A MAN-MACHINE COMMUNICATION SYSTEM FOR BOILING WATER REACTOR CORE MANAGEMENT PLANNING

FUEL CYCLES

KEYWORDS: *boiling water reactor, man-machine communication system, BWR simulator, FLARE model, core management planning, graphic display, control rod program*

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A man-machine communication system has been developed for boiling water reactor (BWR) core management planning to provide a very flexible tool, which is complementary to automated optimization programs that maximize or minimize one particular performance index under certain constraints. A three-dimensional BWR simulator, which can cover a wide range of BWR operating conditions, has been developed and coupled with a graphic display serving as a main input-output controlling device.

The system has been successfully applied to generate a long-term control rod programming of a BWR in which locally poisoned fuel assemblies are loaded. The time required for one cycle analysis is ~3 h, out of which the actual computation time is only 4 min with an average of three trials of rod pattern search per exposure step.

The quick response (~5 sec) and the visualized results on the screen are very helpful in understanding the complicated characteristics of the BWR core, and it is found that this kind of tool has a very great educational effect. A similar approach is expected to be applied in other fields such as core design and safety analysis, as well as in core management.

INTRODUCTION

The development of nuclear engineering has come to the stage at which economically competitive nuclear energy is available. The need for the optimization of reactor core management has, therefore, become more and more important because of its great effect on reactor safety and economy.

We have been developing computer programs to automatically optimize core management planning[1,2] (control rod programming, refueling scheduling, etc.) for boiling water reactors (BWRs). These automated programs try to find an optimal solution that maximizes or minimizes the single performance index under several constraints. The solution is only optimal within the framework of the mathematical models therein formulated. In reality, however there are some factors that are very difficult to quantify. In addition, different performance indices cannot be put easily into a single measure in many cases, which is often called the multi-criteria problem. It would be nearly impossible to take all factors into account.

An alternate and complementary approach is to use a reactor simulator in the most efficient way and let the final decision be left to the care of human beings. Figure 1 shows a procedure for BWR core management planning. Optimization criteria include various aspects that cannot easily be evaluated on a common basis. The result of an automated program that has been optimized for one particular performance index can be an initial guess to the man-machine communication system. In the case that no such automated program is available, a trial-and-error approach with sound engineering judgment is the only way to get a feasible, preferably suboptimal, solution.

This man-machine communication system must be very efficient. The simulated result must be presented in a form that is easy to understand and the input provision must be simple enough for a quick response. Needless to say, the simulator must give the results back within a short period of time.

With this in mind, we have developed a three-dimensional BWR simulator using a graphic display as a man-machine communication device. This paper reports the outline of the system and

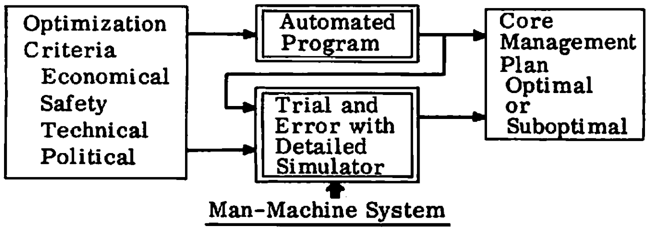

Fig. 1. Method of BWR core management planning.

some experience obtained in an application to long-term control rod programming of BWR.

OUTLINE OF MAN-MACHINE COMMUNICATION SYSTEM FOR BWR CORE MANAGEMENT PLANNING

Hardware Configuration

The hardware we used is the IBM 360/195 system at the IBM Data Center, in Tokyo, Japan. Figure 2 shows the hardware configuration. Simulation is done in the main IBM 360/195 computer and the results are displayed on the cathode ray tube (CRT) of the IBM 2250 graphic display unit through the IBM 2840 controller. The displayed results can be copied on a sheet of paper by the IBM 2285 hard copier. The program function keyboard (PFK) is used to set numerals and erase the displayed results, etc. The function of each key is determined by the definition given by user's program. The alpha numeric keyboard (ANK) is used to edit, change, and generate numerals on the screen. The light pen is used to send data to the central processing unit (CPU) program by pointing to the image displayed on the screen.

