IAEA-TECDOC-816

In-core fuel management: reloading techniques

Proceedings of a Technical Committee meeting and Workshop held in Vienna, 19–21 October 1992

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The originating Section of this publication in the IAEA was:

Nuclear Power Technology Development Section
International Atomic Energy Agency
Wagramerstrasse 5
P.O. Box 100
A-1400 Vienna, Austria

IN-CORE FUEL MANAGEMENT: RELOADING TECHNIQUES
IAEA, VIENNA, 1995
IAEA-TECDOC-816
ISSN 1011–4289

Printed by the IAEA in Austria
August 1995

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FOREWORD

Nuclear core management plays an important part in the assessment of nuclear safety and economics and its effectiveness depends on the tools and techniques used. In the framework of its activities on in-core fuel management, the IAEA has organized the Co-ordinated Research Programmes on In-core Fuel Management Code Package Validation for LWRs and on Safe Core Management with Burnable Absorbers in WWERs and has held several meetings, such as the Specialists Meeting on Advanced Calculational Methods for Power Reactors in France, September 1990, and the Technical Committee Meeting on LWR Core Design Parameters in the Czech Republic, October 1991.

The purpose of the Technical Committee Meeting and Workshop on In-core Fuel Management – Reloading Techniques, convened by the IAEA in Vienna from 19 to 21 October 1992, was to provide an international forum to review and discuss in-core fuel management reloading techniques for light water reactors. A presentation on the history and status of reloading techniques was given by S.H. Levine, Pennsylvania State University, and papers on various computer code descriptions, methodologies and experiences of utilities and vendors for nuclear fuel reloading were presented and discussed. Optimization techniques for reloadings, expert system codes and the number of energy groups used in reloading calculations were discussed in more detail during a workshop session.

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SUMMARY OF THE TECHNICAL COMMITTEE MEETING

1. Technical Sessions

Four technical sessions were held in which papers were presented on the Russian VVER, BWR and the PWR (western European and US design).

One paper on the VVER described a 2-D simulator code based on the FLARE model. This model uses one neutron energy group and only K_∞, M^2, and albedo as variables. Modeling improvement is suggested. Other papers described a code package to analyse the VVER which includes a few groups (4) code together with a coarse mesh nodal simulator code, a thermal hydraulic code, an expert system code to reload VVERs using the Turbo Pascal code, and a method to optimize the axial gadolinium loading in a VVER-440.

The expert system code described how heuristic rules were employed to try to obtain optimum reloads. Optimizing the axial gadolinium was performed by applying Pontryagin's maximum principle and using first-order perturbation theory to estimate the search length. In this way power peaking was reduced by at least 12%.

Both papers on BWRs presented different ways of optimizing the reload of a BWR automatically. Both used heuristic rules to reload the core but such rules were organized differently. The optimum cores were low leakage cores. However, one of the code systems was not completed.

The PWR had several presentations, an advanced transport code for PWRs was presented that develops precise self shielding models i.e., a collision probability model, for calculating cross sections accurately and can also perform criticality calculations. The advanced Westinghouse Vantage 5 fuel assembly design for their PWRs and Vantage 6 fuel assembly design for the VVER was presented. Another paper described the importance in Germany for determining the safety related characteristics of a reload design in advance to assure it will be licensed. This is achieved by defining safety related boundary conditions which in some cases employ correlations to determine the safety of the reactor.

In France, after a joint agreement between the utility and the supplier on the reload pattern determined for a given power plant, two separate safety evaluations for the reloads are performed: one by the utility itself and one by the supplier. In each case, tools based on different codes and calculation schemes have been developed with the aim to fully automate study executions. Presently, the determination of the reloading pattern is not fully automated due to numerous constraints. It seems that in the USA the problem is solved for the PWR for a few constraints. Is it possible to introduce the different European constraints into the US "optimization tools"?

Finally, a Monte Carlo method for optimizing the reload of a PWR was presented together with the Westinghouse power advanced loading pattern search programs. The Monte Carlo method presented an enhanced simulated annealing cooling schedule with a heuristic solution generator to solve this problem more efficiently than the standard method. The Westinghouse reload method assumes an initial EOC power distribution and then proceed to make the core design meet that goal.

2. Workshop Discussion

The following three topics were covered in the workshop discussion:

1. Optimization techniques used to reload reactors.
2. Expert system codes.
3. Number of energy groups caused in reactor calculations.

2.1 Optimization Techniques Used to Reload Reactors

It was stated that optimization of fuel costs is determined by conditions at the end-of-cycle (EOC). Therefore, for anyone to draw conclusions from their calculations they must compare results that occur at EOC.

Important parameters to compare at EOC are ^{235}U inventory, K_{eff} or soluble boron concentration, and maximum $F_{\Delta}H$ for a given fixed BOC ^{235}U inventory. The cycle length for $K_{eff} = 1$ at EOC can also be an important parameter to compare, but it provides the same information as K_{eff} and ^{235}U at EOC for a fixed cycle length.

Maximizing the discharged burnups can also lead to some fuel cost savings but it is a multicycle effect. This is because the fuel assemblies in a single batch must be followed during their lifetime in the different cycles so as to make their discharged burnups as close to the average as possible and as large as possible without violating the burnup constraints. Discharged burnup maximization will increase fuel cost slightly for the first cycle but reduces fuel costs for subsequent cycles.

The difference between the objective function and constraints was explained. The objective function is a goal which is to be minimized or maximized. The actual value established by the code for the objective function is not limited in magnitude; it can be as large or as small as possible. The constraints are limits set on certain variables or parameters which cannot be exceeded. For example the objective function, J, can be the soluble boron concentration at EOC, Sb^{EOC}. Thus:

$$J = -Sb^{EOC} \text{ (the minus maximizes } Sb^{EOC})$$

A constraint can at the same time be applied to the normalized power (power density) NP which must be less than the some maximum value NP_{max}. Thus:

$$\underline{NP} \leq NP_{max}$$

is a constraint where $\underline{NP}$ is a vector which applies to all fuel assemblies in the core at any burnup step.

The optimization calculation which starts with some non optimum value of Sb^{EOC} and within all constraints, changes the reload configuration in small steps to increase Sb^{EOC}. It uses, for example linear programming, to do this while making sure that the NP_{max} constraint is not violated.

The Haling power distribution can be used as a powerful tool in these calculations. This power distribution is one which is constant at all points in the core during depletion. Hence, the burnup of every fuel assembly, $\underline{BU}$, at EOC and BOC has a very simple relation.

$$\underline{BU}^{EOC} = \underline{BU}^{BOC} + \underline{NP}^{H} Pd$$

where Pd is the energy produced during the cycle and $\underline{NP}^{H}$ P is the power produced by the fuel assemblies which is a constant.

