calculation. Some of the predicted values are compared with the corresponding measured ones. These are the insertion depth of the control rods, the neutron flux value at the position of neutron detector and the coolant outlet temperature of fuel assemblies. The measured values at the control rod position in the various conditions of the core give correction factors for the predicted values of the effective multiplication factor, temperature coefficient and power coefficient. The ratios of the predicted values to the measured values are stored in a special data file so that the data may be used for evaluating the accuracy of the predicted values for the subsequent cycle of operation and for correcting the basic data that are used in the SMART.

(3) Fuel management

As for the core components such as fuel assemblies, control rod and reflectors, records are kept of their movements and operations within the site of JOYO. Calculations are made of the changes in the amount of nuclear material with the burn-up or movement of the fuel.

2. 3 Composition

2. 3. 1 General description

This code system is composed of three subsystems, i.e. the detailed subsystem (HONEYCOMB and FDCAL-3), the simplified subsystem (SMART) and the record subsystem (MASTOR). **Fig. 2. 1** shows a flow chart of these subsystems centering around the JOYO. As seen from this chart, SMART and MASTOR are used for normal cycle operation. The detailed calculation subsystem is independent of those subsystems except for its function of providing the basic data. The differences between the detailed method and the simplified method lie in the method of calculation of the characteristics and in the function. That is to say, the former performs detailed calculations by use of stricter calculation models and analytical method while the latter stores numerous basic data so that the values of the various characteristics may be determined by combining or interpolating or extrapolating such data. Due to the functional requirements, the simplified method has to handle an enormously great variety and quantity of characteristics while the detailed method is used only for a limited number of characteristics[2].

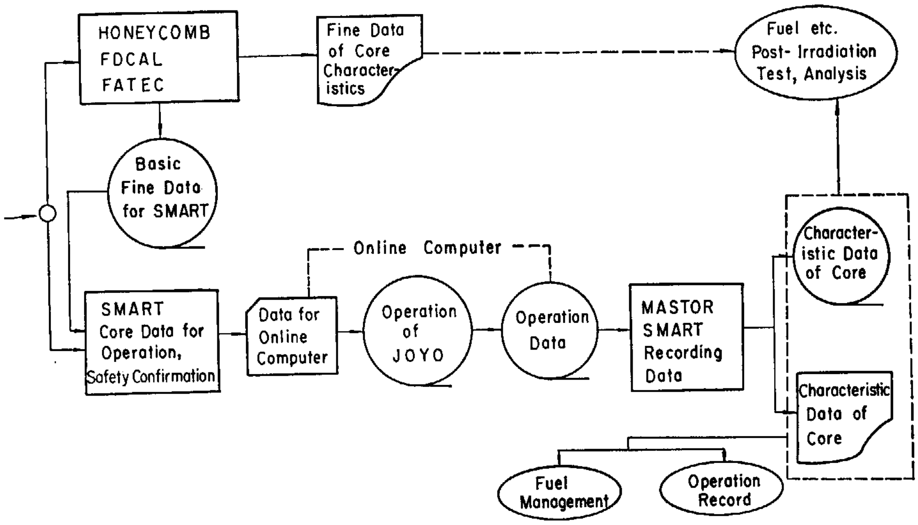


Fig. 2. 1 Conceptual flow diagram of JOYPAC system.

2. 3. 2 Detailed calculation subsystem

This subsystem is designed to analyze the nuclear and thermo-hydraulic characteristics of the core by use of stricter modes and analytical methods. As for the nuclear characteristics (HONEYCOMB), a three-dimensional hexagonal model and the diffusion theory are used to make approximate calculations of (1) the critical search by adjusting the depth of insertion of the control rod, (2) the three-dimensional neutron flux and adjoint neutron flux, (3) the gamma heat distribution, (4) the power distribution, (5) the distribution of power of pin bundles in fuel assemblies, and (6) the burn-up of fuel assemblies and fuel pins. Recalculations are made of the above (1)~(5) after the burn-up of the fuel. Ingenious methods are employed in the calculation of the average neutron flux distribution in the control rod channel and the power distribution of the fuel pin bandle so that the approximate analysis may be made with the same accuracy as the strict analysis. As for the basic data to be provided for the simplified method SMART, there are such data as the neutron flux distribution and adjoint neutron flux and the power distribution within the fuel pin bandle.

The flow distribution in the reactor can be obtained by analyzing the pressure balance on the basis of the coefficient of pressure loss in the various parts of the reactor and the power distribution that has been calculated by HONEYCOMB. The temperature distribution in the fuel assembly is obtained by calculating the temperatures of the coolant, clad and fuel pellets on the basis of the results of calculations made of the pin bandle power distribution and the flow distribution by use of HONEYCOMB. The temperature of the coolant is calculated taking into account the mixing effect in the pin bandle and the temperature distribution of the clad and fuel pellets is calculated in taking into account the heat transfer for the r–θ direction.

The detailed calculated values of the various characteristics within the pin bandle will be of great use in the analysis of the irradiation test.

This subsystem is described in detail in Part 2 of this report.

2. 3. 3 Simplified calculation subsystem

SMART provides all the core data necessary for the operation of JOYO. The method of calculation of these data makes use of the peculiarity of the characteristics of the core and is based on the great deal of information that have been accumulated through the designing of the core of JOYO and the analysis of the core characteristics. That is to say that the core of JOYO is characterized by the fact that the variations of nuclear characteristics with the operation of the reactor are relatively small and a good linearity is obtained on the amounts of changes of the core characteristics with the main causes of variations of nuclear characteristics such as changes in the number of fuel assemblies in the core, the changes in the neutron flux spectrum with burn-up of fuel, and the changes in the depth of insertion of the control rod. This being so, if a reference system is determined and the nuclear characteristics in the specific system and the rate of change of them with the above-mentioned factors are obtained accurately beforehand, the nuclear characteristics of a system deviated from the reference system can be predicted by the superposition in relation to the respective variation factors. The subsystem SMART is used to carefully examine them and evaluate the neutron flux distribution, power distribution, reactivity coefficient and control rod reactivity worth by the above method.

The flow distribution in the reactor is obtained by analyzing the pressure balance by use of a total pressure loss coefficient for each element. As for the fuel pin temperature distribution, the temperature of the various parts of the maximum power pin in the fuel assembly is obtained on the basis of the power peaking coefficient. The coolant temperature is obtained by use of the correction coefficient fitting the mixing effect between subchannels by the assembly flow rate and power distribution. The maximum temperatures, considering the hot spot factors, in each orifice region, are picked out to make judgement as to whether the criteria are satisfied or not.

As the auxiliary systems for SMART, there are EBIS-3 which is a code system for the principle data library control, and RAND which is a system for the data library control for each fuel assembly. These system facilitate the retrieval addition and correction of data. This subsystem is described in detail in Chap. 3.

2. 3. 4 Recording subsystem

The main functions of MASTOR are the editing and output of all data in the record calculation and the data file control and it constitute the recording subsystem in conjunction with SMART. The operation data of JOYO are supplied in the form of magnetic tapes by the online computer, MASTOR changes the file format and reedits it for use by SMART and at the same time makes up the records of movement and operation of the core components. The values of various characteristics that are put out from SMART are printed out in a format suited for the particular use. All the data concerning the operational performance of the reactor are transferred to the master file so that they can be used when required for such purposes as the post-irradiation test analysis. The transfer of data to and from SMART is executed via the work file. This subsystem is described in detail in Chap. 4.

2. 4 Characteristic features

This code system has such characteristic features that (1) it is divided into two subsystems, i.e. the detailed method and the simplified method according to the purpose for which they are to be used, (2) it provides extensive data not restricted to those concerning the core characteristics to give greater convenience for the operation of the reactor, (3) the calculation codes contained in this systems, such as those in the detailed method, are so ingeniously designed as to be superb as independent calculation codes and they are most effectively linked together, and (4) the simplified method of this code system is able to calculate the values of many characteristics very quickly and as accurately as the design values. SMART and MASTOR have been developed to suit the system of JOYO and therefore not for general purposes. However, HONEYCOMB and other codes of the detailed calculation method can be made applicable to other fast reactor systems only by making some minor alterations to them.

The computer to be used for this code system is IBM 360/K 195. The run times and core sizes for these subsystems are shown in TABLE 2. 2.

TABLE 2. 2 Core size and run time

Program name	CPU time	Channel time	Core size
HONEYCOMB	0.75 hr	0.065 hr	484 KB
FDCAL-3 (linked with HONEYCOMB)	0.0282	0.0206	430
SMART	0.027	0.11	256
MASTOR (linked with SMART)	0.03	0.18	512

3. Subsystem SMART

3. 1 General description of calculation methods

It is possible to satisfy the functional requirements of this system as described in the preceding chapter by using the calculation method usually used in design calculation but it calls for enermous amount of computer time and man-power and therefore deemed unsuited for use at every reactor operation cycle. Thus we reached the conclusion that it would be necessary to develop a more simplified method of calculation. We examined the amounts of changes occurring in the various JOYO characteristic values with its operation in reference to the data obtained through the process of design calculation, and found that the amounts of such changes were not so large. So, we decided upon the following basic policy for the development of the simplified calculation method.

Assuming that the characteristic values are all obtained with satisfactory accuracy with regard to the reference system, a method for estimating the amounts of changes in the characteristic values with the reactor operation would be developed.

3. 1. 1 Control rod worth

In the operation planning of JOYO, the blanket fuel assemblies in the fifth row are to be replaced with core fuel assemblies in order to compensate the decrease in reactivity with the burn-up of the core. Therefore, with the reactor operation, the number of core fuel assemblies will increase so that the core system will grow larger in size.

As for the factors causing the changes in the control rod worth with the reactor operation, it is necessary to consider the following:

a) Burn-up of absorber ^{10}B.
b) Increase in the number of core fuel assemblies and increase in size of the core system.
c) Burn-up of core fuel.

Figure 3. 1 shows the relationship between the amount of ^{10}B in the absorber and the control rod worth. From this, it is seen that

a) As 1% of ^{10}B in the absorber burns, the control rod worth decreases by 0.7%.
b) The decrease in the control rod worth with the burn-up of ^{10}B is proportional to the degree of burn-up of ^{10}B.
c) The permissible degree of nuclear burn-up of the control rods of JOYO is about 5% while the corresponding decrease in the control rod worth is 3.5%.

Figure 3. 2 shows the relationship between the degree of burn-up of the core fuel and the control rod worth with the number of core fuel assemblies remaining constant. It is seen that

a) The control rod worth increases with the burn-up of the core fuel and it increases about 3% from the initial worth when the average degree of burn-up of the core reaches 30,000 MWD/T.
b) The changes in the control rod worth with the burn-up of the core is approximately proportional to the changes in the average degree of burn-up of the core.
c) The above proportional coefficient does not depends on the number of core fuel assemblies.

Figure 3. 3 shows the relationship between the number of core fuel assemblies and the control

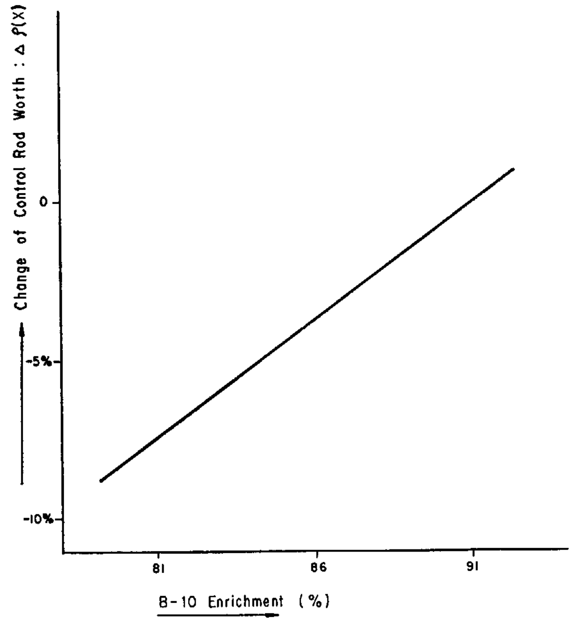


Fig. 3. 1 Control rod worth vs. ^{10}B enrichment.

$\Delta\rho=\dfrac{\rho(91)-\rho(X)}{\rho(91)}$, $\rho(X)$=Reactivity worth of X% enriched control rod.

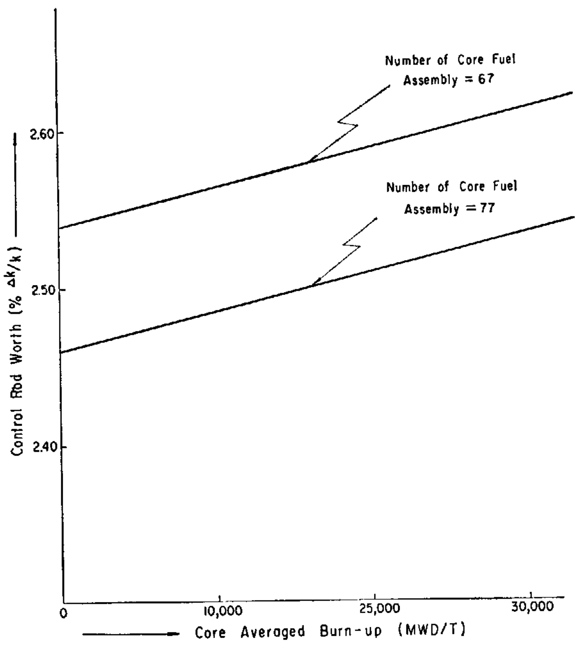


Fig. 3. 2 Control rod worth vs. core burn-up.

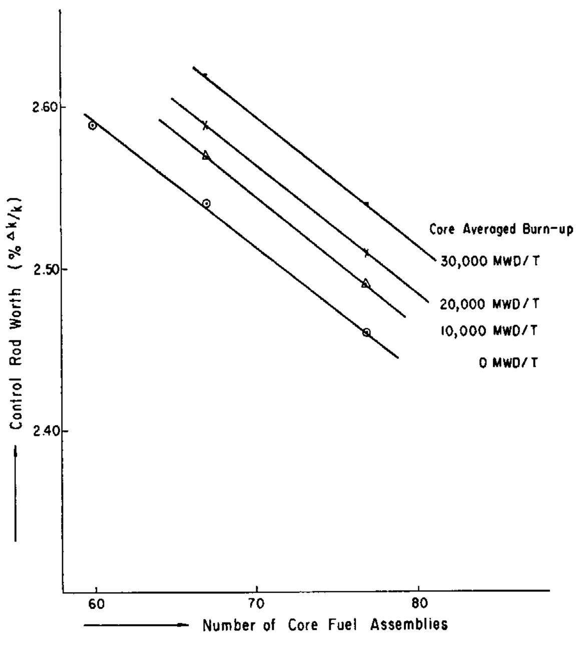


Fig. 3. 3 Control rod worth v.s number of core assemblies.

rod worth with the degree of core burn-up (composition) remaining constant. From this, it is seen that

a) As the number of the core fuel assemblies increases, the control rod worth decreases,

b) As the number of core fuel assemblies increases by ten, the control rod worth decreases by about 3%, that is, the changes in the control rod worth with the changes in the number of core fuel assemblies are relatively small,

c) The changes in the control rod worth is approximately proportional to the number of the core fuel assemblies,

d) The change in the above proportional coefficient due to the burn-up of the core fuel can be neglected.

From the above findings, we decided to use the following formula to calculate the control rod worth ρ (X, Y, Z) with $X\%$ of ^{10}B burn-up in a reactor system where the number of core fuel assemblies is Y and the average core burn-up is Z MWD/T:

$$\rho(X,Y,Z)=\rho(X_0,Y_0,Z_0)\cdot\{1+a(X-X_0)\} \times\{1+b(Y-Y_0)\}\{1+c(Z-Z_0)\}, \tag{3.1}$$

where

$\rho(X_0,Y_0,Z_0)$: The control rod worth in the reference core system

X_0, Y_0, Z_0 : The degree of burn-up of ^{10}B, number of core fuel assemblies, and the average degree of core burn-up in the reference system

a : Coefficient for the changes caused by burn-up of ^{10}B

b : Coefficient for the changes caused by the changes in the number of fuel assemblies

c : Coefficient for the changes caused by the burn-up of the core.

From the fact that the increases in the worth due to the core burn-up are almost offset by the decreases in the worth due to the increases in the number of the core fuel assemblies, the changes in the worth with the reactor operation are estimated to be 4% at most from the worth in the initial core. Therefore, we judged that the worth after the reactor operation can be estimated with a sufficient accuracy by use of Eq. (3.1) provided that the control rod worth in the initial core system has been obtained accurately by the characteristics test or the detailed calculation.

It is well known that when a plural number of control rods have been inserted simultaneously the reactivity worth differs from the simple sum of the individual control rod worths due to the mutual interference effect. It is necessary to take into account the interference effect when obtaining the reactivity worth in the case where a plural number of control rods are inserted.

In the case of the JOYO core, the interference effect is largest when two regulating rods are inserted simultaneously and the reactivity worth at this time is about 5% larger than the simple sum of the worths of the two regulating rods. That is to say that this value is about 0.05 if the interference effect coefficient C is defined by the following formula:

$$C=\frac{\rho_{12}-(\rho_1+\rho_2)}{\rho_1+\rho_2} \tag{3.2}$$

It has been confirmed that the changes in this interference effect coefficient due to the reactor operation can be neglected. Therefore, we decided to use the interference effect coefficient in the initial core through all over the cycles.

3. 1. 2 Effective multiplication factor

TABLE 3. 1 shows the changes in the effective multiplication factor due to the core burn-up. From this, it is seen that

a) The changes in the effective multiplication factor due to the core burn-up are approximately proportional to the changes in the integrated reactor power and the average burn-up of the core.

b) The above proportional coefficient does not vary greatly with the number of core fuel assemblies and core burn-up condition.

TABLE 3. 2 shows the changes in the effective multiplication factor due to the changes in the number of core fuel assemblies.