BWR Simulator

Three-dimensional simulation is inevitable for the detailed analyses of BWR core characteristics. The capability to analyze a wide range of operating performance (from nearly zero power to the rated power) is required. Computing time must also be short enough for a user to be patient while sitting in front of the CRT. It would be <1 min at most.

One such simulator that satisfies the above requirements and is now commonly available is FLARE (Ref. 3), a three-dimensional, one group, nuclear thermal hydraulic coupled program. The question, however, is its accuracy. It has several parameters that have to be adjusted so that the simulated results are as close to the actual operating data as possible. No argument can be made as to its accuracy until such an adjustment is tried.

Prior to this work, an extensive study[4] of the optimum determination of the adjustable parameters of FLARE has been performed using the operating data on neutron flux distribution. It has been clarified that FLARE can calculate the segment power within the average error (standard deviation) of 5% for the whole range of BWR operation. The adjusted parameters are albedos (horizontal, vertical, top, and bottom), mixing coefficients of neutron transport kernels, and coefficients for the void-quality relation. With this confirmation, we have decided to use FLARE as a BWR simulator of the man-machine communication system.

Next, necessary functions for this simulator have been investigated. Figures 3 and 4 are two examples indicating the procedures to make a startup control rod sequence and a long-term control rod programming of BWR. Perfect simulation of dynamical startup process is impossible by the static calculation. However, it can be well simulated by the combination of power and flow searches and xenon density transient calculation. The long-term burnup process can be simulated by the combination of normal nuclear thermal hydraulic calculation and burnup calculation. The

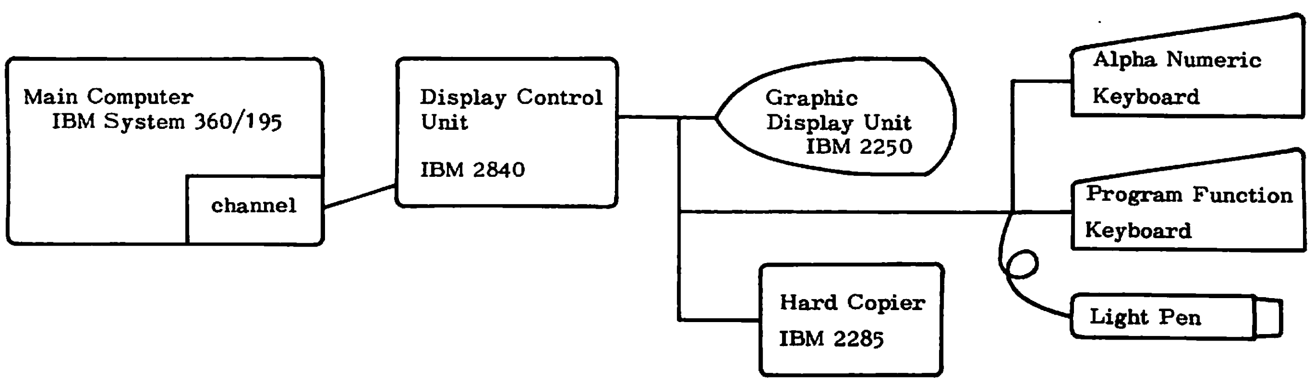

Fig. 2. Hardware configuration.

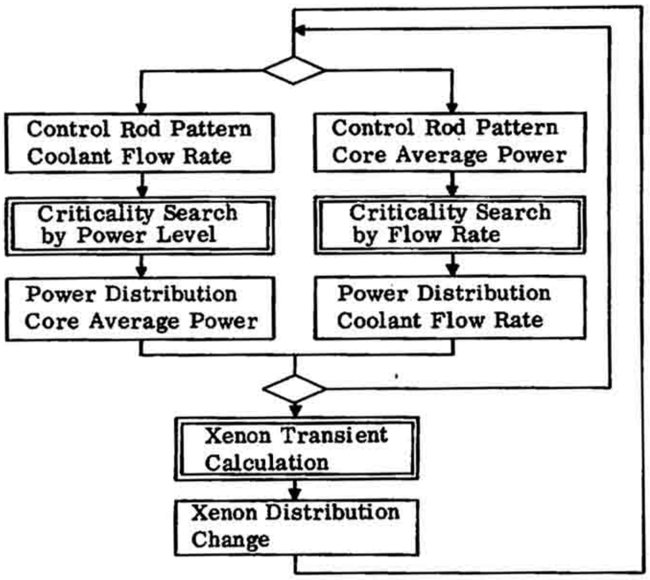

Fig. 3. Simulation procedure to make startup program of BWR.