This distribution implicitly assumes a sophisticated burnable poison design which can maintain a constant power distribution. The Haling Power distribution is the flattest power distribution possible; hence, any use of practical BP's will create higher power densities in the core sometime during the cycle. In addition, a measure of how good the design is with practical BPs can be determined by how much the maximum $F\Delta H$ exceeds that of the Haling power distribution.

Thus the Haling power distribution separates the placement of fuel and BPs into two separate steps. Placing the fuel determines the core life and economics of the reload design and the BPs are designed to reduce the peak power density during the cycle to the lowest value possible.

In Germany, optimization of the reload pattern for the coming-up fuel cycle has to start from a given fuel inventory. This inventory consists of:

- the total amount of new fuel assemblies available in the new fuel store,
- the fuel assemblies inside the core (in the current cycle),
- the older fuel assemblies available in the spent fuel pool.

It is the task of reload design to prepare a loading pattern which meets the utility requirements regarding cycle length and meets all safety-related requirements. Since the new fuel inventory is to be selected from the fuel assemblies available at the plant site, parameters like enrichment, number of burnable absorber (in Germany Gd_2O_3) rods and concentration of absorber material are not free parameters for optimization.

Regarding safety constraints, efforts have been made in Germany over many years to express the safety constraints in terms of parameters (primary design criteria) directly correlated to plant safety and defining margins to technical failure (cf. Fig. 1 of the German contribution). As a consequence of the direct correlation to plant safety, the bulk of these primary design criteria refers to thermal, thermal-hydraulic and mechanical properties of the individual fuel rod. Direct verification of these parameters requires three dimensional pin-by-pin evaluation over the whole reactor cycle and cannot be carried out for a large number of loading scheme candidates.

For this reason, key parameter substitutes suitable for simple verification have to be introduced. It is a characteristic feature of these substitutes (e.g. the fuel-assembly-averaged power density at beginning of cycle) that they do not correlate well with the primary design criteria like minimum DNBR (over the whole reactor cycle) or maximum corrosion lay thickness (at end of cycle). This weak correlation yields a crucial problem for any automatic optimization procedure based on safety constraints:

- If the safety constraints in terms of key parameter substitutes are set up in a restrictive and conservative way, all proposed loading schemes will meet the safety-related boundary conditions. A large variety of loading patterns capable of meeting the safety-related boundary conditions (and probably the most economic ones), however, will be missed due to the restrictive constraints.

- If, on the other hand, weak safety constraints in terms of key parameters substitutes are set up, a large variety of loading schemes will be proposed containing not only feasible loading schemes but also a large number of those violating some of the primary design criteria (e.g. DNBR, corrosion). In this case, the selection of a very small number of reload safety evaluation candidates out of the large variety of proposed loading scheme candidates requires a lot of time and skilled engineer, otherwise the selection procedure will become similar to playing roulette.

To summarize, it appears to be extremely difficult to set up the safety constraints in a way to avoid both extremes discussed above. For actual reload calculation in Germany, therefore, it is more important to have procedures which allow quick performance to reload safety evaluation for a given loading scheme as well as quick preparation of the licensing documents to be submitted to the authorities. The proper tools for this purpose are available.

Even under German boundary conditions, however, automatic optimization procedures may be useful for pre-optimization in case of long-term fuel management with "exotic" utility requirements (e.g. 2-year-cycle).

EDF has a very restrictive approach concerning the LP optimization as compared to what is done elsewhere, as in the USA for example. In France, the type of fuel management is fixed for some of the 55 nuclear reactors: 4-cycle UO_2, 3.70% for most of the 900 MWe PWRs, 3-cycle UO_2-PuO_2

for five 900 MWe PWR units, and 3-cycle UO_2, 3.10% for most of the 1300 MWe reactors. This standardization leads to economic gains in fuel management.

So EDF does not really optimize an LP on the cycle length because this one is defined generically for economic reasons. In fact, EDF "optimizes" its LPs upon physics constraints which are very severe: the radial peaking factors, computed in eight rodded configurations, must remain under a limit; the reactivity shutdown margin, computed at the end of cycle, all rods out except the most antireactive, has to be over a minimum; the moderator temperature coefficient and the discharge burnups have also to meet criteria. An LP has then to comply with many constraints, which makes the search not easy (in fact, in this case, the search is not really an optimization).

The differences between the EDF approach and other approaches are very important as far as the optimization softwares are concerned. These are mainly centered on an economic optimization and don't take into account a high level of validation constraints.

The problem of treatment of constraints consists of two parts:

- *Determination (or estimation) of constraint violations*

  In a modern LP optimization of PWRs one usually calculates cyclewise (LP) parameters for HFP, ARO conditions. This calculation can be either extended to get all other constraints, for non HFP conditions, or one can correlate non HFP parameters with HFP parameters for a particular LP. In either case we get information about constraint violations and transfer them to the optimization method.

- *Using constraints at the optimization level*

  In an optimization procedure one can treat constraints in two general ways: by rules, or by including them as penalty function values into the objective function. Both alternatives have advantages and drawbacks. Treating constraints using special rules makes the search faster, but limited to fewer combinations.

  Putting constraints into an objectives' function as penalties makes a searching space much bigger and the search is therefore slower, but a possibility of finding better solution might be higher.

Accurate determination of all important safety parameters is certainly the biggest problem in loading pattern optimization. To perform LP optimization using stochastic optimization method or expert system approach, one needs a lot of potential solutions to be compared. One needs correlations between safety parameters obtained from the physical model and safety parameters that are not modelled, in order to make a large number of LP evaluations a realizable task. It is a difficult problem to obtain good correlations, but it is solvable. Our experience is that an effort of building an automatic LP optimization tool pays off, both in future LP economics and expert-time consumption.

2.2 Expert System Codes

Expert System codes are usually written in either C-Language, PROLOG, or LISP. They allow efficient programming of logical rules which are employed in in-core fuel management. In all cases, the facts such as fuel element types, burnup, core geometry, etc. must be defined in the code so that the rules can be applied. In addition these expert system codes may have to work with codes in other languages, e.g. FORTRAN. Nevertheless, one trained to program these codes can find a solution. Expert system codes are needed for automatic reload of cores in the future, and hence are beginning to appear in in-core fuel management code packages.

In case of a small number of identical or similar power plants, the development of expert systems means rather shift of work than reduction of work.

Optimization without expert system requires an experienced and skilled engineer.

Development of an expert system requires an experienced and skilled engineer to define the constraints in terms of reactor physics. Besides, a skilled programmer (interpreter) may be required to translate these constraints into a proper expert system language.