From this, it is seen that the effective multiplication factor increases by about 0.30% ΔK to 0.35% ΔK when one fuel assembly is added to the fifth row of the core fuel (to replace a blanket assembly). That is to say that if the change in the effective multiplication factor ΔK

TABLE 3. 1 K_{eff} change due to burn-up

Operation cycle No.	No. of core assemblies	Integrated reactor power in each cycle (MWD)	K_{eff} change (% $\Delta K/K$)
1	68	4,500	0.58
2	70	3,150	0.48
3	71	〃	0.47
4	72	〃	0.46
5	73	〃	0.46
6	74	〃	0.46
7	75	〃	0.46
8	76	〃	0.45

TABLE 3. 2 K_{eff} change due to addition of core assemblies

No. of core assembly before change	No. of added assemblies	K_{eff} change (% ΔK)	Reactivity worth of one added assembly (% ΔK)
65	2	0.705	0.353
	4	1.360	0.340
	7	2.316	0.331
	12	3.745	0.312
67	2	0.655	0.328
	5	1.611	0.322
	10	3.040	0.304
69	3	0.956	0.319
	8	2.385	0.299
72	5	1.429	0.286

(ΔN) due to the addition of ΔN number of core fuel assemblies is expressed by the following formula :

$$\Delta K(\Delta N)=\rho_F\cdot\Delta N. \qquad (3.3)$$

ρ_F changes slightly depending on the initial number of the core fuel assemblies and the number of added assemblies. However, since the changes in ρ_F is relatively small and its value was only about 0.3% ΔK, we judged that the change in ρ_F can be neglected.

TABLE 3. 3 shows the comparison of the amount of change in the effective multiplication factor that was estimated by use of the equation (3.3), neglecting the changes in ρ_F and the value obtained by detailed calculation. From this, it can be said that even if the changes in ρ_F are neglected the changes in the effective multiplication factor due to the changes in the number of core fuel assemblies can be estimated with so high accuracy of about $\pm 0.1\%$ ΔK that there is virtually no problem in this respect. The changes in ρ_F with the core burn-up are also sufficiently small as shown in TABLE 3. 4.

From the above findings, we judged that the effective multiplication factor K in the reactor system after operation can be obtained with sufficient acccuracy by use of the following formula.

$$K=K_0-\alpha'(B-B_0)+\rho_F(N-N_0), \qquad (3.4)$$

K_0 : Effective multiplication factor in the reference core system

B_0 : Average core burn-up in the reference core system

N_0 : Number of core fuel assemblies in the reference core system

α' : Coefficient for the changes due to the core burn-up

As we discussed in § 3. 3. 1, it was estimated that the changes occurring in the control rod

TABLE 3. 3 Comparison of reactivity change calculated by SMART method and 2-dimensional diffusion code

No. of core assemblies before change	No. of added assemblies	SMART (% ΔK)	Diffusion (% ΔK)	Difference (% ΔK)
65	2	0.624	0.705	−0.08
	4	1.248	1.360	−0.11
	7	2.184	2.316	−0.13
	12	3.745	3.745	—
67	2	0.608	0.655	−0.05
	5	1.520	1.611	−0.09
	10	3.040	3.040	—
69	3	0.894	0.956	−0.06
	8	2.385	2.385	—

TABLE 3. 4 Change of reactivity worth of added assembly due to burn-up (based on the core of 67 assemblies)

Core average burn-up (MWD/T)	Reactivity worth of one added assembly (% ΔK)
0	0.308
10,000	0.308
20,000	0.309
30,000	0.309

worth with the reactor operation are small and that the control rod worth in the core system after operation can be calculated with a relatively high degree of accuracy. The control rod positions when the initial criticality was reached in each operation cycle are kept in the form of operation records.

On this case, when making the record calculations, it is possible to semi-experimentally determine the effective multiplication factor by use of the measured value of the control rod position and the calculated value of the control rod worth. In the SMART code, the above-mentioned semi-experimentally obtained effective multiplication factor is compared with the calculated value obtained by use of Eq. (3.4).

When several cycles of operation have been completed, the results of the above comparison can be used for the correction of the coefficients in Eq. (3.4) and for the improvement of prediction calculation accuracy.

3. 1. 3 Reactivity coefficients

The power coefficient and isothermal coefficient can be calculated from Doppler coefficient, mass coefficient (fuel material, structural material, coolant), and shape coefficient. We evaluated the changes with the reactor operation in Doppler, mass and shape coefficient and by use of the design data, and we reached the following conclusion.

a) The changes in the above coefficients with the reactor operation (due to the changes in the core burn-up and in the number of core fuel assemblies) are small as compared with the width of uncertainty that is set in the design.

b) The above changes can be approximated as the linear function of the average core burn-up and the number of core fuel assemblies.

Figures 3. 4 and 3. 5 show these relations.

From the above, for the core composition of the average core burn-up B_c, radial blanket region

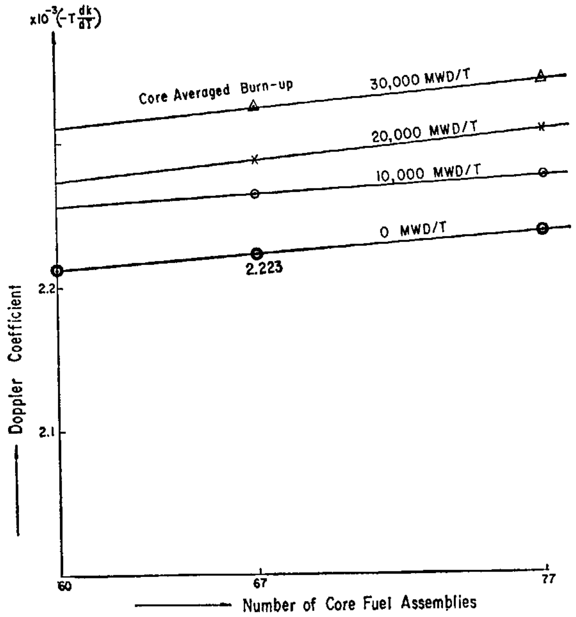


Fig. 3. 4 Core Doppler coefficient vs. number of core assemblies.

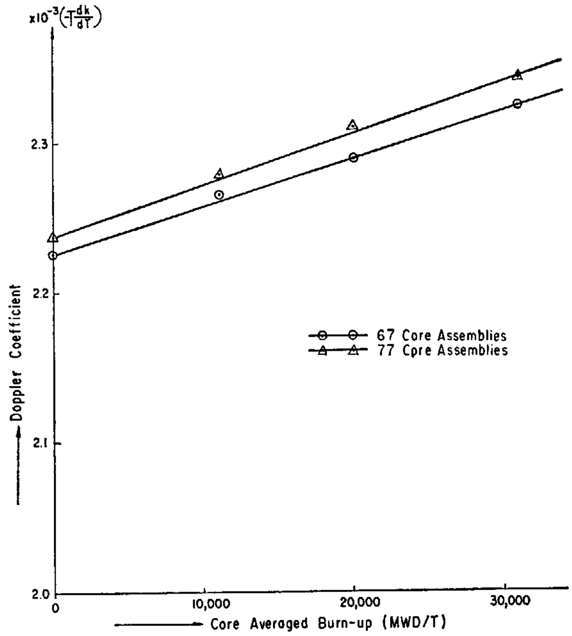


Fig. 3. 5 Core Doppler coefficient vs. core burn-up.

average burn-up B_b, axial blanket average burn-up B_a, and the number of core fuel assemblies N, (mass coefficient, Doppler coefficient, shape coefficient) RC can be calculated by the following formula :

$$RC=RC_0\{1+a(N-N_0)+b(B_c-B_c^\circ) +c(B_b-B_b^\circ)+d(B_a-B_a^\circ)\}, \tag{3.5}$$

where

RC_0 : Reactivity coefficient of the reference system

N_0 : Number of core fuel assemblies of reference system

B_c°, B_b°, B_a° : Regional average burn-up of reference system.

The reactivity coefficient changes so little with the change of burn-up and the number of core fuel assemblies that the reactivity coefficient of the system after operation will be estimated by the equation (3. 5) with satisfactory accuracy, provided that RC_0 of the reference system has been correctly obtained by the characteristics test or detailed calculation.

3. 1. 4 Neutron flux distribution

3. 1. 4. 1 Basic calculation formula

The following things are conceivable as the main causes of the changes occurring in the relative distribution of neutron flux with the reactor operation.

a) The changes occurring with the core burn-up (change in the core composition)

b) The changes occurring in the core arrangement (core system) due to the fuel exchange

c) The changes occurring in the control rod insertion depth

In the case of fast reactors, the space distribution of neutron flux and energy spectrum do not change so much with burn-up and it is believed that the burn-up characteristic can be obtained with a considerably high degree of accuracy by a calculation method (the constant neutron flux method of burn-up calculation) in which the neutron flux distribution and microscopic cross section are kept constant. It was predicted that the changes in the neutron flux distribution with burn-up would be small particularly in such small reactor core as JOYO in which enrichment is so high and yet the degree of burn-up is not so high. In order to confirm this prediction, we calculated the burn-up characteristics by use of the above mentioned constant neutron flux method of burn-up calculation and compared thus obtained value with the one which was obtained by the ordinary method of burn-up calculation, and reached the following conclusion.

a) The changes occurring in the relative distribution of neutron flux with burn-up (changes in the fuel composition) are so small that they may be neglected.

b) The changes occurring in the power distribution with burn-up are largely ascribable to the changes in the density of fission elements.

Therefore, when calculating the neutron flux distribution by the subsystem SMART, we decided to take into consideration the effect of the changes in the core arrangement and also the effect of the changes in the control rod position, neglecting the changes caused by burn-up.

In order to grasp the changes occurring in the relative distribution of neutron flux with the changes in the core arrangement and in the control rod position, we decided to express the space distribution of neutron flux and the energy distribution by the following three kinds of distribution coefficients.

(1) Radial distribution coefficient

The total energy neutron flux density in the zone equivalent to the core height in the core center is standarized at 1.0 and the total energy neutron flux density in the zone equivalent to the core height in various positions within the core is taken as the radial distribution coefficient, that is

$\phi_g(n, m)$……The neutron flux density of the g-th group of the radial No. n and axial No. m

$V(m)$ ………The volume of the m-th axial node.

The above "radial No." is the number that is given to the positions in which the core component elements are loaded as illustrated in **Fig. 3. 6.** The "axial No." is the number given to the zones into which the core component elements are divided axially. These two numbers are used to indicate the various positions within the core.

Radial distribution coefficient $\phi_R(n)$ of the radial No. n is defined as follows.

$$\phi_R(n)=\frac{\sum\limits_{n=NDABL}^{NDCOR}\ \sum\limits_{g=1}^{IG}\ \phi_g(n,\ m)\cdot V(m)}{\sum\limits_{n=NDABL}^{NDCOR}\ \sum\limits_{g=1}^{IG}\ \phi_g(1,\ m)\cdot V(m)}. \tag{3.6}$$

(2) Axial distribution coefficient

The axial distribution coefficient represents the total energy neutron flux density at each axial node when the total energy neutron flux density in the zone equivalent to the core height is standardized at 1.0. That is, the axial distribution coefficient $\{\phi_z(n,\ l), l=1,\ NDMAX\}$ of the radial No. n is defined by the following formula:

$$\phi_z(n,\ l)=\frac{\sum\limits_{g=1}^{IG}\ \phi_g(n,\ l)\cdot V(l)}{\sum\limits_{m=NDABL}^{NDCOR}\ \sum\limits_{g=1}^{IG}\ \phi_g(n,\ m)\cdot V(m)}. \tag{3.7}$$

(3) Spectrum coefficient

The spectrum coefficient $\{\phi_s(n,\ m,\ i),\ i=1,\ IG\}$ is defined by the following formula:

$$\phi_s(n,\ m,\ i)=\frac{\phi_i(n,\ m)}{\sum\limits_{g=1}^{IG}\ \phi_g(n,\ m)} \tag{3.8}$$

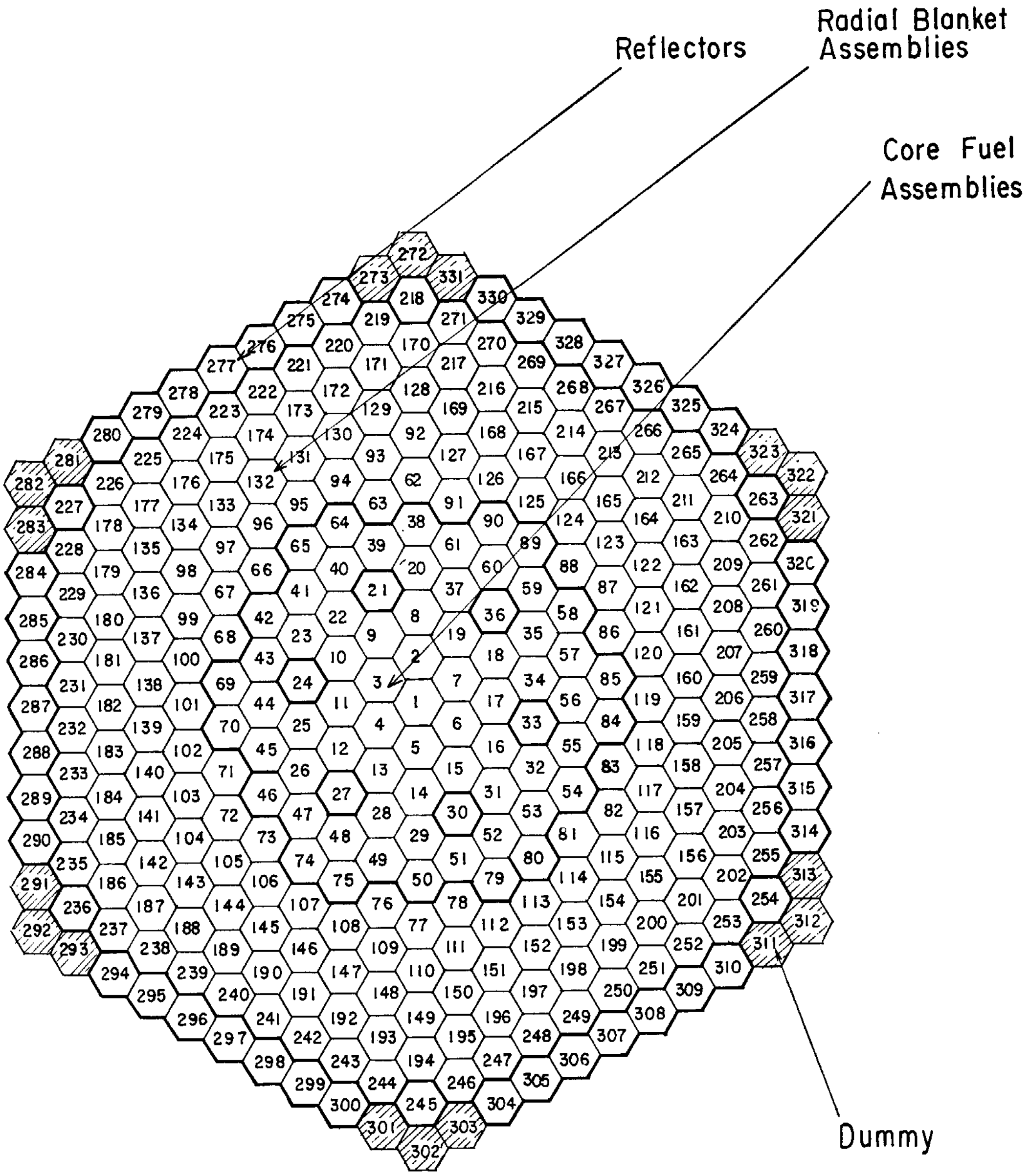


Fig. 3. 6 Core matrix and numbering.

No. 36, No. 27	Regulation rods
No. 21, No. 24, No. 30, No. 33	Safety rods

By using the above distribution coefficients, the neutron flux density in an arbitrary position can be expressed as follows:

$$\phi_g(n, m)=C\phi_R(n)\cdot\phi_Z(n, m)\cdot\phi_s(n, m, g), \tag{3.9}$$

C: Normalization constant.

3. 1. 4. 2 Changes in the neutron flux distribution with the insertion of control rod

The changes occurring in the neutron flux distribution with the insertion of control rods were obtained by making two dimensional x–y and r–z calculations.

(1) Changes in the radial distribution coefficient

Figures 3. 7 and 3. 8 show the changes occurring in the radial distribution coefficient with the insertion of control rods. From these figures, it is seen that the radial distribution coefficient changes as described below.

a) It decreases by 10% to 20% near the control rod insertion position.

b) It increases by 5% to 10% on the opposite side of the control rod insertion position.

c) It changes in almost all positions, not only near the control rod insertion position.

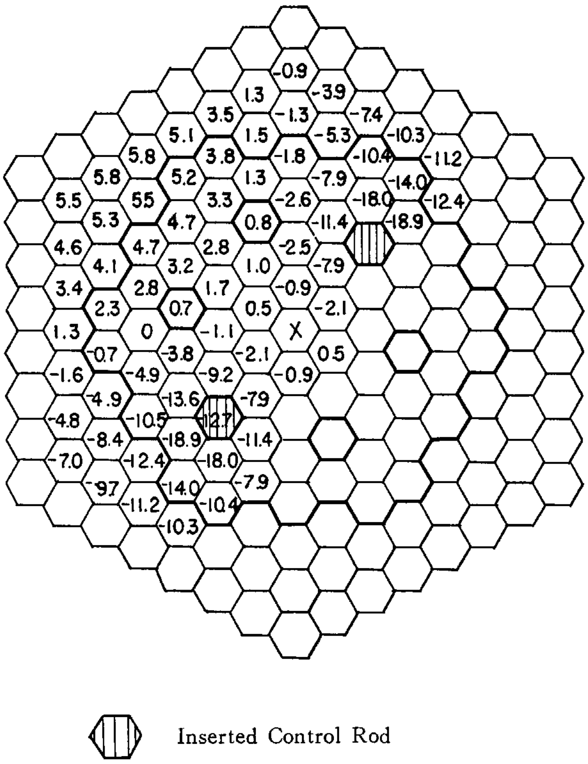


Fig. 3. 7 Change of radial coefficients of neutron flux due to control rods insertion (% unit).