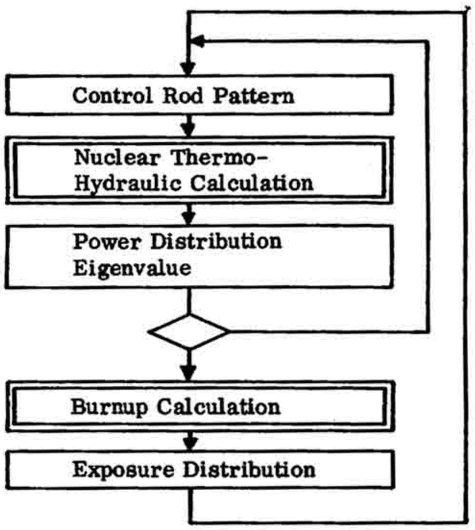

Fig. 4. Simulation procedure to make long-term control rod program of BWR.

necessary functions required for the simulator in making core management planning are summarized in Table I. Functions 2 and 3 are for refueling scheduling and function 7 is for design purpose. These functions must work in any order, according to a user's demand, for the system to exhibit its full merit.

Input Data Preparation

Input data are classified in two sets: standard data and casewise data. The former is a group of data that are common to all cases and are read by the card reader before the calculation is initiated. The latter includes the data which may be altered

TABLE I

Calculational Functions Required for BWR Simulator

1.	Normal power distribution calculation with nuclear-thermal-hydraulic coupled model.
2.	Haling power distribution calculation with nuclear-thermal-hydraulic coupled model.
3.	Exchange of fuel assemblies.
4.	Criticality search by power level.
5.	Criticality search by coolant flow rate.
6.	Xenon transient calculation.
7.	Burnable poison distribution change.
8.	Burnup calculation.
9.	Repeated display of edited results.

in carrying out each of the functions mentioned above. When the standard data are read in, the message of calculational options in Table I is displayed on the screen. By pointing out the option number with the light pen, the next display appears showing a list of the casewise data for the selected case. The user can alter any of these data with the light pen. Figure 5 is the display of the casewise data that appears when the normal power distribution option (function 1) is selected. A message "SELECT SYMBOL OR ROD VALUE" is displayed in the message area above the menu area. The user points to the symbol or the numeral itself with the light pen if he wants to change the value of certain data. Then, the message changes to "INPUT NEW VALUE," and a row of numerals 0 1 2 3 4 5 6 7 8 9 appears in the menu area. New value can be input by pointing to the numeral in sequence, the data being displayed on the right. If the user misinputs the data, he points to the "CANCEL" sign and reinputs the right value. When he finishes the input operation, he points to the "END" sign. Then, the message in the area changes to "PUSH PFK." By pushing the lighted key in PFK, the old value is replaced with the newly input value. After repeating the above procedure as many times as necessary, he finally pushes the "RETURN" key in PFK to initiate the computation. The computing time for the normal power distribution calculation is ~5 s. The input data preparation for other options is nearly the same as explained above.

Display of Calculated Results

The editing of the results is also classified into two sets: standard edit and optional edit. Figure 6 is an example of the standard edit (option 1).