The advantage of creating an expert system is that a bulk of work can be done off-line, i.e. independent from actual reload business.

Many of the safety constraints (key parameter substitutes) do not correlate well with the safety-relevant properties of the individual fuel rod. The resulting rules to be applied in an expert system, therefore, should be rather weak. Regarding this situation, it may be questioned that exact logic and mathematics are really the adequate tools, and it look promising to try the application of fuzzy logic.

2.3 Number of Energy Groups Used in Reactor Calculations

Within Siemens PWR reload design procedure, nodal reactor calculations are always carried out using 2 energy groups. Pin-by-pin solutions are constructed by dehomogenization of the nodal results using heterogeneous form functions for power, burnup and fluxes.

The heterogeneous formfunctions are based on 2D fine mesh calculations for each type of fuel assembly.

At present, the heterogeneous fine mesh calculations are carried out in 2 energy groups for Uranium fuel assemblies. In case of MOX fuel assemblies, 10 energy groups are used.

In India, the operating power reactors are of two types: BWRs and PHWRs. The lattice cell calculations in both the cases use a large number of energy groups. For core calculations for BWR, a one-group nodal code similar to FLARE code is used. The reflector is treated through albedoes which have been tuned. The albedoes do not change from cycle to cycle. The EOC K-eff is normalized to 1.005-1.010. The power distribution is predicted within 5% of TIP readings. The prediction of notchworths is not satisfactory.

PHWR core calculations are done using two-group three-dimensional finite-difference codes. The reflector is treated explicitly through cross-sections. The criticality predictions are within 5 mk (0.5%) whereas power distribution prediction is within 5%.

BACKGROUND PAPER

(Session 1)

HISTORY AND STATUS OF RELOADING TECHNIQUES FOR LIGHT WATER REACTORS

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Abstract

The history and status of reloading techniques for light water reactors is presented by covering both the techniques used to reload the cores and the nucleonic codes used to calculate the core performance characteristics. The nucleonic codes involve a cross section generating code and a core analysis code. The evolution of these codes is described beginning with the early LEOPARD code and then going to present day WIMS, CASMO, CPM2, etc. for the cross section generating codes. The modern accurate nodal codes with pin power reconstruction is presented as the most advanced method for analyzing reloads. The basic equations employed in modern cross section generating codes and modern nodal codes are developed. Optimization techniques used to reload nuclear reactors are briefly reviewed, i.e., giving a short history on this subject and ending with modern optimization techniques. Special emphasis is given to the author's method of using one code system to perform automatically all of the tasks needed to reload the reactor. Two separate codes are described, one involving a combination of C-language and FORTRAN, and the other completely written in FORTRAN. The former is designed for the Westinghouse Beaver Valley PWR's and the other for the TMI-1 PWR. Each code contains the optimization calculations needed to produce an optimal reload design with all of the numerous constraints needed for a practical core. Both codes, however, employ the same basic approach. They begin with a priority loading plan (its source described in the paper) which initially loads the core in an optimum manner with the available fuel. The core pattern is then automatically modified to meet all constraints using the Haling power distribution. Then the burnable poisons are optimally placed in the core. The final design is automatically depleted in the normal manner. These codes can load a core equal to or better than any experienced fuel manager.

I. Introduction

The reloading of the nuclear power reactor requires calculating the core reactivity, power distribution, and isotopic inventory to maintain adequate safety margins and operating lifetime for each core. In addition, the selection of reloading schemes is made to minimize energy costs[1]. These analyses have traditionally begun by using scoping codes which allow fast determination, at some sacrifice of accuracy, of the reloading pattern's characteristics. These scoping calculations quickly screen out the undesirable patterns leaving a cadre of configurations acceptable for future more accurate analyses. This is just the beginning of determining the reload configurations. Extensive and expensive safety calculations, involving thermal hydraulics, transient analysis, etc. maybe be performed to insure compliance with safety standards under an array of assumed power plant accidents.

The numerous calculations and decisions made during this process run the full gamut of reactor physics calculations involving sophisticated coupled thermal hydraulics code. This is particularly true if some of the safety calculations must be repeated. In addition, a knowledge of core behavior is needed to make

decisions when the different reloading patterns are chosen for analyses. It should be mentioned that, when the first core is accepted for the power plant, an envelope of criteria are established which are used for accepting reloads without performing all or any of the safety calculations. Such criteria are maximum normalized power or power density and temperature coefficient for PWRs and some additional criteria for BWRs. Using these criteria, reloads for the second and subsequent cycles can be designed based on simple criteria. Only in very special cases are the safety calculations repeated for a reload.

A reload involves removing depleted or used fuel assemblies from the shut down core, and replacing the removed fuel with fresh fuel. The used fuel is moved to other places in the core, the degree of this shuffling of fuel depends on whether it is a PWR or a BWR. Except for the central fuel assembly, all used fuel are usually moved in a PWR whereas, only some of the used fuel is moved in a BWR.

There are various strategies used to design the reload pattern. The out-in pattern is one wherein fresh fuel is loaded in the periphery of the core, and in subsequent cycles moved inward. The in-out pattern is the reverse. In the first few decades of PWR operation, the out-in patterns were employed. More recently, the in-out procedure has replaced the out-in method to obtain low leakage cores and conserve ^{235}U. However, low leakage cores require the use of burnable poisons. BWRs use the in-out design, placing the fresh fuel inside the core and using depleted or natural uranium blankets on the periphery. In both PWR's and BWR's scattered loading strategy is used for reloading the cores.

Any reload design must be verified by calculations. Such calculations are now very sophisticated starting with the scoping codes and completing the analyses with more complicated highly accurate codes. These codes are generally divided into two sets of reactor physics codes. The first is the cross section generating codes which result in two group cross sections, a fast group and a thermal group. The basic concept here is to take the fundamental cross section data as given in ENDF/B[2] format and via reaction rate equivalence compute the two group constants as shown in Fig. 1. Not shown in Fig. 1 is the code NJOY[3] which is the modern code for transforming ENDF/B data to cross section libraries. NJOY now replaces FLANGE II and/or ETOM and ETOG in Fig 1.