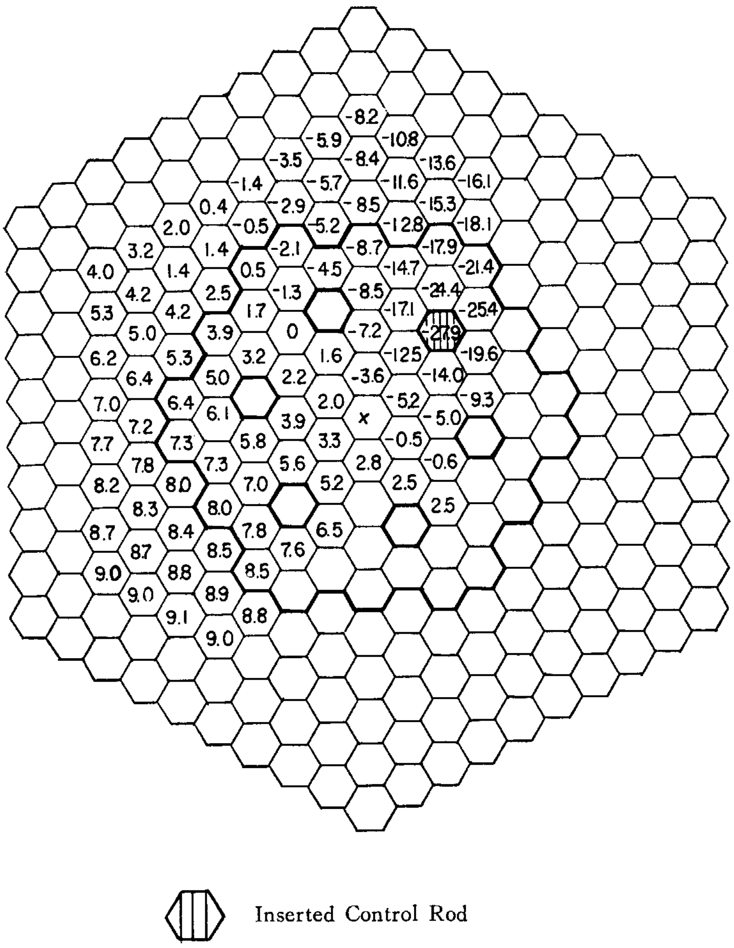


Fig. 3. 8 Change of radial coefficients of neutron flux due to control rod insertion (% unit).

We obtained the amounts of the above changes in the radial distribution coefficient in the system after burn-up (with 72 core fuel assemblies and 20,000 MWD/T of average core burn-up) and thus obtained results well agreed with the values shown in **Figs. 3.7 and 3.8.** Therefore, it is estimated that the changes occurring in the radial distribution coefficient with the insertion of control rods do not depend on the core burn-up and the size of the core system.

(2) Changes in spectrum coefficient

In order to investigate the effect of the changes in the spectrum coefficient on the calculation of the power density and the uranium and plutonium balances due to burn-up, we used the spectrum coefficients at the time of control rod withdrawal and insertion to make calculations of the following quantities and compared thus obtained values.

Fission reaction P_f:

$$P_f=\sum_{i=1}^{IG} (\Sigma_f)_i\,\phi_i. \tag{3.10}$$

^{235}U, absorption reaction A_{25}:

$$A_{25}=\sum_{i=1}^{IG} (N\sigma_a)_i^{25}\cdot\phi_i. \tag{3.11}$$

^{238}U capture reaction:

$$C_{28}=\sum_{i=1}^{IG} (N\sigma_c)_i^{28}\cdot\phi_i. \tag{3.12}$$

^{239}Pu absorption reaction:

$$A_{49}=\sum_{i=1}^{IG} (N\sigma_a)_i^{49}\cdot\phi_i. \tag{3.13}$$

Ratio of neutron flux over 0.1 MeV, $PHI_{0.1}$:

$$PHI_{0.1}=\sum_{i=1}^{3}\phi_i, \tag{3.14}$$

ϕ_i: Spectrum coefficient.

If the above quantities, P_f, A_{25}, C_{28}, A_{49} and $PHI_{0.1}$, which have been calculated by use of the spectrum coefficients at the time of control rod withdrawal and insertion are the same, the same results will be obtained by the use of either spectrum coefficient as far as the power density, the U and Pu balance, integrated irradiation dose ($E\geq 0.1$ MeV) are concerned and therefore there is no need of considering the changes in the spectrum coefficient with the insertion of control rods.

The results of comparison are shown in TABLES 3.5 through 3.7. From these tables, it can be said that the errors arising from neglecting the changes occurring in the spectrum coefficient with the insertion of control rods are as much as shown below and therefore it is necessary to consider the spectrum coefficient changes only at the case of the location adjacent to the control rods.

a) In the position near the control rod

Nuclear fission reaction	3% overestimated
^{235}U absorption reaction	7% overestimated
^{238}U capture reaction	8% overestimated
^{239}Pu absorption reaction	4% overestimated
Integrated irradiation dose	5% underestimated

b) In the position one assembly away from the control rod

Nuclear fission reaction	2% overestimated
^{235}U absorption reaction	4% overestimated
^{238}U capture reaction	4% overestimated
^{239}Pu absorption reaction	2% overestimated
Integrated irradiation dose	2% underestimated

TABLE 3. 5 Change of spectrum coefficients of neutron flux due to control rods insertion (1)
—For the assemblies adjacent to control rod—

Location	P_f	A_{25}	A_{49}	C_{28}	$PHI_{0.1}$
18	5.181 5.032 −2.9	3.319 3.102 −6.5	2.249 2.156 −4.1	1.114 1.032 −7.4	0.7273 0.7581 +4.2
19	5.170 5.030 −2.7	3.300 3.098 −6.1	2.242 2.155 −4.1	1.107 1.030 −7.0	0.7304 0.7589 +3.9
35	5.224 5.076 −2.8	3.395 3.178 −6.4	2.278 2.186 −4.0	1.146 1.062 −7.3	0.7136 0.7450 +4.4
37	5.200 5.055 −2.8	3.354 3.142 −6.3	2.262 2.171 −4.0	1.129 1.048 −7.2	0.7204 0.7510 +4.2
59	5.298 5.141 −3.0	3.521 3.296 −6.4	2.330 2.234 −4.1	1.193 1.107 −7.2	0.6948 0.7274 +4.7
60	5.260 5.095 −3.1	3.454 3.215 −6.9	2.302 2.200 −4.4	1.169 1.077 −7.9	0.7040 −0.7386 +4.9

Note : Followings are the same in TABLEs 3. 5 through 3. 7.
(1) " Location " means position of assemblies shown in Fig. 3. 6.
(2) In each Location, three columns mean as follows :
1st column : values of all control rods " out ",
2nd column : values of two regulating rods "in" (Location No. 27, 36),
3rd column : fraction of difference : (in-out)/out (%).
(3) Each value except $PHI_{0.1}$ was multiplied by the factor 1×10^3.

TABLE 3. 6 Change of spectrum coefficients of neutron flux due to control rods insertion (2)
—For the assemblies apart from control rod by the distance of one assembly pitch—

Location	P_f	A_{25}	A_{49}	C_{28}	$PHI_{0.1}$
7	5.139 5.045 −1.8	3.251 3.127 −3.8	2.221 2.166 −2.5	1.088 1.041 −4.3	0.7378 0.7518 +1.9
8	5.178 5.099 −1.5	3.314 3.203 −3.3	2.247 2.197 −2.2	1.113 1.072 −3.6	0.7278 0.7415 +1.9
34	5.204 5.120 −1.6	3.358 3.240 −3.5	2.264 2.211 −2.3	1.130 1.088 −3.7	0.7202 0.7347 +2.0
58	5.412 5.327 −1.6	3.706 3.588 −3.2	2.410 2.358 −2.2	1.260 1.216 −3.5	0.6701 0.6801 +1.5
61	5.287 5.207 −1.5	3.502 3.387 −3.3	2.322 2.272 −2.2	1.186 1.144 −3.5	0.6970 0.7126 +2.2
89	5.469 5.383 −1.6	3.790 3.672 −3.1	2.447 2.395 −2.1	1.289 1.246 −3.3	0.6583 0.6744 +2.4

TABLE 3.7 Change of spectrum coefficients of neutron flux due to control rod insertion (3)
—For the assemblies apart from control rod by the distance of two assembly pitch—

Location	P_f	A_{25}	A_{49}	C_{28}	$PHI_{0.1}$
1	5.129 5.073 −1.1	3.235 3.152 −2.6	2.215 2.178 −1.7	1.082 1.052 −2.8	0.7405 0.7508 +1.4
3	5.139 5.085 −1.1	3.251 3.171 −2.5	2.221 2.186 −1.1	1.088 1.059 −2.6	0.7378 0.7478 +1.4
10	5.178 5.120 −1.1	3.315 3.232 −2.5	2.247 2.209 −1.7	1.113 1.084 −2.6	0.7278 0.7363 +1.2
23	5.228 5.171 −1.1	3.400 3.322 −2.3	2.280 2.244 −1.6	1.147 1.121 −2.3	0.7133 0.7209 +1.1
42	5.422 5.372 −0.9	3.717 3.647 −1.9	2.416 2.384 −1.3	1.262 1.238 −1.9	0.6699 0.6766 +1.0
67	6.106 6.040 −1.1	4.714 4.632 −1.7	2.880 2.838 −1.5	1.587 1.563 −1.5	0.5616 0.5678 +1.1

c) In other positions, the errors of nuclear fission reaction, ^{235}U and ^{239}Pu absorption reactions and ^{238}U capture reaction are within ±1.5%, ±3% and ±3%, respectively.

(3) Method of compensating the control rod effect

From all the above results, it is considered that the neutron flux distribution at the time of control rod insertion can be obtained by making the following corrections.

(i) For radial distribution coefficient

"The ratio of changes in the radial distribution coefficient with the control rod insertion" which was calculated in the reference system is used without making any alteration to it. When there is not much disparity in position between two regulating rods:

$$\phi_R^{IN}(n) = \phi_R^{OUT}\{1+\beta_O^3(n)\cdot f(X_{av})\}. \tag{3.15}$$

When there is a great disparity in position of two regulating rods, it can be obtained by superposing the changes with the insertion of the individual regulating rods:

$$\phi_R^{IN}(n) = \phi_R^{OUT}\{1+\beta_O^1(n)\cdot f(X_1)\}\{1+\beta_O^2(n)\cdot f(X_2)\}, \tag{3.16}$$

where

$\phi_R^{IN}(n)$: Radial distribution coefficient at the time of control rod insertion and withdrawal

$\beta_O(n)$: Ratio of changes in the radial distribution coefficient with the control rod insertion (complete insertion)

$f(X)$: Function to indicate the relationship between the changes with the control rod insertion and the depth of control rod insertion X

X_1, X_2 : Regulating rod insertion depth

X_{av} : Average insertion depth of two regulating rods.

It has been confirmed that the changes occurring in the distribution when two control rods have been inserted can be obtained with satisfactorily high accuracy by the superposing of the changes occurring when the two regulating rods are inserted separately. In the case of JOYO, since two regulating rods are expected to be operated nearly in the same insertion position, we decided to use Eqs. (3. 15) and (3. 16).

(ii) For spectrum coefficient

The spectrum coefficient at only the location adjacent to the control rod is replaced with the "spectrum coefficient at the time of control rod insertion".

(iii) For the axial distribution coefficient

The inside of the core is divided into a suitable number of regions, paying attention to the relative positions with the regulating rods. The average axial distribution coefficients corresponding to 1/4, 1/2, 3/4 and complete insertion of regulating rod in this region are obtained beforehand. From these axial distributions, the axial distribution coefficient at an arbitrary insertion depth is estimated by interpolation and extrapolation.

3. 1. 4. 3 Changes in the neutron flux distribution with the addition of core fuel assemblies

We made investigations on the radial distribution coefficient and spectrum coefficients in various positions while varying the number of core fuel assemblies in the fifth row of the core, and found the following things.

a) The changes in the distribution coefficient with the changes in the core arrangement were so small in the rows, 0th to 3rd, that they can be neglected.

b) As for the row, 4th to 7th, the changes in the distribution coefficient in the position under observation can be neglected as far as there occurs no change in the state near the position.

c) The axial distribution coefficient changes little even when the core arrangement is changed.

d) The positions in the 4th to 7th rows of the core are divided into 78 different types according to the positions in the rows (the relative positions from the apexes of the hexagon formed with the rows) and the condition in the neighborhood (the numbers of the fuel assemblies near the position under observation and of the blanket assemblies).

The neutron flux distributions in the positions belonging to the same type well agree with one another.

e) The changes in the axial distribution coefficient and spectrum coefficient of the 9th and 10th rows are so small that it is necessary to consider only the changes in the radial distribution coefficient with the changs, in the number of core fuel assemblies in the 5th row.

Accordingly, we decided to use the following method to obtain the various distribution coefficients in the system after the core arrangement has been changed.

a) The distribution coefficients of the 0th to 3rd rows of the core are assumed to the same as those of the reference system, respectively.

b) As for the distribution coefficients of the 4th to 7th rows, only when some change has occurred in the condition near the position under observation, they are replaced with the distribution coefficients corresponding to the condition near the position under observation picked up from among the above-mentioned 78 kinds of distribution coefficients. As for the other positions, they are assumed to be the same as in the case of the reference system.

c) As for the 8 th and 9 th positions and the reflector position and storage rack position, the changes only in the radial distribution coefficient are considered, assuming there are no changes in the axial distribution coefficient and spectrum coefficient with the changes in the core arrangement.

The above-mentioned 78 kinds of distribution coefficients can be determined by making a calculation of the neutron flux distribution of the system containing 70 core fuel assemblies as shown in **Fig. 3. 9.**

As for the neutron flux distribution in the positions outside the blanket (the storage rack position), the changes due to the addition of core fuel assemblies will be considered as described

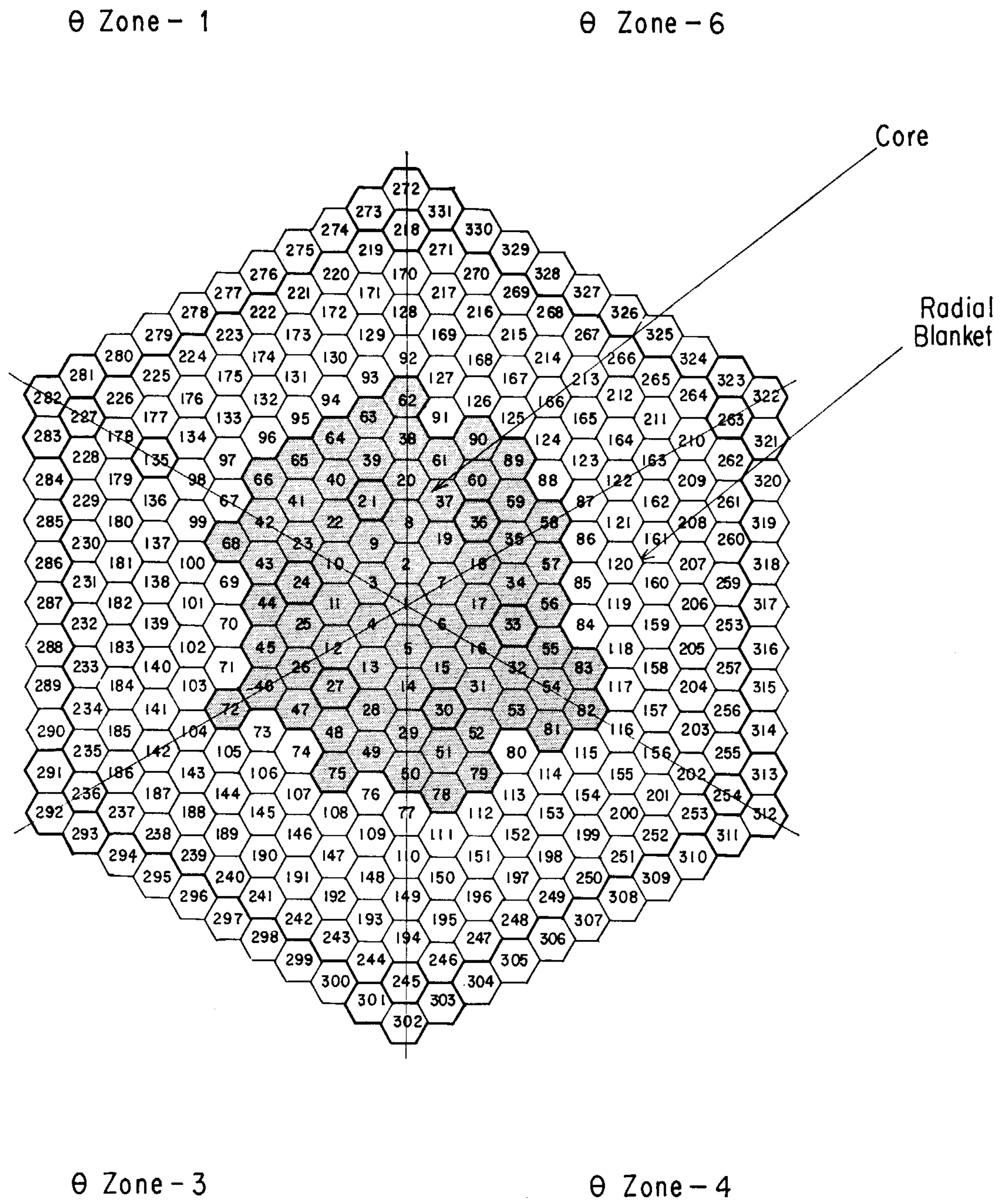


Fig. 3. 9 Special core configuration for the calculation of typical flux distribution in row 4 to row 7.

below.

a) The peripheral zone of the core is divided into six regions as shown in **Fig. 3. 9** and the rate of changes in the radial distribution coefficient with the addition of core fuel assemblies are assumed to be constant within the above regions.

b) It is assumed that there are no changes in the spectrum and axial distribution coefficients and are equal to those of the reference system.

c) The radial distribution coefficient $\phi(n)$ when ΔM number of core fuel assemblies have been added in the above region is approximated by the following formula :

$$\phi(n)=\phi_0(n)\cdot(1+\delta\cdot\Delta M)\cdot(1+\gamma\cdot\Delta N), \tag{3.17}$$

where

$\phi_0(n)$: Radial distribution coefficient in the reference system

δ : Coefficient to indicate the changes with addition of core fuel assemblies

ΔN: Amount of change in the number of fuel assemblies in the storage rack within the region θ

γ: Coefficient to indicate the changes caused by the change in the number of fuel assemblies in the storage rack.