Standard edit summarizes the gross data such as core thermal power, core flow rate, eigenvalue, power peaking, maximum linear heat generating rate, minimum critical heat flux ratio, etc. Edit option for optional edit is displayed in the upper right. Option 2 displays the axial power distributions for the ten assemblies from the maximum power. Figure 7 is an example of this edit. The locations of these ten assemblies are shown in the lower right. Option 3 displays the axial distribution of a selected quantity at a selected location. If this option is selected, a name list of three-dimensional quantities is displayed in the menu area. The user points to the symbol he wants to

```
NORMAL  POWER

    DLS     =   0·000003
    DSIJK   =   0·000005
    DLU     =   0·000005
    PCTP    =   1·000
    PCTF    =   1·000
    NS      =   4
    NV      =  20
    IXESAT  =   1
    IENVLP  =   0
```

	1		3		5		7		9		11
11	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0
	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0
9	0·0	0·0	12·0	0·0	0·0	0·0	11·0	0·0	0·0	0·0	0·0
	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0
7	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0
	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0	
5	0·0	0·0	11·0	0·0	0·0	0·0	0·0	0·0	0·0		
	0·0	0·0	0·0	0·0	0·0	0·0	0·0	0·0			
3	0·0	0·0	0·0	0·0	0·0	0·0	0·0				
	0·0	0·0	0·0	0·0	0·0	0·0					
1	0·0	0·0	0·0	0·0	0·0						