The LEOPARD code was the first to perform this task[4]. It consists of two codes, MUFT to calculate the fast group cross sections, and SOFOCAT to determine the thermal cross sections. MUFT is a 54-group Fourier-transform slowing down code which uses the B_1 and Greuling-Goertzel approximations to calculate the fast group cross sections. MUFT makes two major approximations that limit its accuracy. The fast absorption cross section depends on a derived resonance escape probability developed from Helstrand's experiments.[3] Helstrand's data used a regular array of metal rods in a light water moderator. Hence, when actual fuel assembly heterogeneities are considered, particularly if burnable poison rods are included, the data may lead to errors. Also, metal to water ratio greater than 0.5 will produce results that are suspect because Helstrand's data does not extend beyond this range. The SOFOCAT code has 172 thermal groups and uses the ABH method to homogenize the group constants and the Wigner-Wilkins thermal spectra to obtain the neutron energy flux spectrum.[5] There is no up-scatter in the various thermal groups and the Wigner-

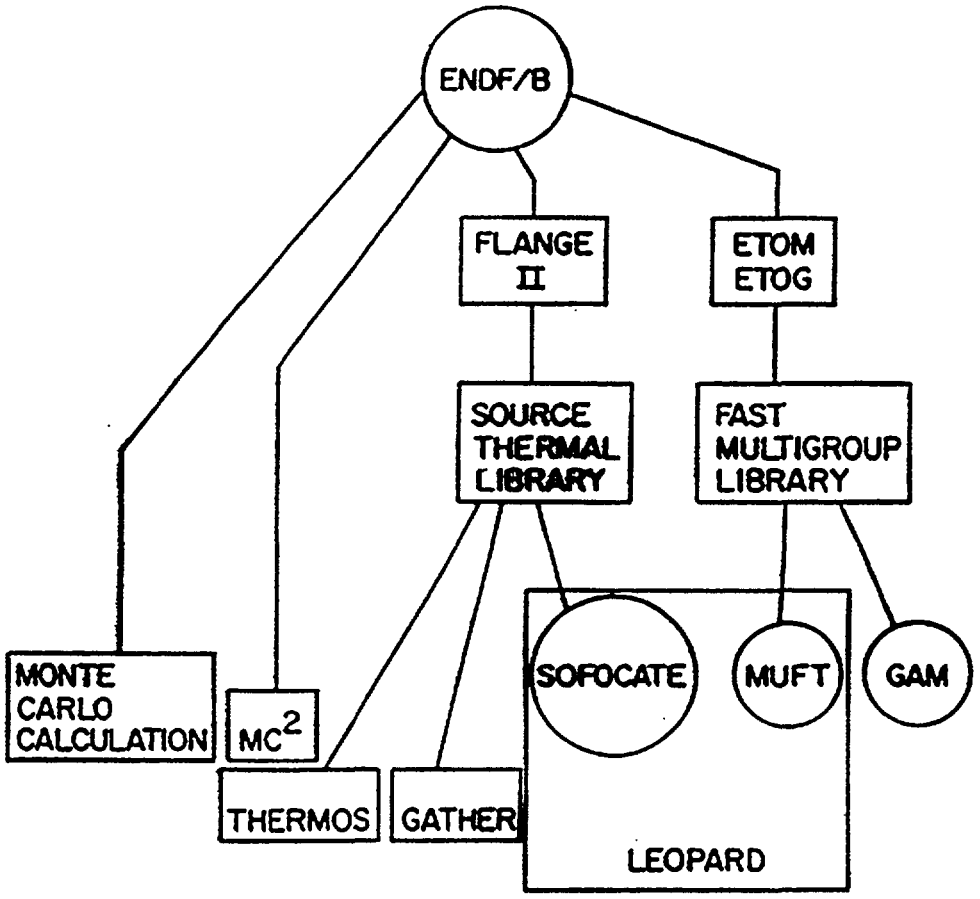

FIG. 1. Block diagram of computer codes used to generate cross-sections.

Wilkins assumes the moderator is a hydrogen gas. Although this code is not expected to be precise, it provided useable cross section data for the first 1 or 2 decades of the operation of nuclear power plants.

The WIMS code[6] introduced a more accurate means for computing the group constants. M.J. Halsall[7] provides an excellent review of the WIMS development which started in late 1963. Today, the WIMS code starts with 69 group data (14 fast, 13 resonance, and 42 thermal). The major improvement in the WIMS treatment involves the more accurate calculations of the resonance effects and the thermal neutron calculation. The basic method utilizes the equivalence concept to take typical heterogeneous geometries and compute equivalent homogenous group constants. In this method collision probability is used as the basis for computing the reaction rates. The WIMSD is designed to handle pencils, slabs, and clusters, but is not designed for annular fuels.

The CPM-2[8], CASMO[9], and PHOENIX[10] codes are designed on the WIMSD model using a similar 69-group nuclear data file. These codes use the following techniques[11]:

(1) CPM-2 uses Collision Probabilities
(2) CASMO uses Transmission Probabilities
(3) PHOENIX uses Discrete Integral Transport Solutions