3. 1. 4. 4 Accuracy of predicted values

We predicted the radial distribution and spectrum coefficients after the core system has changed, using the following procedure, and compared thus obtained values with the those obtained by the detailed calculation method (2-dimensional x-y calculation) to assess the accuracy of the predicted values.

a) The reference system initially has 67 fuel assemblies as illustrated in **Fig. 3. 6.** The distribution coefficient of this system is obtained by the 2-dimensional x-y calculation method.

b) The 2-dimensional x-y calculation method is used for the system shown in **Fig. 3. 9** to determine the 78 kinds of distribution coefficients.

c) The condition near the 4th to 7th row of the core of the system under observation is investigated. If the neighboring condition is found to have changed, the distribution coefficients corresponding to the changed neighboring condition are picked up from among the above-mentioned 78 kinds of distribution coefficients to replace.

The above prediction was made for the following system.

a) A system after 900 days of operation with a power of 50MW with the number of core fuel assemblies kept unchanged.

b) A system after 150 days of operation with a power of 50MW. The radial blanket assemblies, No. 71, No. 81 and No. 91, were replaced with core fuel assemblies according to a prescribed fuel changing program.

c) A system after 320 days of operation with a power of 50MW. Fuel assemblies were added in the radial positions, No. 71, No. 76, No. 81, No. 86 and No. 91. There were 72 core fuel assemblies in total.

TABLES 3. 8 and 3. 9 show the results of comparison of the radial distribution coefficients. In TABLE 3. 9, marked with * are the positions whose neighboring condition changed due to the changes in the core arrangement.

TABLES 3. 10 through 3. 12 show the values of the spectrum coefficients predicted by the simplified method and the calculated values obtained by the detailed method in the case of a system after 320 days of operation with a power of 50MW.

As was discussed in the section under the heading of "Changes in the spectrum coefficient with the control rod insertion", we assessed the accuracy of spectrum coefficients by comparing

TABLE 3. 8 Accuracy of radial coefficient of neutron flux estimated by SMART method (1) —Core inner region

Row No.	Location No.	SMART	x-y diffusional		
			67 ass'y core 50 MW×900 day	70 ass'y core 50 MW×150 day	72 ass'y core 50 MW×320 day
1	2	0.9598	0.9612	0.9617	0.9601
	4	0.9595	0.9609	0.9613	0.9614
2	8	0.8531	0.8583	0.8605	0.8599
	11	0.8813	0.8853	0.8865	0.8842
	15	0.8812	0.8855	0.8869	0.8958
3	20	0.7279	0.7386	0.7441	0.7436
	31	0.7758	0.7837	0.7882	0.8001
	34	0.7754	0.7833	0.7831	0.7983
	25	0.7752	0.7828	0.7870	0.7867

TABLE 3. 9 Accuracy of radial coefficient of neutron flux estimated by SMART method (2) —Core outer region

Row No.	Location No.		67 ass'y core 50 MW×900 day	70 ass'y core 50 MW×150 day	72 ass'y core 50 MW×320 day
4	38	D	0.5752	0.5854	0.5862
		S	0.5615	0.5615	0.5652*
	39	D	0.6943	0.6527	0.6510
		S	0.6377	0.6377	0.6377
	58	D	0.5744	0.5707	0.5953*
		S	0.5605	0.5605	0.5652
	59	D	0.6497	0.6502	0.6649
		S	0.6379	0.6379	0.6379
	61	D	0.6457	0.6559	0.6592
		S	0.6337	0.6747*	0.6747*
	57	D	0.6450	0.6421	0.6662
		S	0.6330	0.6330	0.6747*
5	64	D	0.5158	0.5149	0.5135
		S	0.5016	0.5016	0.5016
	65	D	0.5145	0.5102	0.5085
		S	0.5002	0.5002	0.5002
	66	D	0.4658	0.4596	0.4581
		S	0.4505	0.4505	0.4505
	89	D	0.5162	0.5194	0.5313
		S	0.5018	0.5018	0.5018
	90	D	0.5149	0.5262	0.5329
		S	0.5005	0.5626*	0.5626*
	91	D	0.4669	0.4853	0.4888
		S	0.4513	0.5021*	0.5021*

Note: "D" means diffusion calculation.
"S" means SMART method.

TABLE 3. 10 Comparison of spectrum coefficient of neutron flux estimated by SMART method and diffusion calculation (1)—Core zone

Location No.	38	39	40	41	42	43
ϕ_1	1.122-1	1.165-1	1.212-1	1.182-1	1.062-1	1.168-1
	1.143-1	1.191-1	1.249-1	1.219-1	1.092-1	1.202-1
ϕ_2	2.448-1	2.498-1	2.543-1	2.515-1	2.379-1	2.498-1
	2.467-1	2.526-1	2.587-1	2.360-1	2.414-1	2.536-1
ϕ_3	3.207-1	3.224-1	3.216-1	3.208-1	3.204-1	3.219-1
	3.195-1	3.215-1	3.210-1	3.201-1	3.193-1	3.211-1
ϕ_4	2.664-1	2.590-1	2.541-1	2.585-1	2.742-1	2.592-1
	2.633-1	2.350-1	2.483-1	2.324-1	2.686-1	2.540-1
ϕ_5	4.682-2	4.429-2	4.206-2	4.341-2	5.028-2	4.426-2
	4.696-2	4.389-2	4.046-2	4.230-2	5.014-2	4.341-2
ϕ_6	9.026-3	7.913-3	6.844-3	7.548-3	1.111-2	7.977-3
	9.224-3	7.919-3	6.513-3	7.301-3	1.129-2	7.771-3
P_f	5.348-3	5.306-3	5.272-3	5.295-3	5.417-3	5.308-3
	5.356-3	5.306-3	5.257-3	5.285-3	5.422-3	5.301-3
A_{25}	3.619-3	3.349-3	3.487-3	3.528-3	3.728-3	3.550-3
	3.618-3	3.335-3	3.450-3	3.496-3	3.717-3	3.524-3
A_{49}	1.232-3	1.205-3	1.183-3	1.199-3	1.269-3	1.206-3
	1.229-3	1.198-3	1.167-3	1.184-3	1.262-3	1.194-3
C_{28}	2.368-3	2.333-3	2.312-3	2.330-3	2.416-3	2.340-3
	2.371-3	2.336-3	2.300-3	2.320-3	2.416-3	2.332-3
$PHI_{1\mathrm{MeV}}$	6.777-1	6.887-1	6.971-1	6.905-1	6.645-1	6.885-1
	6.505-1	6.932-1	7.046-1	6.980-1	6.699-1	6.949 1

Note: In each column, upper value is of diffusion calculation and lower is of SMART method. This is the same in TABLE 3. 12 and 3. 13.

TABLE 3. 11 Comparison of spectrum coefficient of neutron flux estimated by SMART method and diffusion calculation (2)—Blanket zone

Location No.	122	123	124	125	126	127
ϕ_1	4.285-2	5.068-2	6.332-2	7.404-2	7.394-2	6.236-2
	4.173-2	4.906 2	6.210-2	7.186-2	7.535-2	5.764-2
ϕ_2	1.507-1	1.662-1	1.841-1	1.973-1	1.965-1	1.802-1
	1.483-1	1.626-1	1.814-1	1.943-1	2.004-1	1.723-1
ϕ_3	2.967-1	3.037-1	3.108-1	3.124-1	3.114-1	3.069-1
	2.933-1	3.016-1	3.076-1	3.109-1	3.121-1	2.993-1
ϕ_4	3.665-1	3.311-1	3.321-1	3.180-1	3.190-1	3.365-1
	3.660-1	3.521-1	3.329-1	3.198-1	3.124-1	3.398-1
ϕ_6	9.971-2	9.028-2	8.057-2	7.374-2	7.423-2	8.304-2
	1.026-1	9.427-2	8.356-2	7.607-2	7.333-2	9.062-2
ϕ_6	4.354-2	3.602-2	2.906-2	2.449-2	2.486-2	3.101-2
	4.793-2	4.039-2	3.250-2	2.711-2	2.634-2	4.029-2
P_f	3.035-4	3.426-4	4.074-4	4.627-4	4.623-4	4.029-4
	2.994-4	3.356-4	4.022-4	4.521-4	4.705-4	3.813-4
A_{25}	8.424-5	7.500-5	7.189-5	6.775-5	6.506-5	7.331-3
	8.723-5	8.115-5	7.436-5	6.969-5	6.861-5	8.015-3
A_{49}	3.233-3	3.056-3	2.886-3	2.773-3	2.782-3	2.933-3
	3.322-3	3.149-3	2.960-3	2.529-3	2.802-3	3.129-3
C_{28}	3.545-3	3.362-3	3.168-3	3.031-3	3.040-3	3.216-3
	3.607-3	3.435-3	3.224-3	3.078-3	3.030-3	3.369-3
PHI_{1MeV}	4.903-1	5.226-1	5.582-1	5.837-1	3.818-1	5.495-1
	4.834-1	3.133-1	5.511-1	5.771-1	5.879-1	5.292-1

TABLE 3. 12 Comparison of spectrum coefficient of neutron flux estimated by SMART method and diffusion calculation (3)—Blanket near core zone

Location No.	92	93	94	95	96	121
ϕ_1	4.348-2	5.090-2	6.376-2	7.120-2	6.220-2	6.077-2
	4.175-2	4.906-2	6.338-2	7.150-2	6.233-2	5.764-2
ϕ_2	1.515-1	1.661-1	1.835-1	1.922-1	1.810-1	1.787-1
	1.483-1	1.626-1	1.826-1	1.934-1	1.817-1	1.723-1
ϕ_3	2.964-1	3.052-1	3.095-1	3.102-1	3.075-1	3.073-1
	2.933-1	3.016-1	3.075-1	3.094-1	3.066-1	2.993-1
ϕ_4	3.657-1	3.513-1	3.330-1	3.236-1	3.350-1	3.381-1
	3.660-1	3.521-1	3.315-1	3.205-1	3.317-1	3.398-1
ϕ_5	9.936-2	9.037-2	8.093-2	7.640-2	8.274-2	8.382-2
	1.026-1	9.427-2	8.282-2	7.680-2	8.385-2	9.062-2
ϕ_6	4.357-2	3.613-2	2.929-2	2.639-2	3.156-2	3.135-2
	4.795-2	4.039-2	3.208-2	2.812-2	3.387-2	4.029-2
P_f	3.069-4	3.438-4	4.098-4	4.483-4	4.023-4	3.946-4
	2.994-4	3.336-4	4.088-4	4.522-4	4.039-4	3.813-4
A_{25}	8.418-5	7.509-5	7.210-5	6.942-5	7.376-5	7.388-5
	8.723-5	8.115-5	7.397-5	7.040-5	7.517-5	8.013-5
A_{49}	3.232-3	3.053-3	2.892-3	2.819-3	2.941-3	2.942-3
	3.327-3	3.149-3	2.948-3	2.851-3	2.985-3	3.129-3
C_{28}	3.541-3	3.364-3	3.174-3	3.084-3	3.215-3	3.230-3
	3.607-3	3.433-3	3.210-3	3.095-3	3.236-3	3.369-3
PHI_{1MeV}	4.914-1	5.222-1	5.568-1	5.736-1	5.507-1	5.468-1
	4.834-1	5.133-1	5.535-1	5.746-1	5.506-1	5.292-1

the nuclear fission reaction (P_f) for a unit neutron flux, the ^{235}U absorption reaction (A_{25}), ^{238}U capture reaction (C_{28}) and ^{239}Pu absorption reaction (A_{49}).

From **TABLES 3. 8** through **3. 12**, the following things were found.

a) As for the 0 through 3rd rows of the core, the differences between predicted values and calculated ones were within $\pm 3\%$ in all the systems, suggesting a high degree of accuracy of the predicted values.

b) As for the 4th and 5th rows of the core fuel loaded positions, the difference between predicted and calculated values were about $\pm 5\%$ at most and less than $\pm 3\%$ in most of the positions.

c) As for the 5th and 6th blanket fuel loaded position, the differences between predicted and calculated values were about $\pm 7\%$ at most and less than $\pm 5\%$ in most of the positions.

d) In the 7th row, there were some cases where the differences between predicted and calculated values were so large as about 10% but less than several per cents as far as the maximum value of the radial distribution coefficient (the maximum in the 7th row) was concerned.

3. 1. 5 Power distribution and neutron irradiation dose

The method of calculating the power distribution and neutron irradiation dose are described below.

The amount of heat generated by gamma-rays within the blanket is so large in the case of JOYO that it cannot be neglected, therefore it is necessary to consider not only the heat generated by nuclear fission (neutron heating) but also the heat generated by gamma-rays.

(1) Amount of heat generated by neutrons $PN\ (m,\ n)$ in the position of radial position No. m and axial node No. n is calculated by the following formula:

$$PN(m,n)=\sum_{l=1}^{MMK}(E_f{}^n)^l\cdot\sum_{i=1}^{IG}(N\sigma_f)_i{}^l\phi_i(m,n), \tag{3.18}$$

where

$(E_f{}^n)^l$: Energy that is released by nuclear fission (excepting gamma-ray)

$\phi_i(m,n)$: Neutron flux density

l : Nuclide

i : Energy group.

According to results of the "measurement of reaction rate distribution for individual nuclides" made in the FCA mockup experiment, the energy $(E_f{}^n)^l$ released by nuclear fission is adjusted for individual nuclides and regions.

(2) Amount of heat generated by gamma-rays

As a result of examination of the gamma-ray heat distribution within the JOYO core, we reached the conclusion that it would be permissible to consider the gamma-ray transport effect only in the peripheral zone of the core such as the 5th and 6th rows of blankets and in other areas, it can be assumed that the generated gamma-rays are instantly turned into energy. Therefore, the density of heat generated by gamma-rays $PG(m,\ n)$ can be calculated by the following formula.

a) The density of heat generated by gamma-rays in the position where the gamma-ray transport effect can be neglected:

$$\begin{aligned}PG(m,n)=\sum_{l=1}^{MMK}\{&\sum_{i=1}^{IG}(E_f{}^\gamma)^l(N\sigma_f)_i{}^l\phi_i(m,n)\\&+\sum_{i=1}^{IG}(E_c{}^\gamma)^l(N\sigma_c)_i{}^l\phi_i(m,n)\\&+\sum_{i=1}^{IG}(E_{\text{in}}{}^\gamma)_i{}^l(N\sigma_{\text{in}})_i{}^l\phi_i(m,n)\}\end{aligned} \tag{3.19}$$

where

$(E_f^{\gamma})^l$: Energy generated by gamma-ray per fission of nuclide l

$(E_c^{\gamma})^l$: Energy generated by gamma-ray per capture reaction of nuclide l

$(E_{in}^{\gamma})_i^l$: Energy generated by gamma-ray per inelastic scattering reaction.

b) The amount of heat generated by gamma-ray in the blanket assemblies adjacent to the core fuel assemblies can be obtained by adding a value, which is obtained by multiplying the sum total of gamma-ray energy generated in adjacent core fuel assemblies by a definite ratio, to the value calculated by Eq. (3.19).

(3) Integrated irradiation dose

The neutron irradiation dose over 0.1MeV and 1MeV, that is, $PHI_{0.1}$, and $PHI_{1.0}$ are calculated by the following formulas:

$$PHI_{0.1}=\sum_{i=1}^{IG}\beta_i\phi_i\Delta t,$$

$$PHI_{1.0}=\sum_{i=1}^{IG}\alpha_i\phi_i\Delta t,$$

where

ϕ_i : Neutron flux density of the i-th group

Δt : Duration of irradiation

α_i : Coefficient to be used when calculating a neutron flux over 1MeV

β_i : Coefficient to be used when calculating a neutron flux over 0.1MeV.

Above α_i and β_i are the values that are determined by group collapsing and spectrum, and are to be obtained in advance by the detailed calculation method.

3.1.6 Flow distribution

This subsystem employs the following simplified calculation method.

a) In the calculation of pressure loss of the core elements, we do not use the pressure loss coefficients from the shapes of the various parts but use a total pressure coefficient beforehand.

The following is the formula for the calculation of pressure loss in each core component element:

$$\Delta P=C(T_0)(a_2T^2+a_1T+a_0)W^n+\Delta P_{\text{head}}, \tag{3.20}$$

where

$C(T_0)$: Pressure loss coefficient at temperature T_0

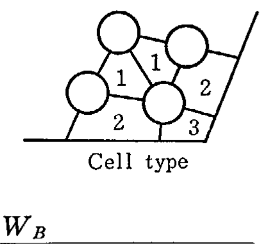
W : Weight flow rate

n : Exponent

a_2, a_1, a_0 : Coefficients of the fitting formula expressing the dependence of pressure loss on temperature, $a_2T_0^2+a_1T_0+a_0=1.0$

ΔP_{head} : Static pressure loss.

b) The flow rate in each subchannel in the assembly is obtained on the basis of the formula of pressure balance between the subchannels, that is, the flow rate in cell type 1 is obtained by the following formula:



Cell type

$$W_1=\frac{W_B}{n_1+n_2\left(\frac{A_2}{A_1}\right)\left(\frac{D_{e2}}{D_{e1}}\right)^a+n_3\left(\frac{A_3}{A_1}\right)\left(\frac{D_{e3}}{D_{e1}}\right)^a}, \tag{3.21}$$

where

W_B : Flow rate in the assembly

n_1, n_2, n_3 : Number of cells in the cell types 1, 2 and 3

A_1, A_2, A_3 : Subchannel cross sections of cell types 1, 2 and 3

D_{e1}, D_{e2}, D_{e3} : Equivalent diameters of the cell types 1, 2 and 3

a : Exponential parameter.

3. 1. 7 Thermal characteristics

As for the calculation of thermal characteristics, we used the simplified calculation method in the introduction of the coolant mixing effect, that is, we calculated the values of thermal characteristics without taking into consideration the mixing effect and then made corrections for the coolant mixing effect mainly for the following reasons.

a) At the present time there is no well-established method of calculation for the coolant mixing effect.

b) Calculation of thermal characteristics including the mixing effect takes computing time.

c) The feedback of experimental results is made easier by the introduction of the mixing effect as a correction factor.