SELECT SYMBOL OR ROD VALUE

Fig. 5. An example of casewise data display (function 1).

```
          STANDARD EDIT                                      OPTION

                                                       1 STANDARD EDIT
PTH           1724·1 MW(TH)       PCTP   1·000         2 PLOT AXIAL POWER
W         56806000·0 LB/HR        PCTF   1·000           DIST. (LARGEST 10)
                                                       3 PLOT AXIAL DIST. OF
LAMBDA      1·01373                                      SPECIFIED QUANTITY
PMAX        2·3088 AT( 8,11, 2)                        4 AVERAGE DIST.
MLHGR      16·0998 KW/FT                               5 P-F MAP
MCHFR       2·3490 AT( 8,11, 2)                        6 PRINT
                                                       7 END OF EDIT
UAV         0·5888
EAV         6·9995 GWD/T
EMAX       10·9740 AT( 1,10, 6)
DE          0·0
TIME        0·0

        SELECT OPTION NO·
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Fig. 6. An example of standard edit display.

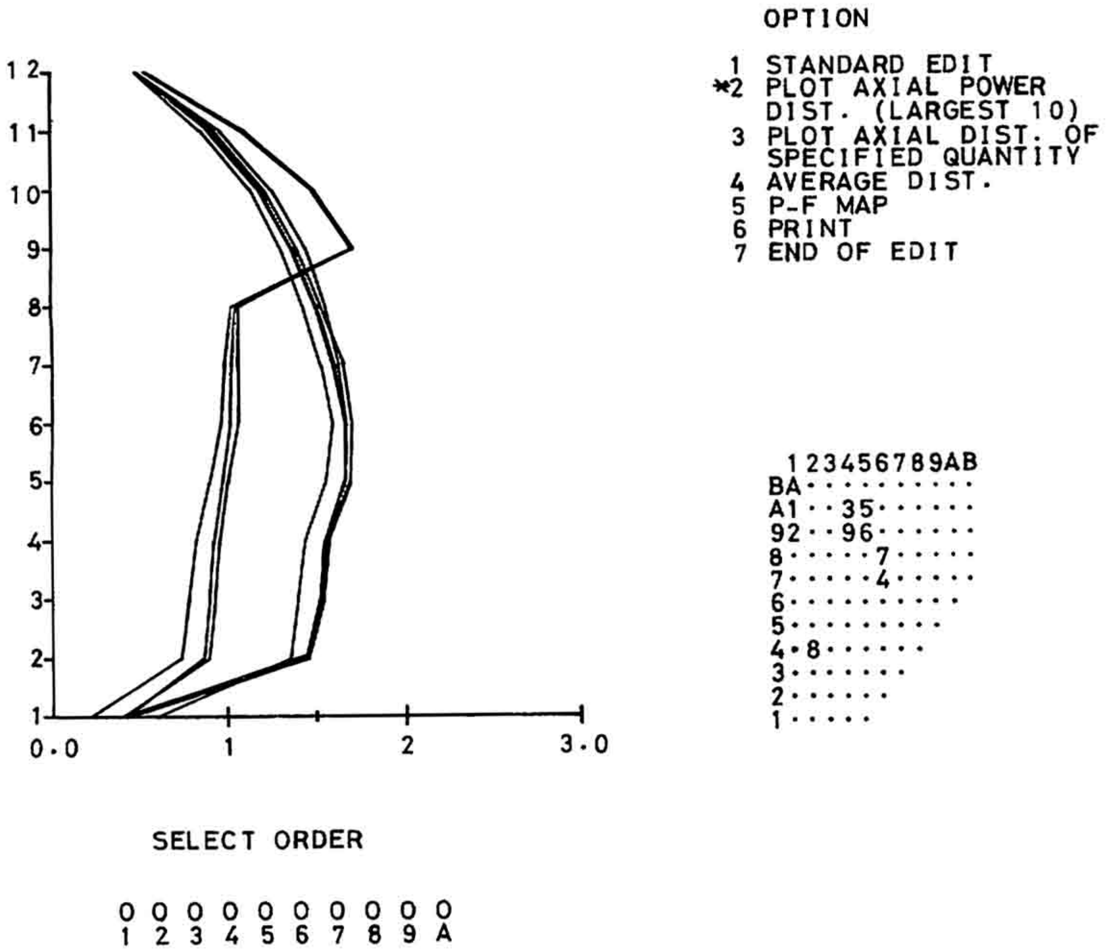

Fig. 7. An example of optional edit display (option 2).

see. Then the core map appears on the right. He can point to the location in the sequence where he wants to look at the distribution. If he wants to see another quantity, he has to push the lighted key in PFK to erase the graph and repeat the same procedure. The main quantities that can be displayed are power, void, infinite multiplication factor, exposure, exposure-weighted void, control fraction, xenon, iodine, critical heat flux ratio, linear heat generating rate, its change rate, etc. Option 4 displays the channel averaged above quantities and other two-dimensional quantities (coolant flow rate and control rod depth) in the core map. Figure 8 is an example of this edit for exposure distribution. Option 5 displays a trajectory in power-flow map. This is useful in understanding the simulated startup process. Provision is made to go back to any point previously calculated for the purpose of modifying the operating trajectory. Information necessary to restart the computation is stored in a work disk. Option 6 is for printing out the successful results by the line printer.

APPLICATION TO LONG-TERM CONTROL ROD PROGRAMMING

In an attempt to demonstrate the practicability of this system, a long-term control rod programming of BWR loaded with 8 × 8 type fuel assemblies has been generated by the trial-and-error method.