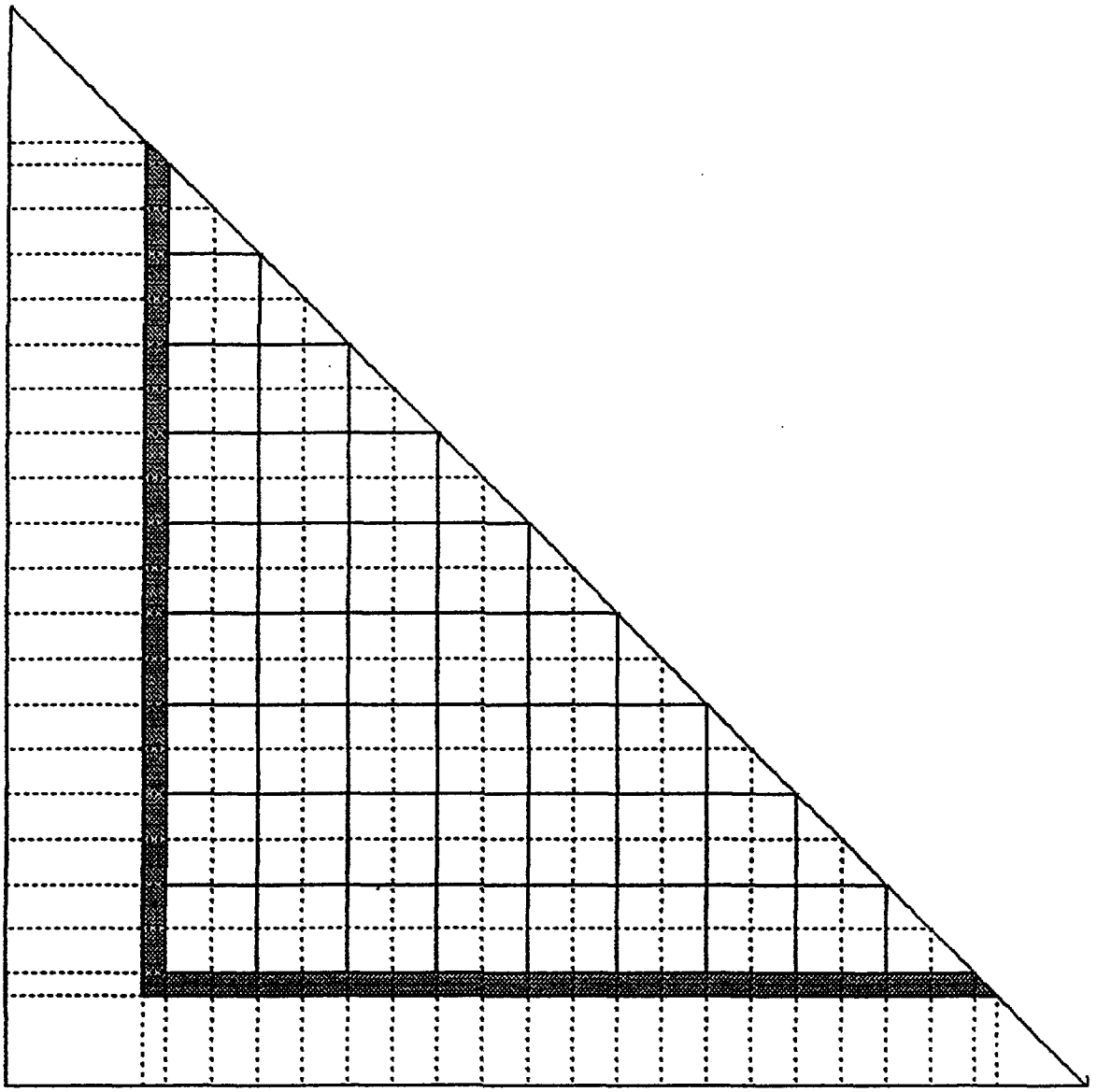

FIG. 2. Geometry used by CPM-2 in 2-D calculation.

In addition, these codes analyze a detail 2D geometric description of a fuel assembly as shown in Fig. 2.

The group cross sections are then used in a core analysis diffusion theory code to provide the core characteristics during core operation. There have been a large number of such codes produced for general use. For fuel management purposes a two dimensional code may suffice for PWR analysis, but clearly a 3 dimensional code is required for BWR analyses. Although 1½ neutron energy groups code have been used in the past and continue to be used in some cases, for future use it is strongly recommended that the minimum number of groups used should be two groups. The 1½ groups approximation assumes the ratio of fast to thermal flux, $\frac{\phi_1}{\phi_2}$, is independent of position in a fuel assembly. This is not true in scattered arrays of different fuel assemblies.

The finite difference codes such as MCRAC[12] (EXTERMINATOR-2)[13], COBAYA-2[14], CITATION[15], VENTURE[16], and PDQ-7[17] are either 2 or 3 dimensional and 2 or multigroup codes. MCRAC is a 2 dimensional, 2 group code with core depletion capability, COBAYA-2 is a component of the PWR Core Analysis System, SEANAP, and is used to perform the detailed pin-by-pin 2 group 2 dimensional calculation as a function of burnup and different thermal hydraulic conditions. CITATION, VENTURE, and PDQ-7 are 3 dimensional, multigroup diffusion theory codes used to obtain benchmark data; however, PDQ-7 is a standard PWR depletion code used to help license PWR reload designs.

Today, the more modern core analysis codes are nodal and most are commercial. The MCRAC code has been transformed into a modern 2-D nodal code ADMARC, but one designed specifically for a particular power plant, either the TMI-1 or the BEAVER VALLEY Unit 1 and 2 reactors. Other core analysis nodal codes are[18]:

ANC/POLCA	Westinghouse
PANACEA	General Electric
ROCS/PDQ/M	Combustion Engineering
FLAME/PDQ	Babcock & Wilcox
POLCA	Asea - Atom
PRESTO	Scand power
SIMULATE	Studsvik
MEDIUM/PIN POW for PWR	KWU
RS3D for BWR	KWU

SEANAP[19] also uses nodal codes, e.g. LOLA in conjunction with its other in-core fuel management codes.

In summary, the present most advanced methods for performing in-core fuel management employ accurate cross section generating codes like CASMO, WIMS2D, and CPM-2 and use advanced nodal codes for core analysis. The out-in loading patterns used in the early decades of nuclear power have and are now being replaced with low leakage cores that employ burnable poisons as developed via optimization methods. Also, a scattered loading pattern is used to reload the core.

Optimization studies began in the nineteen sixties. Some of the first optimization studies minimized the costs directly using very coarse nucleonic models. One of the very first was by Wall and Fenech[20] who performed a multicycle optimization analysis employing a one-dimensional three region equivalent volume core model to minimize fuel costs directly. They developed an optimal sequence for zonal refueling of the core.

Melice[21] was the first to apply sequential analysis of separate optimization calculations to provide an overall optimization analysis. His method was to maximize the beginning-of-cycle (BOC) k_{eff} of a PWR subject to power peaking and other safety constraints. He used the out-in reload method thus defining the placement of fuel in the core periphery. Even though this method precluded the use of low-leakage cores, fuel costs were reduced.

Sauar[22] was among the first to apply optimization to a BWR. He employed linear programming to minimize the costs directly. Through a sequence of optimization steps, Sauar optimization process ended with a direct loading using the three dimensional code, FLARE, to provide final results. Many other beneficial studies were performed by Terney[23]; Suzuki and Kiyose[24], and Naft and Sesonske[25] to provide direction in improving the optimization of reloads. They used the Haling power distribution to greatly simplify the depletion analysis during the optimization calculation and proved that the end-of-cycle (EOC) condition defined the optimum strategy. Huang and Levine[26] used the Lagrange Multiplier method to optimize the EOC k_{eff} of the TMI-1 reactor. Their calculations produced low leakage cores at EOC. Chang and Sesonske[27] used an exhaustive direct search algorithm to optimize a low-leakage configuration for the PWR. They maximized the core's power density within the maximum allowed fuel assembly constraints. The potential benefits of using lumped burnable poisons (LBPs) to effect low-leakage cores were studied in detail, and it was concluded that low-leakage cores using LBP's were promising.

These early studies on optimization technique for refueling cores laid the foundation for the current techniques which can automatically reload cores in an optimum manner. This is described in Section III on optimization. In Section II, a brief description is presented of methods used by industry and others to perform the reload calculations.

Section IV describes how expert system codes are employed with the optimization calculations to eliminate manual tasks required to complete the reload design and provide a completely automatic method for determining an optimum reload. The conclusion is given in Section V.