The correction factor β of the mixing effect is defined by the following formula :

$$\beta=\frac{\Delta T_{\text{mix}}}{\Delta T_{\text{no·mix}}}, \tag{3.22}$$

where

$\Delta T_{\text{no·mix}}$: Coolant temperature rise when the mixing effect is not considered

ΔT_{mix} : Coolant temperature rise when the mixing effect is considered.

The fitting is possible for such β as the function of the radial peaking coefficient f_P for the assembly and the assembly flow rate W if there is no change in the shape of the assembly. Assuming that the coefficient of the fitting formula has been obtained beforehand by the detailed calculation method, the coolant temperature is calculated by the following formula :

$$\Delta T_{\text{mix}}=\Delta T_{\text{no·mix}}\cdot\beta, \tag{3.23}$$

where

$$\beta=\frac{A_0+A_1f_P+A_2f_P^2+A_3f_P^3+A_4f_P^4}{1+B_1f_P+B_2f_P^2+B_3f_P^3+B_4f_P^4} \times\frac{C_0+C_1W+C_2W^2+C_3W^3+C_4W^4}{1+D_1W+D_2W^2+D_3W^3+D_4W^4}. \tag{3.24}$$

3. 2 Composition

3. 2. 1 Composition

The subsystem SMART consists of the simplified calculation code SMART and the auxiliary codes RAND and EBIS-3. The auxiliary code RAND has the function of registering the composition of each assembly in the Assembly File. The Assembly File is a file on which the data concerning composition, burn-up, neutron irradiation dose, etc. of each core element are recorded. The simplified calculation code SMART reads those data from the file at the beginning of each cycle. Those data are written out to the another file (New Assembly File), after being modified for the changes of those by reactor operation. The auxiliary code RAND is so designed as to make up and revise the above-mentioned Assembly File.

The various interpolation and extrapolation formulas which were shown in § 3. 1, the characteristic values, group constants and physical properties of the reference system are recorded in a special file, called Detail Data File. The auxiliary code EBIS-3 has the function of revising the above-mentioned file with input data in the form of card. Therefore, the use of this auxiliary code EBIS-3 makes it easy to revise the interpolation and extrapolation formulas, the basic data and physical properties data. The code SMART is composed of the following subprograms.

a) Program for calculating the physical properties relating reactivity
   It is used to calculated the control rod worth, effective multiplication factor and reactivity coefficient at the beginning and end of operation cycle.
b) Program for the synthesis of three-dimensional neutron flux
   It is used to obtain the three-dimensional neutron flux distribution after the core pattern has changed.
c) Program for power distribution calculation
   It is used to calculate the neutron flux density and power density during the power operation by using the above-mentioned three-dimensional neutron flux distribution.
d) Program for making a power distribution for thermal calculation
   It is used to extract the hottest bundle in each orifice region and also to edit the power distribution into a format suited for thermal calculation.
e) Flow rate calculation program
   It is used to calculate the flow distribution in the core.
f) Thermal characteristics calculation program
   It is used to calculate the thermal characteristics of main core elements.
g) Burn-up calculation program
   It is used to calculated the integrated power and burn-up during operation cycle.
h) Safety confirmation program
   It is used to make judgement as to whether the characteristic values satisfy the design criteria or not, considering various uncertainties.
i) Long distance neutron flux calculation program
   It is used to calculate the thermal neutron flux at the start-up channel position considering the change of the neutron source intensity, and also to calculate the neutron flux density at the material irradiation rack position.
j) Print routine
   It is a routine to print out collectively the main results of calculations.
k) Input data processing program
   It is used for checking the input data and various files.
l) Calculation procedure determining program
   It is used to determine the required calculation items and procedure on the basis of the input data.
m) Program for making the operation records for simplified calculation
   It is used to record the various quantities (the ratio of calculation to measurement value, eg., coolant outlet temperature of assemblies, neutron flux at nuclear instrumentations etc.) required for the subsequent simplified calculation.

The configuration of the computer system used for making prediction calculations is shown in **Fig. 3. 10** and that for use in making record calculation is shown in **Fig. 3. 11.**

3. 2. 2 Input and output

The data of the atomic number density, integrated power, etc, in the last stage of the par-

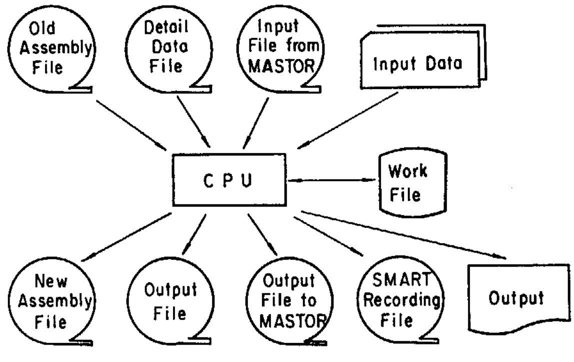


Fig. 3.10 Equipments structure of prediction calculation.

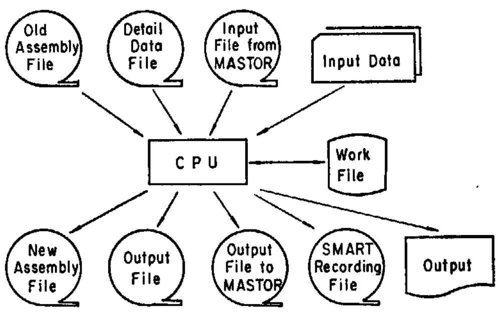


Fig. 3.11 Equipments structure of recording calculation.

ticular operation cycle are written out to the New Assembly File, which in turn serves as a file of assembies for input in the calculation for the following operation cycle. By repeating this process, it is possible to make prediction calculations for several cycles ahead. If the Assembly File obtained by the calculations for each cycle are kept, it will be possible to restart the prediction or record calculation from any cycle.

The SMART calculates so many kinds and handles very large quantities of data. It would be uneconomical and not necessary to print out such data at all times. Therefore, the SMART is so designed that all of the calculation results are once written to the Output File and only the data specified for printing are written in from the file for printing out.

The main input data are also registered on this Output File and it is so designed that when making prediction calculations for the next operation cycle it will be necessary to input very few data cards in order to change the data registered on this file. The use of this function makes it very easy to make up the card input data.

In the case of record calculation, the main input data are read in from Input File prepared by MASTOR recording subsystem and the main calculation results are written out not only to the Output File but also to the output file for MASTOR. The data are transferred to and from the recording subsystem through these input and output file.

The Record File of SMART is designed to correct, by using computer system, basic (detail) data or predicted value of simplified subsystem, if necessary. Therefore, this file contains selected data such measurement and calculation values as effective multiplication factor and coolant outlet temperature of assemblies to make corrections. The card input data for the simplified code SMART are relatively simple, because it is so designed that the atomic number density of each assembly is read in from the Assembly File and various fitting coefficients is read in from the Detail Data File.

The input data are of the following five kinds.

a) Calculation control data

b) Operation schedule

c) Specification of core arrangement

(b) and c) not necessary in record calculation)

d) Print option

e) Debug option (normally not necessary)

There is no need of such data as the specifications of mesh width and the number of mesh points that are required in ordinary nuclear calculations. It is possible to make up all input data only by determining the core arrangement and the operation schedule. The time required for prediction calculation or record calculation for one operation cycle is about 80 seconds when IBM 360/K195 computer is used.

4. Subsystem MASTOR

4. 1 General Description

This is a subsystem for recording the operating performance of JOYO. Its main functions are as follows.

a) To edit plant operation records and input/output data to the subsystem SMART.

b) To make the records of the movement of core elements

c) To sum up nuclear material weights and to make balance sheets

d) To edit data for post-irradiation test analysis

Therefore, most of the contents of the code concern the data control, input and output. The MASTOR does not function independently but is linked with the record calculation SMART. The flow of MASTOR and SMART comprises four steps as illustrated in **Fig. 4.1.**

STEP 1: The operation data file of JOYO from the on-line computer system is converted into a file for MASTOR.

STEP 2: The operation data and the movement data of the core elements are edited, checked and printed out. At the same time a data file for the calculation of characteristics to be delivered to SMART is made up and written out to the work file. The reactor power, coolant flow rate and temperature at the reactor inlet and outlet are plotted out.

STEP 3: This is a step for making calculations in SMART on the operation performance data obtained in STEP 2. The calculation results are written out to the work file.

STEP 4: The output from SMART is printed out and all data in the record subsystem are written to the master file, which in turn is referred to the record calculation in the next operation cycle and also is used for data retrieval in the analysis of post-irradiation tests.

4. 2 Contents of subsystem

4. 2. 1 Editing and recording of operation data

The editing of operation data is broadly classified into two groups; the data required for the calculation of the characteristic values of the core and those necessary for recording the plant history. The formers include the reactor power, the duration of operation at the power level, data concerning the coolant, and the control rod withdrawal position, etc. As for these data, different kinds of values, i.e. instantaneous values, average values and integrated values are required depending on the characteristics concerned. For example, when calculating the reactivity coefficients from the control rod reactivity worth and insertion depth, the instantaneous values at each power level are required. The average power during an operation cycle is required in the burn-up calculation and the information at the rated power is required in the calculation of thermal characteristics. The MASTOR processes the data from the on-line computer system to meet all such requirements. The data from the on-line computer system include the event codes representing the state of the plant at the time when the individual data were put out. The MASTOR identifies and select the characters of data by these codes, and treats operation data to be usable in the SMART.

As for the data of the plant history, there are the records of the days of actual operation, integrated power, and scrams and other changes from normal. The MASTOR makes up the records of such data on the basis of the output from the on-line computer system. The main operation data to be delivered to the SMART are shown in TABLE 4.1.

TABLE 4. 1 Operation data for SMART

Data	Contents
(1) Loading pattern of each core element	Identification No., location in the core and kinds
(2) History of the reactor power	· Start up and shut down days · Averaged reactor power and its duration time for every effective plant events
(3) Rated power conditions	Rated reactor power, coolant flow rate and temperature
(4) Control rods position	Withdraw length of each control rod at effective plant events and power levels
(5) Neutron detector read out value	Start up channels
(6) Coolant outlet temperature of fuel S/A	At rated power

4. 2. 2 Records of core element movement

The JOYO core elements comprise the core fuel assemblies, blanket fuel assemblies, control rods, reflectors and neutron source. They are carried out to the reprocessing plant or the material inspection facilities via the new fuel vault, the reactor core, the fuel storage rack in the core and the spent fuel storage pond. The MASTOR traces the history of movement of each core elements and it also displays the contents of those elements for each location.

4. 2. 3 Summing of nuclear material weights

This subsystem sums up the nuclear material weights in JOYO, and the increases and decreases in the weights from the preceding operation cycle are calculated and recorded. The nuclides to be traced are uranium and plutonium. Calculations are made of the amounts of their movement for each location and the amounts of change with burn-up in the core. The amounts of change with burn-up are calculated on the basis of the output from the SMART.

4. 2. 4 Supply of data for post-irradiation test

Of the data relating to the in-pile behavior, which are required in the irradiation test analysis, those which are supplied by the JOYPAC include (1) nominal characteristics data such as power of fuel pins and its distribution, coolant flow rate and temperature distribution, (2) the integrated values such as burn-up, neutron irradiation dose and their distributions, and (3) the operation history including the transient states such as start-up, shut down and scram. The MASTOR supplies these data in the printed form or filed on a magnetic tape. This is the information on each assembly during one cycle of operation, and is not capable of systematically editing the data for many cycles with regard to the assembly under observation. The each characteristic value is the representative one in the assembly estimated by the SMART and more detailed characteristics can be obtained by use of the detailed subsystem of JOYPAC system.

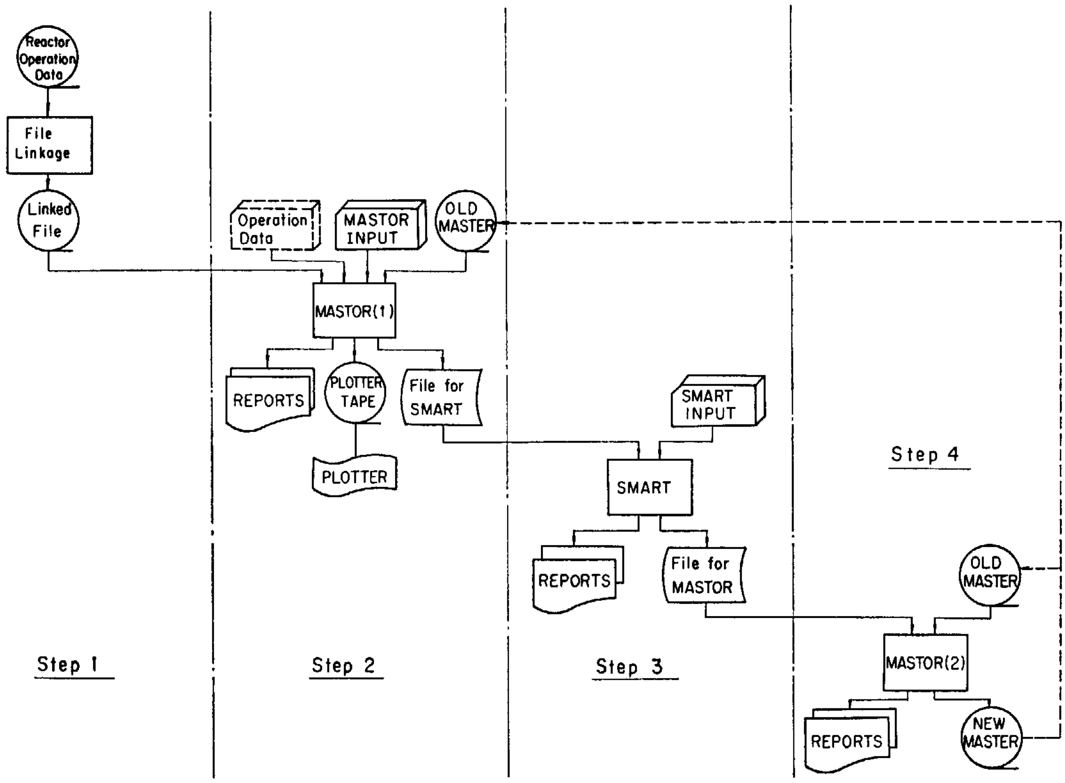


Fig. 4. 1 Schematic diagram of recording subsystem.

4. 3 Composition and input/output

The subsystem MASTOR is capable of executing the four steps, which were shown in **Fig. 4.1**, in a single set up, except the special processing in the initial run. In order to execute the SMART, the MASTOR is divided for Steps 1, 2 and 4, therefore its JCL (Job Control Language) and input data are divided into the following three parts.

(1) JCL and input data for Step 1 and Step 2
(2) JCL and SMART input data for Step 3
(3) JCL for Step 4

The input data are that of on-line operation and the core element movement in part (1) and the input card for option in part (2), so the input in the usual run is very simple. The operation data is in the form of magnetic tape alone when the on-line computer is operating normally. In the event of failure of the on-line computer system, it is possible for the operation data to be put in by cards alone or by both cards and tape. Each operation data is given the code indicating the character of the respective data (plant condition). They are the reactor start-up, shut-down, criticality, the reaching of the rated power, scram, restart in the event of computer failure, automatically periodic and operator's demand output and so forth. Only less than half of them are recorded automatically due to the sequence of the on-line computer system, then the operator demand is quite important in relation to the main core characteristics. Therefore, in order to cover the operator's miss operation, it is so designed that these codes may be added or corrected by card input. The samples of print output of MASTOR are shown in Appendix.

The MASTER FILE is an important input/output file in this subsystem and is devided in blocks as follows.

(1) The arrangement of core elements in the core, core storage rack and pond
(2) In-pile characteristic data of the core elements
(3) Data of assemblies in the new fuel vault
(4) Data of assemblies in the storage pond
(5) Data of assemblies in the core storage rack
(6) Data of surveillance test pieces
(7) Scram information
(8) Data of the stay time of each fuel assembly in the core
(9) Data of nuclear material weight
(10) On-line operation data

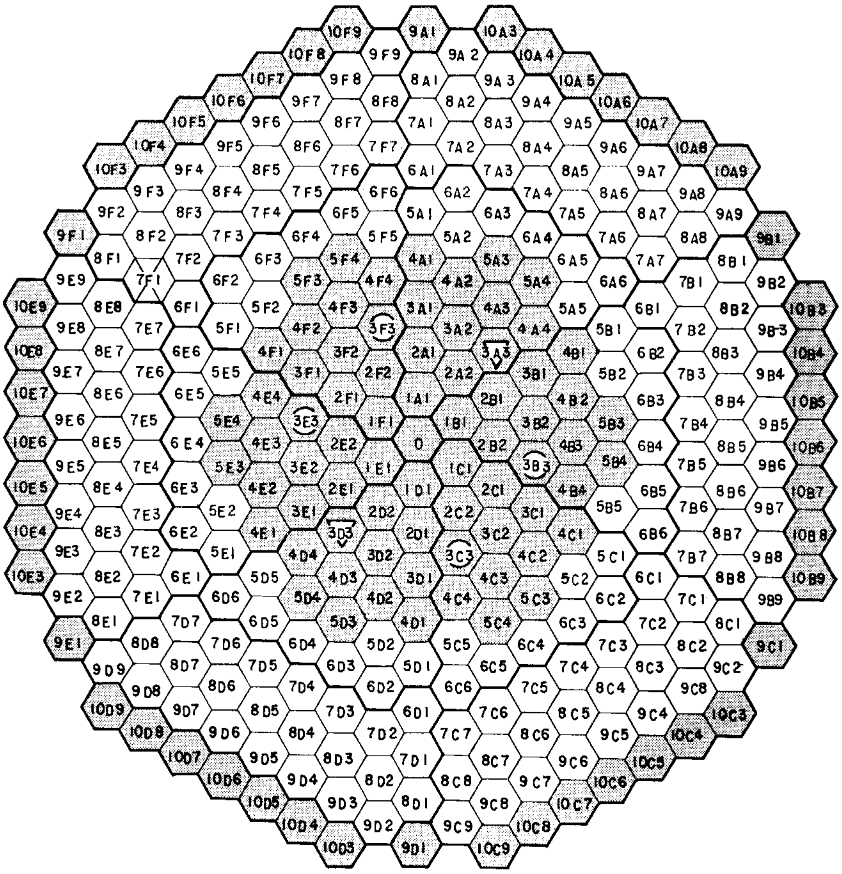
The plotter plots out such data as the reactor power, coolant flow rate and inlet/outlet temperatures throughout a cycle of operation.