Figure 9 is the core geometry. Two types of fuel assemblies with different Gd_2O_3 burnable poison distributions are loaded as shown in this figure. The axial distribution of the poison concentration of type 1 fuel has been determined so as to make the axial power distribution flatter. Optimization of the distribution is another field in which this system can be applied (function 7). Design criteria for the thermal characteristics are

OPTION

1 STANDARD EDIT
2 PLOT AXIAL POWER DIST. (LARGEST 10)
3 PLOT AXIAL DIST. OF SPECIFIED QUANTITY
*4 AVERAGE DIST.
5 P-F MAP
6 PRINT
7 END OF EDIT

2036	2038	2043	2047	2047	2039	2003	1940	1849	1878	1218
2446	2437	2445	2471	2479	2478	2440	2358	2201	1935	1219
2452	2446	2461	2501	2518	2521	2499	2430	2251	1932	1195
2048	2067	2093	2183	2204	2254	2246	2366	2183	1834	1103
2052	2079	2106	2186	2194	2223	2181	2249	2011	1401	887
2467	2493	2516	2533	2507	2432	2291	2076	1500	1010	
2450	2483	2514	2537	2488	2352	2117	1542	1057		
2040	2123	2174	2374	2312	2115	1553	1075			
1847	1922	1962	2120	2010	1511	1064				
1815	1825	1821	1773	1380	1006					
1161	1158	1136	1063	866						

(10** 3)

PUSH PFK

P	U	K	O E	V	CT	XE	I
CHFR	LHGR	LHGO	DLT	PCGR	DPGR	ROD	FLOW

Fig. 8. An example of optional edit display (option 4).

that the minimum critical heat flux ratio (MCHFR) is greater than or equal to 1.9 and that the maximum linear heat generating rate (MLHGR) is less than or equal to 13.4 kW/ft. The latter criterion was found to be more severe than the former in an actual application.

Although no theoretical background is known for the optimal rod withdrawal policy of the axially nonuniform Gd_2O_3 poisoned core, the basic idea of the pattern change ($A_1 \to B_1 \to A_2 \to B_2 \to A_1 \ldots$) and the deep and shallow rods principle, both of which have been verified to be effective for the unpoisoned core,[5] were adopted in preparing the guess pattern.

Pattern change was performed in every 1000 MWd/T, and for each pattern the control rod configuration that satisfies the criticality condition (k_{eff} = 1.0 ± 0.002) and the design criteria mentioned above were searched for by trial-and-error.

As an example of how the rod pattern was modified on the man-machine communication basis, the procedure taken for the search at 7000 MWd/T is explained below. Figure 10 shows the three trials of the control rod pattern.

The control rod pattern in the first trial did not satisfy both the criticality condition and the thermal constraint (λ = 1.0104, MLHGR = 13.54 kW/ft).