II. Code Description

There are many codes, as described above, that are used in in-core fuel management. In this section, a brief summary is given of the modern cross-section generating codes and how PSUI-LEOPARD uses the CPM-2 code to provide similar accuracy. Also, a very brief review of nodal codes is presented. It is important to describe these codes first, because they are the tools for performing in-core fuel management.

II.1 Cross Section Generating Codes

The modern cross section generating codes such as CPM-2, CASMO, and WIMS2D permit an exact geometric description of the fuel assembly although some approximations are made in the calculations. They employ two-dimensional integral transport equations to develop solutions. The first step is to use a supercell type analysis of each different pin type in the fuel assembly to obtain the corresponding 69 energy group spectra. The geometry for this type calculation is a supercell established similar to that for the LEOPARD calculation. Once these calculations have been performed and the 69 group reaction rates have been homogenized over each of the supercells, the program collapses the reaction rates to a smaller number of groups (ie. 25) conserving the reaction rates. This is followed by making another simulation calculation using the 25 groups to improve the pin by pin geometry spectrum and multigroup cross sections. In this

calculation the complete fuel assembly is cylindricized and a one dimensional calculation is made to obtain the 25 group fluxes and reaction rates in each cylindrical region. This provides group cross-sections, approximately flux weighted, for each pin in the fuel assembly for developing a fewer ($\leq$12 groups) group structure. Using the fewer group structure (the user chooses the number of groups), a detailed two-dimensional transport theory solution is them obtained, with the pin by pin geometry of the fuel assembly. This solution provides the accurate fuel assembly two group cross-sections.

At this point, the fuel assembly is depleted in burnup steps. This requires that at the end of each burnup step, a new complete calculation be performed with the new number densities, beginning with the 69 group calculation for each different type of pin in the fuel assembly, followed by collapsing the groups to fewer groups as described above. The calculations at the end of each burnup step is continued until the end-of-life (EOL) is reached.[11]

The calculational method for the two-dimensional integral transport theory begins with the steady state Boltzmann equation:

$$\underline{\Omega}\cdot\nabla\phi\ (\underline{r},E,\underline{\Omega}) + \Sigma(\underline{r},E,\underline{\Omega}) = \int\int \Sigma_S\ (\underline{r},E'\rightarrow E,\underline{\Omega}'\rightarrow\underline{\Omega})\ \phi\ (\underline{r},E',\underline{\Omega}')\ dE'\,d\underline{\Omega}' + S(\underline{r},E,\underline{\Omega}) \tag{1}$$

The solution to this equation in multigroup formulation of the flux $\Phi_g\ (\underline{r})$ at position $\underline{r}$ is:

$$\phi^g(\underline{r}) = \int_o^\infty q^g(\underline{r}')\ \frac{\exp[-\rho_g(\underline{r}'\rightarrow\underline{r})]}{4\pi\ s^2}\ d^3r' \tag{2}$$

where

g = group

and the source term is

$$q_g(\underline{r}') = \int_{E_g} dE \int_o^\infty dE' \int_{4\pi} d\underline{\Omega} \int \Sigma_S(\underline{r}',E'\rightarrow E,\underline{\Omega}'\rightarrow\underline{\Omega})\ \phi\ (\underline{r}',E,\Omega')\ d\Omega' + \frac{\chi_g}{k}\int_o^\infty \nu\,\Sigma_f(\underline{r}',E') \int_{4\pi} \phi\ (\underline{r}',E',\underline{\Omega}')\ d\underline{\Omega}'\ dE' \tag{2-1}$$

where

$$\rho_g\ (\underline{r}'-\underline{r}) = \int_{E_g} dE \int_o^s \Sigma_t\ (\underline{r}-s'\left(\frac{\underline{r}-\underline{r}'}{s}\right), E)\ ds' \tag{3}$$

Σ_t = macroscopic total cross section for energy E

s = *neutron direction distance from* r' *to* r *or* $s = |r - r'|$

and the other terms are standard.

The neutron source q_g ($\underline{r}'$) at position $\underline{r}'$ is the sum of the fission and scattered neutrons appearing in group g.

Eq (2) is then transformed into Eq (4) by conserving reaction rates as follows:

$$\Sigma_m^g \phi_m^g V_m = \sum_n \int_{V_m} d^3r \int_{V_n} q_g(r') \frac{\exp[-\rho_g(r'-r)]}{4\pi s^2} d^3r' \Sigma_m^g \qquad (4)$$

where

To formulate the method of collision probabilities used in CPM-2, the above equation is modified to be:

$$\Sigma_m^g \phi_m^q V_m = \sum_n q_n^g P_{n\to m}^g \qquad (4)$$

where

q_n^g = average neutron source for group g in region n

$$P_{n\to m}^g = \int_{V_m} d^3r \int_{V_n} \exp\frac{[-\rho_g(r'-r)]}{4\pi s^2} d^3r' \Sigma_m^g \qquad (4\text{-}1)$$

and

$P_{n\to m}^g \equiv$ the first-flight probability which is the probability a neutron in group g, born isotopically in region n has its first collision in region m.

Eq. (4) can also be written as

$$\phi_m^g = \sum_n T_{n\to m}^g q_n^g \qquad (5)$$

where

$$T_{n\to m}^g = \frac{P_{n\to m}^g}{\Sigma_m^g V_m} \qquad (6)$$

and

$T^{g}_{n\to m}$ is called the Transport Matrix. Note that once the $P^{g}_{n\to m}$'s in Eq (4-1) are solved for a configuration and energy groups, the Transport Matrix can be calculated using non-geometric variables. i.e., $\sum^{g}_{m}$ *and* V_m

The above equation is modified to represent an infinite array of supercells with the BP in the center of each supercell, and then solved using the Bickley-Naylor function of order 3. CPM-2 first collapses the 69 groups solution to 25 groups and then cylindricizes the fuel assembly as described previously. These calculations provide the two group constants homogenized for a fuel assembly as a function of burnup, temperature, etc. The B_1 approximation is used to obtain leakages affects in the CPM-2 cross sections. Also, as described previously, the fuel assembly is depleted in steps in the standard manner.

Other professional codes, like CASMO and WIMS2D, follow a similar technique. The PSU-LEOPARD codes normalize their results to CPM-2 calculational data[28]. The following scheme is used to make this normalization.

(1) The 3 fast group resonance escape probability and Pu production are separately modified in PSU-LEOPARD as a function of BP, burnup, and enrichment to give correct results.

(2) The thermal depletion of the BP solid rod calculation is computed separately using transport theory calculational techniques (collision probability theory), and the BP flux to supercell flux ratio in CPM-2 is used to normalize these results so as to obtain absolute values of the fluxes.

(3) The k_∞ of the fuel assemblies are then made to agree by calculating the flux peaking factor appropriately in PSU-LEOPARD.