5. Application of Code System for JOYO

5. 1 Information from plant system

5. 1. 1 General description of JOYO[3)]

JOYO is a sodium-cooled fast-flux experimental reactor and final thermal power of the plant is 100 MW. The core zone is composed of core fuel assemblies and, radial blanket assemblies and reflectors as illustrated in **Fig. 5. 1.** The reactivity control is conducted by the operation of



	Core Assembly	67
	Radial Blanket	191
	Reflector	48
	Safety Rod	4
	Shim Rod	2
	Neutron Source	1

Fig. 5. 1 JOYO core matrix.

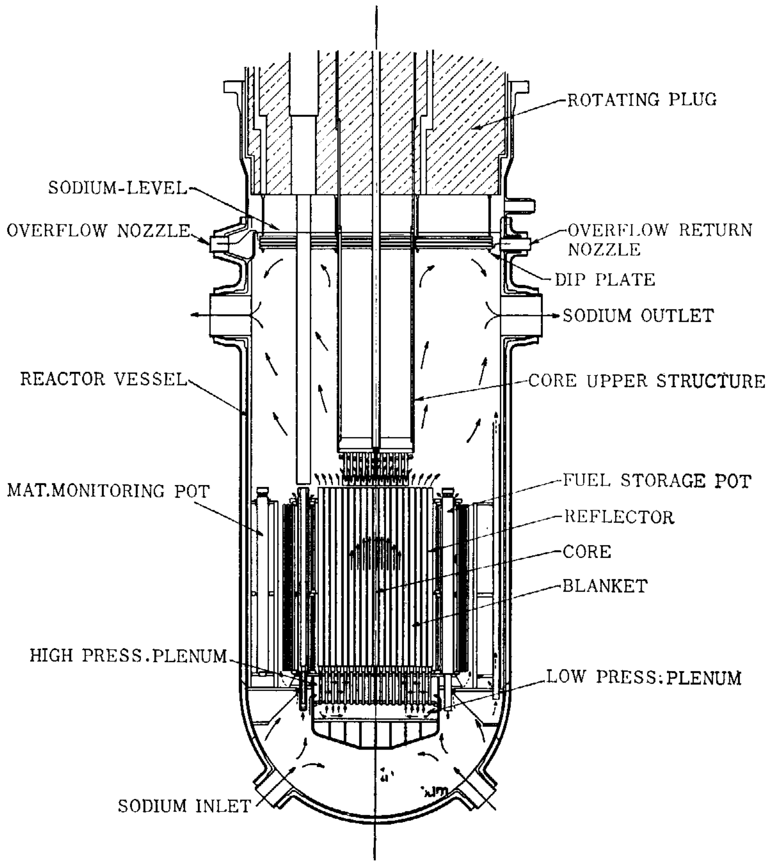


Fig. 5. 2 Sodium-flow in reactor vessel.

four safety rods and two regulating rods with B_4C pellets as absorbing material. The cooling system consists of the primary and the secondary, each comprising two loops. The heat of the primary system is transferred to the secondary system via the intermediate heat exchanger, where it is dispatched into the atmosphere by means of four main coolers. On the coolant flow in the reactor vessel, as shown in **Fig. 5. 2**, it flows into the high-pressure plenum through the inlet plenum and flows into the core fuel assemblies. Part of the coolant in the high-pressure plenum flows into the low-pressure plenum to cool the blanket assemblies, reflectors and control rods. The fuel assemblies are exchanged through the double rotating plugs on the top of the reactor vessel.

For the core fuel assemblies, an integral type is employed and the core stack length is 60 cm, and the axial blanket length is 40 cm for upper and lower, respectively. The assembly comprises 91 fuel pins with triangular pitch. The radial blanket assembly comprises 19 fuel pins with uranium oxides, and the stack length is 140 cm. The main parameters of the core and plant are shown in **TABLE 5. 1.** The above-mentioned core is called the MK-I core and is scheduled for about two years of operation and thereafter a core for irradiation called MK-II will be used. The change of the core will be made because MK-I is aimed mainly as simulating the typical fast reactor core but MK-II is so designed as to fully function as the irradiation bed for the fast reactor fuel and material. For this purpose, the core volume will be reduced and the core fuel assembly structure will be altered so that it will withstand a high power density and blanket assemblies will be replaced with reflectors.

5. 1. 2 Fuel exchange and operation cycle

One cycle of JOYO comprises 45 days of steady operation and 15 days of shut-down. The

TABLE 5.1 JOYO plant parameters

A. General	
1. Reactor type	Sodium cooled experimental fast reactor
2. Thermal power, MWt	50
B. Core	
1. Active core size, Height/Diam., m	0.6/0.73
2. No. of fuel assemblies, Core/Blanket	67/191
3. Fuel inventory, PuO_2/UO_2, te	0.15/0.7
4. Assembly pitch, cm	8.15
5. Power density, Av./Peak, kW/l	202/374
6. Max. burn up, MWD/te	25,000
C. Core fuel	
1. Material	PuO_2-UO_2
2. Enrichment, U^{235}/U, w/o $PuO_2/(PuO_2+UO_2)$, w/o	23 17.7
3. Form	Pellet
4. No. of pins per assembly	91
5. Pin diam., mm	6.3
6. Cladding material	SUS 316
7. Cladding thickness, mm	0.35
D. Reactor vessel	
1. Material	SUS 304
2. Shape	Cylindrical
3. Dimensions, Height/Diam., m	10/3.6 I.D.
4. Wall thickness, mm	25
E. Coolant	
1. Material	Sodium
2. Temperature, In/Out, °C	370/435
3. No. of primary loops	2
4. Flow rate/loop, ton/hr	1,100
G. Heat exchangers	
1. Type, IHX/Main	Shell and tube/Fin tube type dump heat exchanger
2. Number, IHX/Main	2/4
3. Tube materials IHX/Main	SUS 304/2.25 Cr-1Mo
H. Reactivity control	
1. No. of control rods, Safety/Regulating	4/2
2. Material	B_4C pellet
3. Drive mechanism	Motor
I. Refueling	
1. Method	Double rotating plugs
2. Shutdown period, months	2
J. Containment	
1. Type	Cylindrical semi-double wall
2. Dimensions, Height/Diam., m	54/28 I.D.

number of operation cycles a year is about five. The fuel exchange is of the scattered system based on the fuel inspection schedule. In accordance with the fuel inspection schedule, one to two of the core and blanket assemblies with different degrees of burn-up are extracted after each cycle operation to investigate their inpile behaviors by post-irradiation test. In order to compensate the decreases in reactivity with burn-up, blanket assemblies will be replaced with core fuel assemblies.

5.1.3 Reactor instrumentation and on-line computer

The instrumentation of the JOYO plant is roughly classified into the neutron instrumentation, process instrumentation, failed fuel detector system and radiation monitoring system, each linked with the safety protection system and on-line computer as required. In order to monitor the core performance, it is desirable to have as many detecting systems as possible, but the types and number of such detectors usable for this purpose are limited due to the high-temperature sodium atmosphere. The neutron detecting system consists of three systems, i.e. the start-up channels, intermediate channels and linear power channels, and it is located in between the reactor vessel and the safety vessel. The in-pile nuclear instrumentation is for use in the characteristic test at low power condition and such instruments capable of being used during normal operation are now in the development stage. The instruments for the coolant are mainly for measuring the flow rate, reactor vessel inlet and outlet temperatures and the fuel assembly outlet temperature. In order to grasp the inpile flow characteristics, it is necessary to measure the flow rate in each assembly and the plenum pressure. In JOYO, partial measurements will be made in the performance test stage before the normal operation.

The on-line computer system receives as input about 220 kinds of analog signals from the reactor plant, and consists of a central processing unit with a memory unit of 24 kilo-wards and peripheral equipments. Its composition is shown in **Fig. 5. 3.** This computer system performs the following functions.

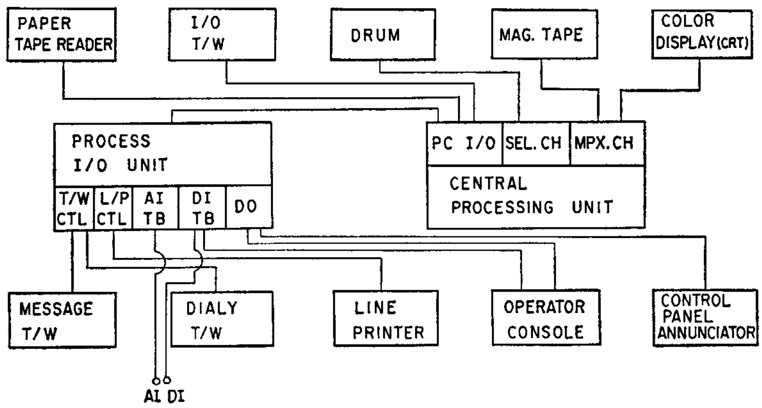


Fig. 5. 3 Schematic diagram of JOYO online computer.

a) Warning monitor…To issue a warning when the value of analog input signal has exceeded the preset upper or lower limits.
b) Event recall…To collect input data before and after an accident and output them.
c) Performance calculation…To calculate the reactor power by heat balance, watch for abnormal reactivity insertion, monitor radiation dose and to make operation diary.
d) Dialog with the operator…To supply data as requested by the operator.
e) Magnetic tape output…To supply operation data to the JOYPAC system.

As for the magnetic tape output, it selects only the data relating to the core performance, and the items of such data are shown in TABLE 5. 2. These data are put out every one hour of normal operation, at the reactor start-up and shut-down, and at important operation events. The upper and lower limit preset values of warning and abnormal reactivity calculation data are supplied for each operation cycle from the subsystem SMART of this code system.

Treatments have been made for the data from the reactor plant to prevent such erroneous signals as noise. Furthermore, the data are smoothed for the items important for the core characteristics by use of the following formula :

$$\hat{Y}_n=\alpha\hat{Y}_{n-1}+(1-\alpha)Y_n,$$

where

Y_n : Raw measured data

$\hat{Y}_n$: Smoothed data

TABLE 5. 2 Output parameters from on-line computer

Parameters	Contents
1. Operation code	Specific code name of this data block
2. Date and time	Output date and time
3. Reactor power	Heat balance, MW
4. Integrated power	Integrated reactor power from start up time of the operation cycle, MWD
5. Neutron detector output	Relative measured values of every 8 neutron detector chennels
6. Control rods position	Draw-out length of every 6 control rods, mm
7. Coolant inlet temperature	Coolant reactor inlet temperatures of every 2 loops, ℃
8. Coolant outlet temperature	Coolant reactor outlet temperatures of every 2 loops, ℃
9. Coolant flow rate	Primary coolant flow rates of every 2 loops, kg/sec.
10. Fuel assembly outlet temperature	Coolant outlet temperatures of every 115 fuel assemblies, ℃

α : Constant $(0<\alpha<1)$

n : Scanning order.

5. 2 Information for fuel inspection

The fast reactor is presently in the development stage to its practical application and it is extremely important to accumulate the data relating to the irradiation of fuel and material in the reactor. Around the JOYO site are provided such post-irradiation test facilities as the Fuel Monitoring Facility, the Material Monitoring Facility and the Alpha-Gamma Facility. If the results of post-irradiation test are analyzed and thus obtained results are to be effectively reflected in the reactor design, it is necessary to know accurate history of irradiation of the test samples. With regard to these irradiation tests, this code system provides such information as the history of operation of the reactor including its transient state, the heat generation removal and burn-up characteristics. The main items of such information are shown in TABLE 5. 3. Of these, those relating to the core characteristics are calculated by the subsystem SMART or HONEYCOMB as required.

As for the supply of data to the irradiation tests, apart from the ordinary printed form of output, detailed operation data are supplied by the MASTER FILE (magnetic tape) of the subsystem MASTOR.

TABLE 5. 3 Output parameters of fuel assemblies for inspection

1. Loaded location in the reactor	11. Radial power peaking factor across the pin bundle
2. Type of the assembly	12. Linear heat rate, av. and max.
3. Identification No.	13. Clad nominal max. temp.
4. Bundle power	14. Pellet nominal max. temp.
5. Cumulative power from initial loading	15. Burnup, av. and max.
6. Cumulative days from initial loading	16. Fluence, max.
7. Cumulative scram No.	17. Nuclear materials contained in assembly and their deviation during the cycle
8. Coolant flow rate	18. Axial distributions of burnup, linear heat rate and neutron flux (>0.1 MeV)
9. Coolant outlet temp., calculated and measured	
10. Axial power peaking factor	

5. 3 Documentation of operation performance

Such main operation data as the core thermal power, coolant flow rate, temperature, and control rod withdrawal position are recorded in the operation diary required by the regulation. In this code system, these operation data are summarized and recorded from the viewpoint of the core performance. That is to say, the data as shown in TABLE 5. 2 are printed out during the cycle of operation, from the start-up to the end, at any arbitrary point of time or every hour during the transient state of the reactor and every day during the normal operation. Apart from the above, the integrated number of days of operation, integrated operation power, average power during cycle, and the history of accidents as scram are recorded.

As for the records of the movement of fuel assemblies and nuclear material weight balance, they are as explained in relation to the subsystem MASTOR in the preceding paragraph.

6. Concluding Remarks

The JOYPAC system has made it possible to calculate the various items of information necessary for the operation of the fast experimental reactor JOYO by a single code system. This is not only the first system of this kind for fast reactors but also a unique system having outstanding features which is able to handle the great variety of data and adopts the ingenious methods to calculate such data.

The simplified subsystem SMART, which was described in Part 1 of this report, employs simplified calculation methods for various characteristics on the basis of the great deal of information obtained during the process of designing the JOYO core in an attempt to greatly reduce the time required for making calculations. The accuracy of the calculation methods used in the SMART has proved to be statisfactory as a result of comparison with the results obtained by the calculation methods used in the designing of JOYO. It will be further improved in the future by reappraising the basic data and evaluation methods on the basis of the measured data to be obtained by the performance tests after criticality and actual reactor operation.

The detailed subsystems discussed in Part 2 are a nuclear and a thermohydraulics code systems using detailed calculation models and analytical methods. Worthy of special mention is its function to obtain the power distribution in the fuel assembly in three dimensions and to calculate not only the flow distribution in the reactor but also the temperature distribution in the fuel assembly, including the coolant mixing effect. Each of them is really a very unique achievement without parallel anywhere else in the field of fast reactor technology.

The efficient use of these two subsystems will greatly increase the operating safety and efficiency of JOYO and improve its functions.

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2) KATSURAGI, S., TAKANO, H., HASEGAWA, A., SUZUKI, T., KIKUCHI, Y. : "Development of the JAERI Calculation Systems for Fast Reactors", Intern. Sym. (Tokyo, Oct. 1973).
3) OYAMA, A. : "Development Program of New Types of Power Reactor Fast Breeder and Heavy Water Reactor", Intern. Sym. on Nucl. Power Tech. and Econom. (Taipei, Jan. 1975).

Appendix : Sample Output of the Recording Subsystem MASTOR

LIST

1. Reactor Operation Data
2. Core Nuclear and Thermal Characteristics
3. Core Elements Performance Data
   Core Fuels, Blankets, Control Rods, Reflectors, Neutron Source and Surveillance Test Pieces
4. Balance Sheets of Nuclear Material Weights
5. History of Movement of Fuel Assemblies
6. Loading Map of Assemblies
7. Map of Assembly Characteristics

REACTCR CPERATICN DATA PAGE

MARK	CATE	POWER (MW)	CM.POWER (MWD)	COUNTER (NV,PPP)	FLOW (KG/SEC)		TEMP(DEG)	IN	OUT	CONTRCL ROD (MM)	R/R		S/R
ST	77.10. 1.13.30	0.0	C.C	2.999E 01	(A)	301.4	(A)	250.0	250.0	(3A3)	0.0	(3B3)	0.0
				2.999E 01	(B)	301.4	(B)	250.0	250.0	(3D3)	0.0	(3C3)	0.0
												(3E3)	0.0
												(3F3)	0.0
PR	77.10. 1.14. 0	0.C	0.0	1.500E 02	(A)	301.4	(A)	250.0	250.0	(3A3)	0.0	(3B3)	900.0
				1.500E 02	(B)	301.4	(B)	250.0	250.0	(3D3)	0.0	(3C3)	900.0
												(3E3)	450.0
												(3F3)	900.0
CR	77.10. 1.14.30	0.0	0.0	3.499E 03	(A)	301.4	(A)	250.0	250.0	(3A3)	450.0	(3B3)	900.0
				3.499E 03	(B)	301.4	(B)	250.0	250.0	(3D3)	430.0	(3C3)	900.0
												(3E3)	900.0
												(3F3)	900.0
PR	77.10. 1.15. 0	2.000E 00	2.080E-02	2.000E 00	(A)	300.9	(A)	256.0	257.0	(3A3)	460.0	(3B3)	900.0
				2.000E 00	(B)	300.9	(B)	256.0	257.0	(3D3)	450.0	(3C3)	900.0
				2.000E C0								(3E3)	900.0
												(3F3)	900.0
DM	77.10. 1.15.30	4.000E 00	4.170E-02	3.999E 00	(A)	301.7	(A)	368.0	370.0	(3A3)	465.0	(3B3)	900.0
				3.999E 00	(B)	301.7	(B)	368.0	370.0	(3D3)	463.0	(3C3)	900.0
				3.999E 00								(3E3)	900.0
												(3F3)	900.0
PR	77.10. 1.16. C	1.000E 01	1.042E-01	1.005E 01	(A)	301.5	(A)	370.0	383.0	(3A3)	465.0	(3B3)	900.0
				1.010E 01	(B)	301.5	(B)	370.0	383.0	(3D3)	465.0	(3C3)	900.0
				1.015E 01								(3E3)	900.0
												(3F3)	900.0
DM	77.10. 1.16.30	2.000E 01	2.082E-01	2.010E 01	(A)	301.5	(A)	370.0	396.0	(3A3)	470.0	(3B3)	900.0
				2.020E 01	(B)	301.5	(B)	370.0	396.0	(3D3)	470.0	(3C3)	900.0
				2.030E 01								(3E3)	900.0
												(3F3)	900.0
PR	77.1C. 1.17. C	2.500E 01	2.602E-01	2.510E 01	(A)	283.1	(A)	370.0	405.0	(3A3)	472.0	(3B3)	900.0
				2.520E 01	(B)	320.0	(B)	370.0	401.0	(3D3)	472.0	(3C3)	900.0
				2.530E 01								(3E3)	900.0
												(3F3)	900.0
DM	77.10. 1.17.30	3.000E 01	4.164E-01	3.020E 01	(A)	301.5	(A)	370.0	409.0	(3A3)	480.0	(3B3)	900.0
				3.030E 01	(B)	301.5	(B)	370.0	409.0	(3D3)	480.0	(3C3)	900.0
				3.030E 01								(3E3)	900.0
												(3F3)	900.0
PR	77.10. 1.18. C	3.800E 01	4.997E-01	3.830E 01	(A)	301.5	(A)	370.0	419.0	(3A3)	483.0	(3B3)	900.0
				3.840E 01	(B)	301.5	(B)	370.0	419.0	(3D3)	484.0	(3C3)	900.0
				3.840E 01								(3E3)	900.0
												(3F3)	900.0
DM	77.10. 1.18.10	4.000E 01	5.205E-01	4.040E 01	(A)	301.5	(A)	370.0	422.0	(3A3)	485.0	(3B3)	900.0
				4.040E 01	(B)	302.4	(B)	360.0	412.0	(3D3)	485.0	(3C3)	900.0
				4.050E 01								(3E3)	900.0
												(3F3)	900.0