To see the profile of the power distribution, the edit option 2 shown in Fig. 11 was selected. The upper figure is the axial power distribution of the peak power assembly and the lower map shows the locations of the ten assemblies from the maximum power (not the channel power but the segment power). The power distribution is distorted and has a gross peak at the upper right of the quadrant. Since the eigenvalue is larger than the desired value, the deep control rod was inserted near the peak power assembly in expectation that the power peak would also be reduced. The result is the second trial. The eigenvalue actually became closer to 1.0 but MLHGR became larger than the expectation. By looking at Fig. 11, it is seen that the power peak shifted to the lower left of the quadrant because of the newly inserted deep rod. The next action would be to insert shallow rods around the peak powered region. The third trial in Fig. 11 shows the final result. The criticality condition required the adjustment of the three shallow rods. The resulting eignevalue and MLHGR are 1.0018 and 12.95 kW/ft, respectively, both of which satisfy the constraints.

The control rod programming at other exposure steps was generated in a similar way. It is relatively easy to find a rod pattern that makes the axially flattened power distribution well under the thermal limit near the beginning of cycle (BOC). However, too much flattened distribution

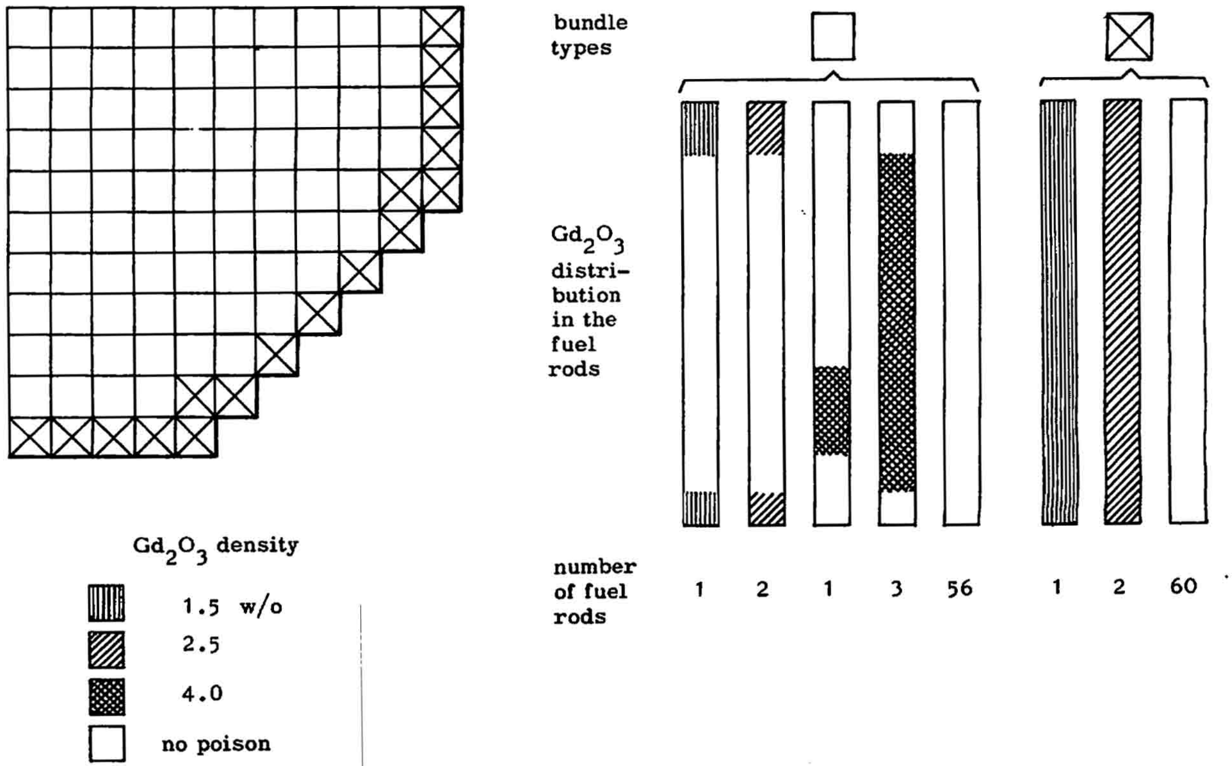

Fig. 9. Core geometry and fuel type.

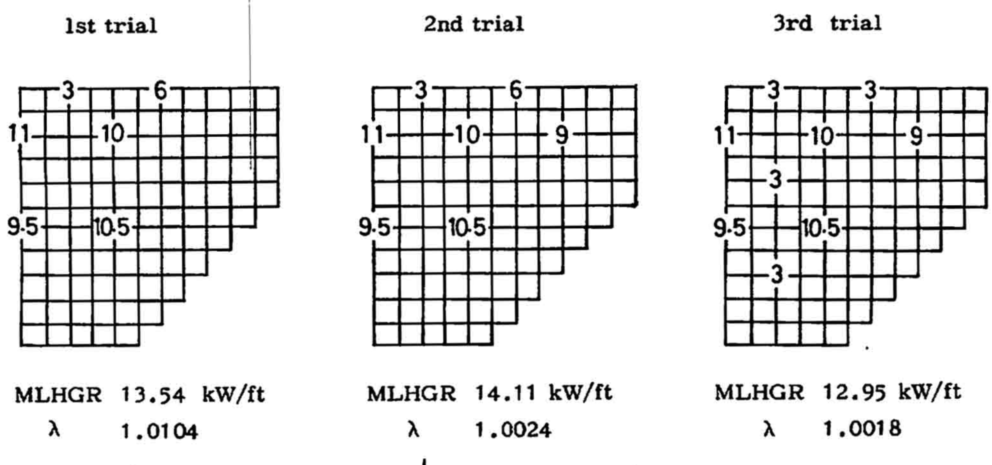

Fig. 10. Modification of control rod pattern at 7000 MWd/T (B_2 pattern).

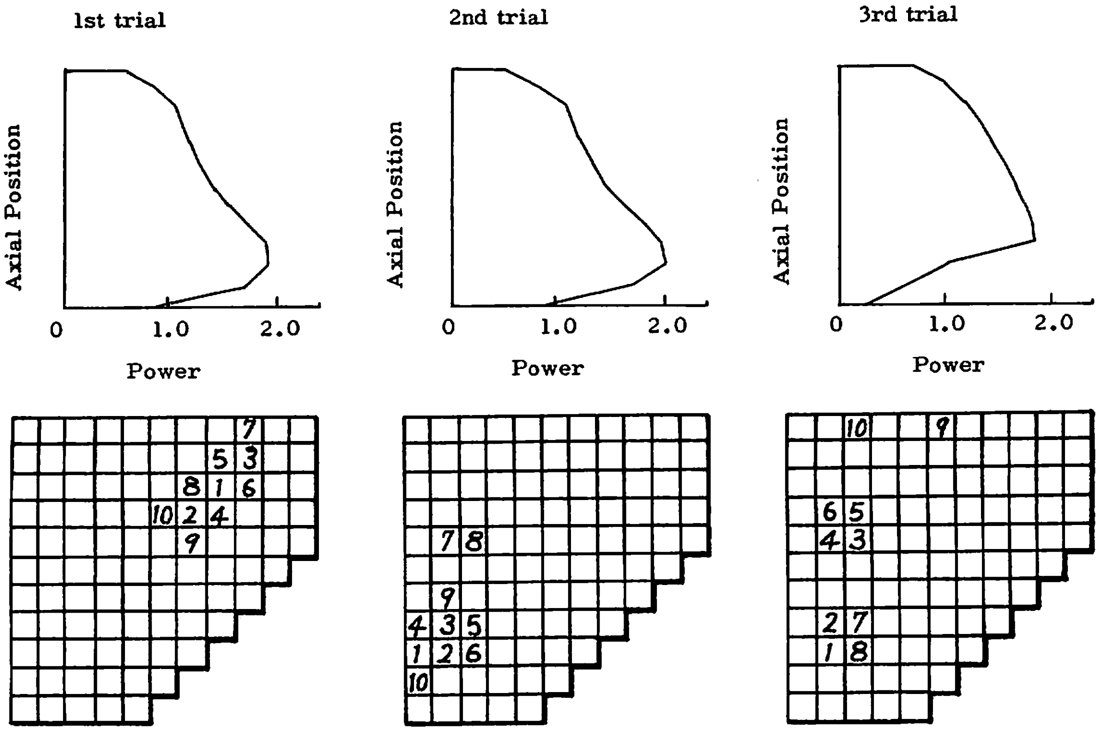

Fig. 11. Axial power distribution of peak power assembly (above) and location of 10 assemblies from the maximum power (below) in the search at 7000 MWd/T.

near the BOC results in the exposure distribution which is not suitable for the power flattening near the end of cycle (EOC). Figure 11 shows examples of typical power distribution at the latter half of the cycle. The adoption of the deep and shallow principle implicitly means the operation which shifts the power peak down near the bottom. Therefore, the best way is to make the most of this principle considering the geometrical symmetry. The generated rod programming with this in mind required the average number of three trials per exposure step and occupied ~ 3 h, out of which only 4 min was used in the CPU.