There are two separate PSU-LEOPARD codes, one for the TMI-1 reactor and the other for the two Beaver Valley PWRs. This is because the TMI-1 reactor is built by B&W and has only one BP design. It always uses 16 pin geometry and varies the boron content by the percentage boron put in the pins. The Beaver Valley reactors are of the Westinghouse design which has 3 separate and distinct designs, i.e., Pyrex BP with air region in the center, WABA (wet annular burnable absorber), and IFBA (Integral Fuel Burnable Absorber). The Pyrex and WABA BP's keep the same amount of boron in the pins and vary the boron content by having designs with different numbers of pins, i.e., 4,8,12,16, and 20. The IFBAs vary the boron content by having two types of Zr B_2 coatings on the selected fuel pins. They have a standard coating and another with 150% thicker coating. Also, the number of fuel pins coated with the boron in a fuel assembly vary. The WABA and TMI-1 BPs have a Al_2O_3-B_4C matrix whereas the Pyrex BPs are of borosilicate glass.

This requires the Beaver Valley PSUI-LEOPARD to have 4 subroutines, BVF, PYREX, WABADP, and IBFADP to perform the analyses for all of Westinghouse BP designs. The subroutine BVF calculates volume fractions for the various

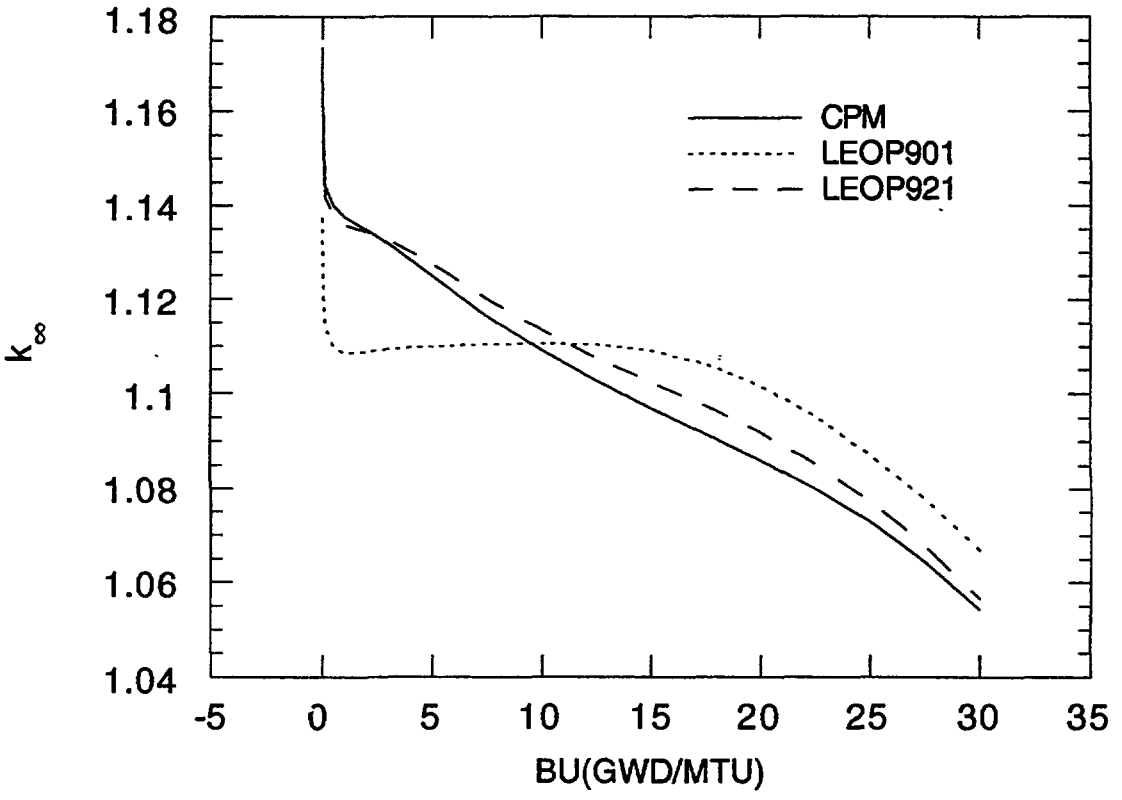

FIG. 3. k_∞ comparison for the case with 5.5 wt% enrichment and 4.2 wt% BP.

designs. The remaining three subroutines then perform the analyses for the three different BP types as indicated by their names.

Figures 3 and 4 show final results of these calculations. Fig. 3 compares CPM-2 and PSUI-LEOPARD k_∞ for the TMI-1 reactor and Fig 4 does the same for the Beaver Valley Reactor with WABAs. Shown on both of these figures are the recent improvements made on these codes. LEOP921 and NLEOPARD, in Figures 3 and 4, show results of these improvements on the TMI-1 and Beaver Valley PSUI-LEOPARD codes, respectfully.

II.2 Core Analysis or Simulator Codes

The key to performing accurate core analysis is to have available accurate two group cross-sections calculated by the cross-section generating codes. These cross-sections can then be employed in a two or three dimensional 2 group core analysis code with thermal feedback to obtain the power distribution and k_{eff} at the beginning of cycle (BOC), and at each subsequent burnup step. To use a code such as VENTURE or PDQ7 for a BWR is out of the question, because of the need to compute void production and control rod position in a three dimensional configuration. The time required to perform this calculation is too long for most computers. These type calculations are also very expensive to use for BWRs, and every PDQ7 calculation performed for a PWR is normally for licensing and benchmark purposes. A calculation for licensing must be able to obtain maximum pin-power density values. For nodal codes, this requires reconstruction of the pin-power distribution in the hottest fuel bundles.