SUMMARY CF OPERATION DATA

MAIN EVENT OF PLANT OPERATION

CCMMENT ;

START-UP DATE	1977 YEAR	10 MONTH	1 DAY	13 HOUR	30 MINUTES	
RATED POWER AT BOC	1977 YEAR	10 MONTH	1 DAY	18 HOUR	50 MINUTES	AT 5.000E 01 (MW)
SHUT DOWN CATE AT ECC	1977 YEAR	11 MONTH	5 DAY	20 HOUR	0 MINUTES	
SCRAM	1977 YEAR	10 MONTH	7 DAY	3 HOUR	16 MINUTES	AT 4.550E 01 (MW)
RE-START-UP CATE	1977 YEAR	10 MONTH	30 DAY	11 HOUR	1 MINUTES	

INTEGRAL DATA

RATEC POWER AT BOC ; 5.00CE 01 (MW)
RATED POWER AT ECC ; 7.500E 01 (MW)
AVERAGE POWER DURING THIS CYCLE ; 4.306E 01 (MW)
OPERATICN DAYS FROM START-UP TO SHUT DOWN ; 11 (DAY)
CUMULATIVE POWER DURING THIS CYCLE ; 511.0 (MWD)
CUMULATIVE POWER FROM INITIAL CRITICAL DAY
TO END OF THIS CYCLE ; 1533.0 (MWD)
CUMULATIVE DAYS FRCM INITIAL CRITICAL DAY
TO END OF THIS CYCLE ; 33 (DAY)