Generally speaking, it is felt that long-term control rod programming for a Gd_2O_3 poisoned core is not very difficult even by a trial-and-error method. The man-machine communication system worked better than expected. Another merit found in applying the core management planning is its educational effect. The quick response to the input data and the visualized results helped a great deal in intuitively understanding the complicated phenomena in the BWR core (mainly the effect of void feedback and control rod on the power distribution) even by an unexperienced reactor engineer.

The true merit of this system strongly depends on the choice of the quantities to be displayed and controlled and the method used for display and input. Some improvement must be made in the function of fuel assembly exchange for better and rapid understanding of the situation. With these modifications, however, we believe that this kind of approach will be widely used in a variety of reactor engineering, particularly core design and safety analysis, as well as core management planning.

CONCLUSIONS

The man-machine communication system using a graphic display as a main input-output device has been developed to offer a new tool complementary to the automated optimization programs for BWR core management planning. The three-dimensional BWR simulator with a nuclear-thermal-hydraulic coupled model incorporated in the system can carry out the following functions in any order: normal power distribution calculation, Haling power distribution calculation, exchange of fuel assemblies, criticality search by power level, criticality search by coolant flow rate, burnable poison distribution change, and burnup calculation.

For demonstration purposes, this system has been successfully applied to long-term control rod programming of a BWR in which 8 × 8 type fuel assemblies are loaded. The time required to generate a complete control rod programming throughout a cycle is ~3 h, out of which the actual computation time occupied by the CPU is only 4 min. The average number of trials of rod pattern search per exposure step is three.

The quick response to the input data (~5 s) and the visualized display results are very helpful in understanding the complicated nuclear-thermal-hydraulic-coupled characteristics of BWR. Its educational effect is worth mentioning as an unexpected by-product.

In conclusion, the man-machine communication system developed here is very useful in the core management planning of BWR. It will also be used in other fields such as core design and safety analyses.

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