SUMMARY OF SCRAM AND SLOW SCRAM

CYCLE NO	POWER (MW)	YR	MCNT	DAY	HR	MIN	SCRAM	SLOW-SCRAM	COMMENTS;
1	45.50	'75	10	7	3	16	1	0	
2	45.50	'76	10	7	3	16	2	0	
3	45.50	'77	10	7	3	16	3	0	

```
                                   NUCLEAR CHARACTERISTICS (CORE & BLANKET)

K-EFF.  100 -C    250 -C    370 -C    RATED
  BOC   1.0169    1.0114    1.0263    1.0080
  EOC   1.0355    1.0299    1.0255    1.0064

REACTIVITY CHANGE (DK/K)
        100-250 C      250-370 C    370 C-RATED   EXCESS(AT RATED)
  BOC   -5.54E-03       1.49E-02    -1.83E-02       8.00E-03
  EOC   -5.64E-03      -4.39E-03    -1.91E-02       6.42E-03

ISOTHERMAL COEF. (DK/K/C)                POWER COEF. (DK/K/MW)
        100-250 C       250-370 C      370-  4.0MW     4.0- 20.0MW    20.0- 30.0MW    30.0- 40.0MW    40.0- 50.0MW
  BOC   -3.69E-05        1.24E-04      -4.15E-03       -2.51E-05      -6.46E-05       -3.12E-05       -3.04E-05
        100-250 C       250-370 C      370-  3.0MW     3.0- 10.0MW    10.0- 30.0MW    30.0- 75.0MW
  EOC   -3.76E-05       -3.66E-05      -5.01E-03       -1.29E-04      -6.76E-05       -4.02E-05

CONTROL ROD WORTH (DK/K)                     SHUT DOWN MARGIN (DK/K)
        REG. ROD(2)   SAFE ROD(4)                      100C/6/5              250C/2R         250C-RATED/4S/3S
  BOC    3.96E-02      7.87E-02     ;       8.44E-02 /  6.40E-02            1.02E-02        7.11E-02 /  5.12E-02
  EOC    3.56E-02      7.88E-02     ;       8.51E-02 /  6.47E-02            1.09E-02        6.92E-02 /  4.93E-02

ROD REACTIVITY MAX. INSERTION RATE (DK/K/MM)
          R.ROD          S.ROD
  BOC   4.39E-05       4.46E-05
  EOC   4.39E-05       4.46E-05

NEUTRON PEAK FLUX (NV)   1.95E 15

BURN UP (MWD/TON)
           CORE AV/PK            AX.BLKT AV/PK          RAD.BLKT AV/PK
        1698.6 /   3856.0         28.1 /    116.2        15.9 /    221.1
 PK-LOC            C01C1                    C01C1                  B05F2

KINETICS PARM. AT BOC           SOURCE INTENSITY (CURIE)
  BETA         5.01E-03     ;      BOC   4.22E 01
  L.P.(SEC)    2.78E-07     ;      EOC   1.82E 03



                                        THERMAL CHARACTERISTICS

THERMAL POWER (MW)
     FISSION                              GAMMA                                     TOTAL
       CORE         R/B        A/B        CORE        R/B        A/B        CORE        R/B        A/B        REFL.      TOTAL
  BOC 4.169E 01  2.349E 00  7.644E-01  3.368E 00  1.584E 00  2.967E-01  4.505E 01  3.933E 00  1.061E 00  7.302E-02  5.000E 01
  EOC 6.250E 01  3.554E 00  1.148E 00  5.053E 00  2.373E 00  4.457E-01  6.755E 01  5.927E 00  1.593E 00  1.092E-01  7.500E 01

POWER DENSITY ; AV/PK (KW/L)
             CORE                           R/B                          A/B
  BOC   1.892E 02 /  3.221E 02     2.584E 00 /  4.051E 01     3.342E 00 /  1.292E 01
  EOC   2.836E 02 /  4.812E 02     3.894E 00 /  6.087E 01     5.018E 00 /  1.890E 01

PEAKING FACTOR ; RAD/AX/TOT
                CORE                           R/B
  BOC   1.417 /  1.201 /  1.703     6.731 /  2.329 / 15.676
  EOC   1.414 /  1.200 /  1.697     6.713 /  2.329 / 15.633

LINER HEAT RATE ; AV/PK (W/CM)
           CORE               R/B                A/B
  BOC   119.6 /  203.6       7.8 /  122.6       2.1 /    7.6
  EOC   179.3 /  304.2      11.8 /  184.3       3.2 /   11.3

TEMPERATURE ; NOM/HS (DEG) /LOC
  BOC                    CLAD                              PELLET
    CORE         500.8 /   542.2 / C0000        1269.0 /  1520.6 / C0000
    BLKT         458.3 /   502.4 / B07A4         747.3 /   881.6 / B05F2
  EOC
    CORE         565.3 /   627.3 / C0000        1878.8 /  2306.8 / C0000
    BLKT         504.1 /   571.1 / B07A4         984.7 /  1202.7 / B05F2

FLOW DISTRIBUTION (KG/SEC)
  CORE    444.2    R/B      100.7    R.RODS    9.0    S.RODS    4.9    REF.     3.0    RACKS    21.6    AUX      3.7
  OTHERS   16.2    TOTAL    603.1

PRESSURE DROP (KG/CM**2)
  INLET   2.3769     H.P.PLENUM   2.2067     L.P.PLENUM   0.3350
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RECORD OF GENERAL CORE PERFORMANCE AT RATED POWER PAGE 13

LOCATION	C04E2	C04E3	C04E4	C04F1
TYPE	CORE	CORE	CORE	CORE
IDENTIFICATION NO	PPJ01A	PPJ01B	PPJ01C	PPJ01D
BUNDLE POWER (KW)	605.6 / 908.2	638.2 / 955.2	626.3 / 536.4	568.4 / 849.2
CUMULAT.POWER(MWD)	18.9	19.9	19.5	17.7
DAYS IN CORE	793	793	793	793
NO OF SCRAM	3	3	3	3
BUNDLE FLOW (KG/S)	6.221	6.221	6.221	6.220
OUT.TEMP.NOMI.(C)	446.4 / 484.8	450.5 / 490.8	449.0 / 488.4	441.6 / 477.3
OUT.TEMP.T/C (C)	418.0 / 442.0	418.0 / 442.0	418.0 / 442.0	418.0 / 442.0
AXIAL PEAKING FAC	1.200 / 1.198	1.200 / 1.198	1.200 / 1.198	1.200 / 1.198
RADIAL PEAKING FAC	1.138 / 1.137	1.111 / 1.111	1.132 / 1.133	1.162 / 1.163
QL. MAX (W/CM)	148.1 / 221.6	152.4 / 227.7	152.4 / 227.7	142.1 / 212.1
QL. AV (W/CM)	108.5 / 162.7	114.3 / 171.0	112.2 / 167.7	101.9 / 152.2
CLAD TEMP MAX.(C)	478.6 / 532.7	482.3 / 537.8	481.9 / 537.2	474.0 / 525.3
PELET TEMP MAX.(C)	965.5 / 1360.7	991.4 / 1402.7	987.8 / 1396.8	932.4 / 1300.5
BURN UP MAX(MWD/T)	2251.4	2315.9	2315.8	2158.9
BURN UP AV (MWD/T)	1620.4	1706.6	1674.9	1520.4
CUMU.FLUX MAX(NVT)	2.67E 21	2.79E 21	2.72E 21	2.38E 21

MASS BALANCE

	C04E2 CORE (GR)	AXIAL BLKT (GR)	TOTAL (GR)	C04E3 CORE (GR)	AXIAL BLKT (GR)	TOTAL (GR)
U-235	2148.22	30.13	2178.35	2147.64	30.13	2177.77
DEV	-4.13	-0.03	-4.16	-4.31	-0.03	-4.34
U-236	2.60	0.02	2.62	2.69	0.02	2.71
DEV	0.83	0.01	0.84	0.86	0.01	0.87
U-238	7213.52	15109.43	22322.95	7213.21	15109.20	22322.41
DEV	-1.89	-1.30	-3.19	-1.99	-1.37	-3.36
PU239	1543.25	3.66	1546.91	1543.00	3.87	1546.88
DEV	-1.35	1.21	-0.14	-1.43	1.28	-0.15
PU240	384.33	0.00	384.33	384.32	0.00	384.32
DEV	0.21	0.00	0.21	0.21	0.00	0.21
PU241	74.50	0.00	74.50	74.49	0.00	74.49
DEV	-0.07	0.00	-0.07	-0.07	0.00	-0.07
PU242	14.17	0.00	14.17	14.17	0.00	14.17
DEV	0.01	-0.00	0.01	0.01	0.00	0.01
PUTOT	2016.26	3.66	2019.92	2015.99	3.88	2019.86
DEV	-1.19	1.21	0.02	-1.28	1.28	-0.00

	C04E4 CORE (GR)	AXIAL BLKT (GR)	TOTAL (GR)	C04F1 CORE (GR)	AXIAL BLKT (GR)	TOTAL (GR)
U-235	2147.73	30.13	2177.86	2148.61	30.14	2178.75
DEV	-4.29	-0.03	-4.31	-4.00	-0.02	-4.03
U-236	2.71	0.02	2.73	2.63	0.02	2.65
DEV	0.87	0.01	0.88	0.84	0.01	0.85
U-238	7213.31	15109.32	22322.63	7213.83	15109.77	22323.60
DEV	-1.96	-1.33	-3.29	-1.79	-1.19	-2.98
PU239	1543.12	3.77	1546.88	1543.59	3.33	1546.92
DEV	-1.39	1.24	-0.15	-1.24	1.10	-0.14
PU240	384.38	0.00	384.38	384.48	0.00	384.48
DEV	0.23	0.00	0.23	0.26	0.00	0.26
PU241	74.50	0.00	74.50	74.53	0.00	74.53
DEV	-0.07	0.00	-0.07	-0.06	0.00	-0.06
PU242	14.18	0.00	14.18	14.18	0.00	14.18
DEV	0.01	-0.00	0.01	0.01	0.00	0.01
PUTOT	2016.17	3.77	2019.94	2016.77	3.34	2020.10
DEV	-1.22	1.24	0.02	-1.03	1.10	0.07

AXIAL DISTRIBUTION

AXIAL POSI.	C04E2 BURN-UP (MWD/T)	QL (W/CM)	FLUX (N/CM**2-S)	C04E3 BURN-UP (MWD/T)	QL (W/CM)	FLUX (N/CM**2-S)
1	12.7	0.9	5.69E 13	13.4	0.9	6.03E 13
2	51.3	3.4	2.42E 14	54.4	3.6	2.56E 14
3	1428.4	92.1	5.94E 14	1504.4	97.1	6.35E 14
4	1804.6	116.9	7.54E 14	1900.6	123.2	8.07E 14
5	1969.5	129.0	8.32E 14	2074.2	136.0	8.90E 14
6	1880.3	125.7	8.11E 14	1980.3	132.5	8.68E 14
7	1551.2	107.6	6.94E 14	1633.7	113.4	7.42E 14
8	1088.7	79.3	5.11E 14	1146.6	83.6	5.46E 14
9	33.5	2.6	1.97E 14	35.5	2.8	2.08E 14
10	5.7	0.5	3.95E 13	6.1	0.5	4.18E 13

AXIAL POSI.	C04E4 BURN-UP (MWD/T)	QL (W/CM)	FLUX (N/CM**2-S)	C04F1 BURN-UP (MWD/T)	QL (W/CM)	FLUX (N/CM**2-S)
1	13.1	0.9	5.86E 13	11.5	0.8	5.19E 13
2	52.9	3.5	2.49E 14	46.7	3.1	2.21E 14
3	1476.4	95.3	6.09E 14	1340.2	86.5	5.20E 14
4	1865.3	120.9	7.74E 14	1693.2	109.8	6.60E 14
5	2035.6	133.5	8.54E 14	1847.9	121.2	7.29E 14
6	1943.5	130.1	8.32E 14	1764.3	118.1	7.10E 14
7	1603.3	111.3	7.12E 14	1455.5	101.1	6.07E 14
8	1125.3	82.0	5.24E 14	1021.5	74.5	4.47E 14
9	34.5	2.7	2.03E 14	30.5	2.4	1.79E 14
10	5.9	0.5	4.07E 13	5.2	0.4	3.60E 13

INFORMATION OF CONTROL ROD PAGE 2

LOCATION	E03B3	E03C3	E03E3	E03F3
TYPE	SROD	SROD	SROD	SROD
IDENTIFICATION NO	SROD01	SROD02	SROD06	SROD04
STORED DAY IN CORE	75/ 9/ 5	75/ 9/ 5	76/ 1/ 5	75/ 9/ 5
NO OF SCRAM	3	3	2	3
AXIAL POSITION (MM)				
BOC	900.	900.	900.	900.
EOC	900.	900.	900.	900.
REACTIVITY WORTH (DK/K)				
BOC	1.97E-02	1.97E-02	1.97E-02	1.97E-02
EOC	1.97E-02	1.97E-02	1.97E-02	1.97E-02
REACTIVITY MAX. INSERTION RATE (DK/K/MM)	4.46E-05	4.46E-05	4.46E-05	4.46E-05
PELLET HEAT GEN. (W/CC)				
MAX.	31.3	31.3	31.3	31.3
AV.	5.3	5.3	5.3	5.3
COOLANT FLOW (KG/SEC)	1.216	1.216	1.216	1.216
CLAD MAX. TEMP (DEG)	384.2	384.2	384.2	384.2
B10 BURN UP (A/O)				
MAX.	1.58E-01	1.58E-01	1.04E-01	1.58E-01
AV.	2.76E-02	2.76E-02	1.84E-02	2.76E-02
CUMULATIVE MAX. FLUX (NVT)	2.58E 20	2.58E 20	1.70E 20	2.58E 20

AXIAL DISTRIBUTION

NODE	E03B3 B10 B-UP (A/O)	HEAT (W/CC)	FLUX (NV)	E03C3 B10 B-UP (A/O)	HEAT (W/CC)	FLUX (NV)
1	1.00E-01	1.98E 01	5.96E 13	1.00E-01	1.98E 01	5.97E 13
2	3.75E-02	7.41E 00	2.23E 13	3.75E-02	7.42E 00	2.23E 13
3	2.06E-02	4.07E 00	1.23E 13	2.06E-02	4.08E 00	1.23E 13
4	1.26E-02	2.48E 00	7.47E 12	1.26E-02	2.49E 00	7.48E 12
5	7.79E-03	1.54E 00	4.62E 12	7.81E-03	1.54E 00	4.63E 12
6	4.85E-03	9.53E-01	2.87E 12	4.85E-03	9.54E-01	2.87E 12
7	3.15E-03	6.17E-01	1.86E 12	3.15E-03	6.18E-01	1.86E 12

NODE	E03E3 B10 B-UP (A/O)	HEAT (W/CC)	FLUX (NV)	E03F3 B10 B-UP (A/O)	HEAT (W/CC)	FLUX (NV)
1	6.59E-02	1.98E 01	5.96E 13	1.00E-01	1.98E 01	5.96E 13
2	2.47E-02	7.41E 00	2.23E 13	3.75E-02	7.42E 00	2.23E 13
3	1.36E-02	4.07E 00	1.23E 13	2.06E-02	4.08E 00	1.23E 13
4	8.29E-03	2.48E 00	7.47E 12	1.26E-02	2.49E 00	7.48E 12
5	5.13E-03	1.54E 00	4.62E 12	7.81E-03	1.54E 00	4.63E 12
6	3.19E-03	9.53E-01	2.87E 12	4.85E-03	9.54E-01	2.87E 12
7	2.07E-03	6.17E-01	1.86E 12	3.15E-03	6.17E-01	1.86E 12

SURVEILLANCE INFORMATICN PAGE 2

SURVEILLANCE -PCSITION	R25			M1			M2		
IDENTIFICATICN NC	SURV10			TTJT01			SURV12		
	GROUP 1	GROUP 2	TCTAL	GROUP 1	GROUP 2	TOTAL	GROUP 1	GROUP 2	TOTAL
1	1.148E 18	6.403E 16	5.C48E 18	7.089E 17	4.652E 16	3.498E 18	7.075E 17	4.643E 16	3.487E 18
2	2.843E 18	1.754E 17	1.101E 19	1.716E 18	1.219E 17	7.626E 18	1.712E 18	1.217E 17	7.603E 18
3	6.703E 16	1.867E 18	1.654E 19	1.778E 18	1.227E 17	1.160E 19	1.771E 18	1.222E 17	1.157E 19
4	7.944E 18	2.213E 18	1.960E 19	2.1C7E 18	1.454E 17	1.375E 19	2.098E 18	1.448E 17	1.371E 19
5	8.645E 18	2.4C8E 18	2.133E 19	2.292E 18	1.583E 17	1.456E 19	2.283E 18	1.576E 17	1.492E 19
6	8.641E 18	2.4C7E 18	2.132E 19	2.291E 18	1.582E 17	1.496E 19	2.282E 18	1.575E 17	1.491E 19
7	7.928E 18	2.209E 18	1.956E 19	2.102E 18	1.451E 17	1.372E 19	2.094E 18	1.445E 17	1.368E 19
8	6.681E 18	1.861E 18	1.648E 19	1.772E 18	1.223E 17	1.157E 19	1.765E 18	1.218E 17	1.153E 19
9	2.840E 18	1.749E 17	1.097E 19	1.713E 18	1.216E 17	7.604E 18	1.710E 18	1.213E 17	7.581E 18
10	1.219E 18	6.715E 16	5.304E 18	7.510E 17	4.884E 16	3.675E 18	7.495E 17	4.874E 16	3.664E 18

INFORMATION OF REFLECTOR
AND NEUTRON SOURCE

LOCATION	R10E9	R10F3	R10F4	R10F5	R10F6	R10F7	R1CF8	R1CF9
TYPE	NLIT	NLIT	NLIT	NLIT	NLIT	NLIT	NLIT	NLIT
IDENTIFICATICN NC	NLIT20	NLIT21	NLIT22	NLIT23	NLIT24	NLIT25	NLIT26	NLIT27
STORED DAY IN CORE	74/1C/10	74/10/10	74/10/10	74/10/10	74/10/10	74/10/10	74/1C/10	74/10/10
GAMMA HEAT (W/CC)								
MAX.	0.6	0.7	0.8	0.9	C.9	C.9	C.8	C.7
AV.	0.2	0.2	0.2	0.3	0.3	0.3	0.2	0.2
COOLANT FLCW(KG/S)	0.062	0.062	0.062	0.062	0.C62	0.062	C.C62	0.C62
CUMU.FLUX MAX(NVT)	7.23E 19	8.35E 19	1.09E 20	1.29E 20	1.37E 20	1.32E 20	1.15E 2C	9.14E 19

NEUTRCN SOURCE

LOCATION	SC7F1
ICENTIFICATICN NO	SOUR01
STCRED DAY IN CORE	75/ 9/25
STRENGTH OF SOURCE (CURIE)	
AT REACTOR START UP	4.22E 01
AT EOC	1.82E 03
CHANNEL FLOW (KG/S)	0.216
CLAD MAX. TEMP (C)	376.7

ACCOUNTING RECORD CYCLE NC 3 PAGE 7

ACCOUNTING PERICD '77/10/ 1 ------ '77/11/ 5

UNIT ; GRAMMS

		C.FUEL	C.A/B	C.R/B	RK.FUEL	RK.A/B	RK.R/B	PL.FUEL	PL.A/B	PL.R/B
U -235	ECC	148194.4	2079.5	13207.6	8606.8	120.6	0.0	2160.7	30.2	139.8
	BOC	148471.7	2080.7	13216.8	8607.9	120.6	0.0	2160.7	30.2	139.8
	DEV	-277.3	-1.2	-9.2	-1.1	-C.0	0.C	0.0	0.0	0.0
U -236	ECC	182.3	1.4	8.2	7.5	0.1	0.C	0.1	0.0	C.1
	BCC	120.9	C.9	5.5	7.2	C.1	0.0	0.1	0.0	0.1
	DEV	61.5	0.5	2.7	0.3	C.0	0.C	0.0	0.0	0.0
U -238	EOC	497714.4	1042553.8	6619976.0	28860.3	60442.3	0.0	7219.2	15113.3	70052.4
	BCC	497858.8	1042305.9	6620376.0	2886C.8	60442.6	0.0	7219.2	15113.3	70052.4
	DEV	-144.4	247.9	-400.0	-0.5	-0.3	0.C	0.0	0.0	0.0
U.TCTAL	EOC	646C91.1	1C44634.6	6633191.C	37474.7	60563.0	0.0	9380.0	15143.5	70192.3
	BCC	646451.3	1044387.4	6633598.0	37475.9	60563.3	0.C	9380.0	15143.5	70192.3
	DEV	-360.2	247.2	-407.0	-1.2	-0.3	0.C	0.0	0.0	0.0
PU-239	ECC	106465.9	244.8	1171.4	6177.2	10.3	0.0	1547.4	0.1	12.6
	BCC	106577.5	160.0	783.3	6177.5	1C.0	0.0	1547.4	0.1	12.6
	DEV	-111.6	84.8	388.1	-0.3	0.3	0.0	0.0	0.0	C.0
PU-240	ECC	26516.4	0.1	0.6	1536.5	C.0	0.0	383.7	0.0	0.0
	BCC	2649C.9	0.1	0.2	1536.3	C.0	0.C	383.7	0.0	0.0
	DEV	25.5	0.1	0.3	C.2	0.0	0.0	0.0	0.0	0.0
PU-241	ECC	5139.6	0.0	0.0	298.3	C.0	0.0	74.7	-0.0	0.0
	BOC	5145.9	0.0	0.0	298.3	C.0	0.C	74.7	-0.0	0.C
	DEV	-6.3	C.0	0.0	C.0	C.0	0.0	0.0	0.0	0.0
PU-242	ECC	577.9	0.0	0.1	56.6	0.0	C.C	14.1	C.0	0.0
	BOC	976.5	C.0	0.1	56.6	0.0	0.0	14.1	0.0	0.0
	DEV	1.4	0.0	-0.0	C.0	-0.0	0.0	0.0	0.0	0.0
PU.TOTAL	ECC	139C99.7	244.9	1172.0	8068.6	10.3	0.C	2C20.0	14.2	12.6
	BOC	139190.8	160.1	783.6	8068.7	10.0	0.0	2020.0	0.1	12.6
	DEV	-91.1	84.9	388.4	-C.0	C.3	C.C	0.0	14.1	0.0
PU.FIS.TOTAL	EOC	111605.4	244.8	1171.4	6475.5	10.3	C.C	1622.1	0.1	12.6
	BCC	111723.4	160.0	783.3	6475.7	10.0	0.0	1622.1	0.1	12.6
	DEV	-117.9	84.8	388.1	-0.2	0.3	C.0	C.0	0.0	0.0
NUMBER OF BUNDLE		69	69	189	4	4	0	1	1	2

ACCOUNTING RECORD (MOVE ONLY)   CYCLE NO   3   PAGE   9

ACCOUNTING PERIOD   '77/10/ 1 ------ '77/11/ 5

UNIT : GRAMMS

		C.FUEL	C.A/B	C.R/B	RK.FUEL	RK.A/B	RK.R/B
U -235	PEC	146320.9	2051.0	13286.6	4308.7	6C.3	69.9
	BCC	148471.7	208C.7	13216.8	8607.9	120.6	0.C
	DEV	2150.8	29.7	-69.9	4299.2	6C.3	-69.9
U -236	PEC	125.4	1.0	5.5	2.6	C.0	C.C
	BOC	120.9	C.5	5.5	7.2	0.1	0.0
	DEV	-4.5	-0.0	-0.0	4.5	0.0	-0.C
U -23E	PEC	490627.1	1027525.7	6655400.0	14432.5	30222.7	35028.1
	BOC	497858.8	1C42305.9	662C376.C	28860.8	60442.6	0.0
	DEV	7231.8	14780.2	-35024.0	14428.3	30219.9	-35028.1
U.TOTAL	PEC	637C73.3	1C29577.6	6668692.0	18743.8	30283.1	35098.1
	BCC	646451.3	1044387.4	6633598.0	37475.9	60563.3	0.0
	DEV	9378.C	14809.8	-35094.0	18732.0	3028C.2	-35C98.1
PU-239	PEC	105014.E	166.3	791.2	3090.2	3.7	4.6
	BCC	106577.5	160.0	783.3	6177.5	10.C	0.0
	DEV	1562.8	-6.3	-8.0	3087.3	6.3	-4.E
PU-240	PEC	26119.4	C.1	0.3	767.9	0.0	0.0
	BCC	26490.9	0.1	0.2	1536.3	C.C	0.0
	DEV	371.6	-C.0	-0.0	768.4	0.0	-0.C
PU-241	PEC	5069.9	0.0	0.0	149.2	C.0	0.0
	BOC	5145.9	0.0	0.0	298.3	0.0	C.C
	DEV	76.C	-C.0	-0.0	149.0	0.0	-0.0
PU-242	PEC	962.9	0.0	0.1	28.3	C.0	0.0
	BOC	976.5	C.0	0.1	56.6	0.0	0.0
	DEV	13.7	-0.0	-0.0	28.3	0.0	-0.C
PU.TOTAL	PEC	137166.8	166.4	791.6	4035.6	3.7	4.6
	BCC	139190.8	16C.1	783.6	8068.7	10.0	0.C
	DEV	2024.C	-6.3	-8.0	4033.1	6.3	-4.6
PU.FIS.TOTAL	PEC	110084.6	166.3	791.2	3239.4	3.7	4.6
	BOC	111723.4	16C.0	783.3	6475.7	10.0	0.0
	DEV	1638.8	-6.3	-8.0	3236.3	6.3	-4.6
NUMBER OF BUNDLE		69	69	189	4	4	C

HISTORY OF FUEL BUNDLE AND BLANKET MOVEMENT   PAGE 16

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IDENT.NO  TYPE      REAC
PPJD15    CORE   75/ 9/ 5 C04D1

IDENT.NO  TYPE      REAC
PPJD16    CORE   75/ 9/ 5 C04D2

IDENT.NO  TYPE      REAC
PPJD17    CORE   75/ 9/ 5 C04D3

IDENT.NO  TYPE      REAC
PPJD18    CORE   75/ 9/ 5 C04D4

IDENT.NO  TYPE      REAC
PPJD19    CORE   75/ 9/ 5 C04E1

IDENT.NO  TYPE      REAC                RACK               REAC
PPJXC1    CORE   75/ 9/ 1 C0000     76/ 1/ 1 R1        77/ 9/23 C03C1

IDENT.NO  TYPE      REAC                RACK
PPJX02    CORE   75/ 9/ 1 C01A1     76/ 1/ 2 R2

IDENT.NO  TYPE      REAC                RACK
PPJX03    CORE   75/ 9/ 1 CC1B1     77/ 9/25 R17

IDENT.NO  TYPE      REAC
PPJX04    CORE   75/ 9/ 1 CC1C1

IDENT.NO  TYPE      REAC
PPJXC5    CORE   75/ 9/ 1 C01C1

IDENT.NO  TYPE      REAC
PPJX06    CORE   75/ 9/ 1 CC1E1

IDENT.NO  TYPE      REAC
PPJX07    CORE   75/ 9/ 1 C01F1

IDENT.NO  TYPE      REAC                REAC
PPJXC8    CORE   75/ 97 2 C02A1     77/ 9/25 C02D1

IDENT.NO  TYPE      VAUL                REAC
PPJD12    TEST   76/11/ 5  3- 3     77/ 9/25 C05E3
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FUEL MOVE DATA

PAGE 1

NUMBER OF FUEL MOVE = 15 CCMMENT FUNCTICN CHECK

FUEL-ID	DATE	BEFCRE MCVE ID	BEFCRE MCVE LCCATICN	AFTER MCVE ID	AFTER MCVE LCCATICN	DISPATCH
NKJDCB	'77/ 9/25	REAC	CC5D2	PCCL	22- 1	
NKJDC2	'77/ 9/24	RACK	R3	PCCL	1- 5	
PPJDX5	'77/ 9/25	VALL	1- 5	REAC	CC1B1	
PPJDX6	'77/ 9/23	VALL	1- 6	REAC	C05D2	
PPJDOG	'77/ 9/23	REAC	CC3C1	RACK	R6	
PPJDOR	'77/ 9/24	PCCL	1- 5	PCCL	2- 5	
PPJDC6	'77/ 9/25	REAC	CC2D1	REAC	CC2A1	
PPJD1S	'77/ 9/25	REAC	CC5E3	RACK	R3	
PPJD1Z	'77/ 9/25	VALL	3- 3	REAC	CC5E3	
PPJXC1	'77/ 9/23	RACK	R1	REAC	CC3C1	
PPJX03	'77/ 9/25	REAC	CC1B1	RACK	R17	
PPJX08	'77/ 9/25	REAC	CC2A1	REAC	C02D1	
RROD01	'77/ 9/23	RACK	R15	RACK	R7	
RRODC5	'77/ 9/23	REAC	DC3A3	PCCL	2- 3	
SLITC1	'77/ 9/23	REAC	RC9A1	PCOL	3- 5	

LIBRARY DATA IN REACTOR AFTER CHANGE

PAGE 2

LCCATICN	TYPE	FUEL-ID.	DATE	AXCO
C04C2	CORE	PPJD16	'75/ 9/ 5	YES
CC4D3	CORE	PPJD17	'75/ 9/ 5	YES
C04D4	CORE	PPJC18	'75/ 9/ 5	YES
C04E1	CORE	PPJD19	'75/ 9/ 5	YES
CC4E2	CORE	PPJD1A	'75/ 9/ 5	YES
CC4E3	CORE	PPJC1B	'75/ 9/ 5	YES
CC4E4	CORE	PPJD1C	'75/ 9/ 5	YES
CC4F1	CORE	PPJC1C	'75/ 9/ 5	YES
C04F2	CORE	PPJC1E	'75/ 9/ 5	YES
C04F3	CORE	PPJD1F	'75/ 9/ 5	YES
C04F4	CORE	PPJC1G	'75/ 9/ 5	YES
E05A1	BLKT	NKJD01	'74/1C/10	NO
C05A2	CORE	PPJDX4	'76/ 1/ 3	YES
C05A3	CORE	PPJC1H	'75/ 9/ 6	YES
CC5A4	CORE	PPJD1J	'75/ 9/ 6	YES
EC5A5	BLKT	NKJD03	'74/1C/10	NC
E05B1	BLKT	NKJD04	'74/10/10	NO
B05B2	BLKT	NKJD05	'74/10/10	NO
CC5B3	CORE	PPJD1K	'75/ 9/ 6	YES
CC5B4	CORE	PPJC1L	'75/ 9/ 6	YES
E05B5	BLKT	NKJD06	'74/10/10	NO
BC5C1	BLKT	NKJD07	'74/10/10	NO
E05C2	BLKT	NKJD08	'74/1C/10	NO
CC5C3	CORE	PPJD1M	'75/ 9/ 6	YES

THE CCRE MAP INFCRMATION (NO-2/4)

CF BUNDLE POWER (KW)
AND AVE.BURN-UP (MWD/T)

PAGE 2
BUNDLE POWER (KW)
AVE.BURN-UP (MWD/T)
LOCATION

5.0 4.0 BC9A2								
	6.2 5.C B09A3							
8.5 6.9 B08A2		7.3 5.9 BC9A4						
	1C.6 8.5 B08A3		7.9 6.4 B09A5					
18.4 14.5 B07A2		12.2 9.9 B08A4		7.9 6.3 B09A6				
	22.7 18.0 B07A3		12.8 10.4 B08A5		7.2 5.8 B09A7			
37.8 26.9 BC6A2		26.1 2C.7 B07A4		12.1 9.8 B08A6		6.1 4.9 B09A8		
	55.9 37.C B06A3		20.9 17.7 B07A5		1C.3 8.4 B08A7		4.8 3.9 B09A9	
49E.3 883.8 C05A2		58.9 40.C B06A4		18.0 15.2 B07A6		8.2 6.6 BC8A8		
	537.4 1356.3 C05A3		40.9 31.0 B06A5		14.3 12.0 B07A7		6.2 5.0 B08B1	
632.2 1664.2 C04A2		484.7 1300.2 C05A4		27.7 23.6 B06A6		10.7 8.9 B07B1		4.3 3.6 BC9B2
	598.0 1599.6 C04A3		72.6 49.8 B05A5		20.1 17.1 B06B1		7.7 6.4 B08B2	
7C8.2 1891.9 C03A2		578.5 1548.1 C04A4		47.1 36.2 B05B1		14.3 12.0 B07B2		5.3 4.5 BC9B3
			541.6 1451.1 C04B1		28.0 23.8 B06B2		9.6 8.1 B08B3	
8C8.0 215E.0 C02A2		661.2 1767.C C03B1		74.3 50.8 B05B2		18.2 15.4 B07B3		6.3 5.3 BC9B4
	778.5 2079.5 C02B1		605.8 1621.2 C04B2		41.8 31.6 B06B3		11.3 9.5 B08B4	
898.0 786.6 C01B1		729.5 1549.5 C03B2		501.7 1343.8 C05B3		21.2 17.9 B07B4		6.9 5.8 BC9B5
	829.9 2217.7 C02B2		638.4 1707.2 C04B3		51.5 35.9 B06B4		12.0 10.1 B08B5	