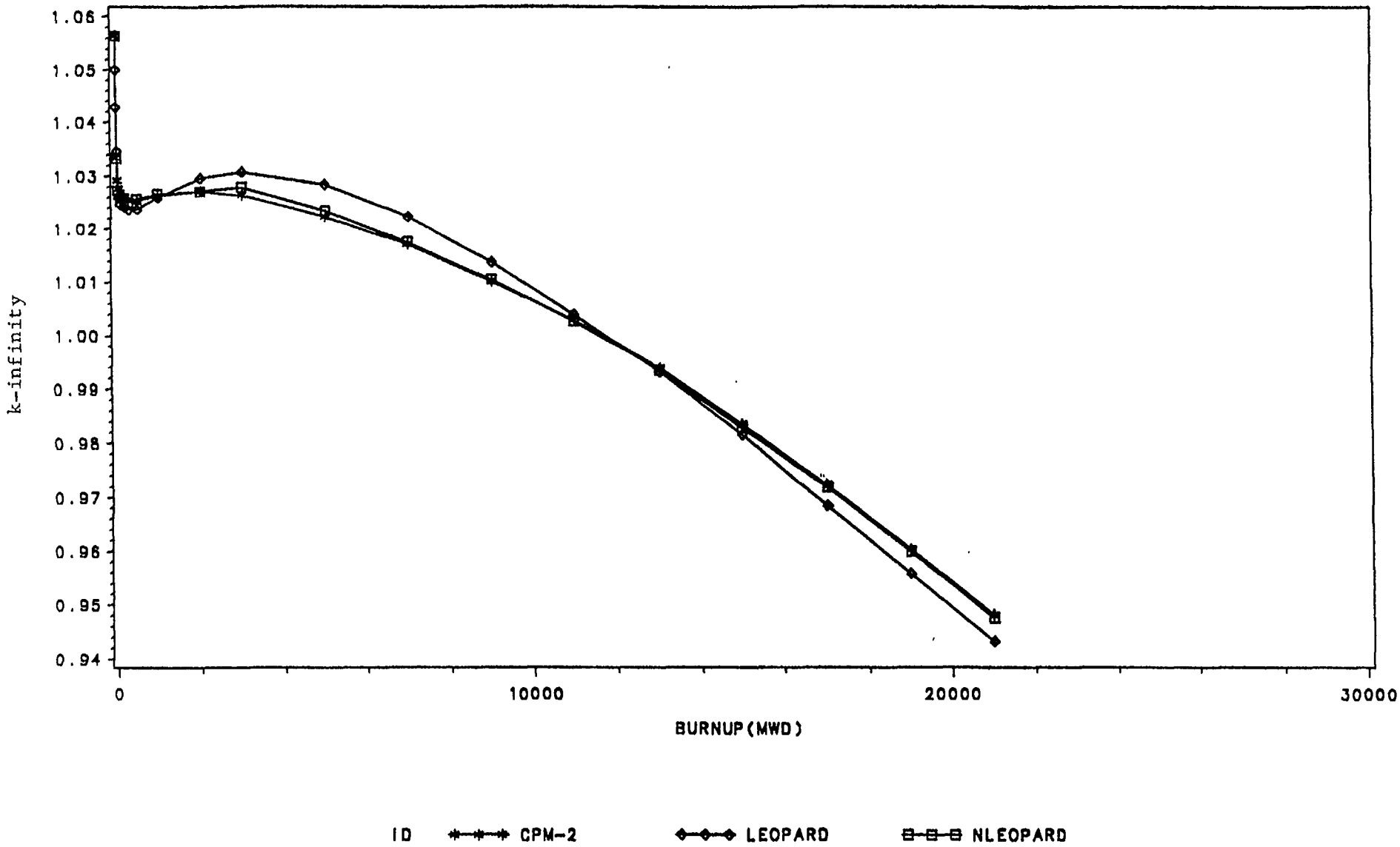

FIG. 4. Comparison of Beaver Valley-1 k_{∞} using WABA BPs.

Modern nodal codes with pin-power reconstruction are, in my opinion, the best method for doing this. The modern nodal code analyzes the core with four nodes per fuel assembly and with 12 to 16 nodes axially. This code computes accurate power distributions and k_{eff}'s without the need to normalize to multigroup finite difference codes. Modern nodal codes begin with solving the transverse-integrated diffusion equation.

The process[28], then is to integrate the multi-group diffusion equation, Eq. (7), over the two directions v and w transverse to the u-direction (u = x or y or z) of a given node i:

$$\int_{-a_v}^{+a_v} dv \int_{-a_w}^{+a_w} dw [-\nabla \cdot D_g (\underline{r}) \nabla \phi_g (\underline{r})] + \int_{-a_v}^{+a_v} dv \int_{-a_w}^{+a_w} dw \Sigma_{tg} (\underline{r}) \phi_g (\underline{r})$$

$$= \int_{-a_v}^{+a_v} dv \int_{-a_w}^{+a_w} dw q_g (\underline{r}) \qquad (7)$$

where

$q_g (\underline{r})$ = sum of fission and slowing down source term for group g

$$a_t = \frac{\Delta t}{2}; \quad t = u,v,w \quad ; \quad v,w = x,y,z \quad , \quad u \neq v \neq w$$

and

Δt = width of node i in the t-direction.

After integrating Eq. (7), it can be shown that it reduces to:

$$-D_{ig} \frac{d^2 \phi_{ig}(u)}{du^2} + \Sigma^i_{tg} \phi_{ig} (u) = Q_{ig} (u) \qquad (8)$$

where

$$Q_{ig}(u) = \sum_{g'=1}^{G} \Sigma^i_{g'g} \phi_{ig'} (u) + \chi_g \sum_{g'=1}^{G} \nu \Sigma^i_{fg'} \phi_{ig'} (u) \qquad (8\text{-}1)$$

$$-L^i_{vg} (u) - L^i_{wg} (u)$$

and

$$\phi_{ig} (u) = \int_{-a_v}^{+a_v} dv \int_{-a_w}^{+a_w} dw \phi_g (u,v,w) / \Delta v \Delta w \qquad (8\text{-}2)$$

$$L^{i}_{tg}(u) = \int_{-a_w}^{+a_w} dw \int_{-av}^{+a_v} -D_{ig} \frac{\partial^2 \phi_g(u,v,w)}{\partial t^2} \, dv/\Delta v \Delta w, \qquad t=v,w \tag{8-3}$$

In developing the modern nodal code, use is made of the Green function using the following one-dimensional point source diffusion equation for the node i,

$$-D_{ig} \frac{d^2 \psi_{ig}(u_o,u)}{du^2} + \Sigma^{i}_{tg} \psi_{ig}(u_o,u) = \delta(u-u_o)$$

$$-a_U \leq u_o \leq +a_U \qquad -a_U \leq u \leq +a_U \tag{9}$$

where $\delta(u-u_o)$ is the Dirac delta-function.

Also used are two intermediate functions,

$$F(u) = -\psi_i(u_o,u) \frac{d\phi_i(u)}{du} \tag{10}$$

and

$$G(U) = \phi_i(u) \frac{d\psi_i(u_o,u)}{du} \tag{11}$$

Using Green's theorem and method of solution, the result is the fundamental nodal equation that is the basis of modern nodal codes.[28]

$$[1 + D_i \frac{d\psi_i(\pm a_U, \pm a_U)}{dn_U}] \phi_{i\pm U} + D_i \frac{d\psi_i(\pm a_U, \mp a_U)}{dn_U} \phi_{i\mp U} +$$

$$\psi_i(\pm a_U, \pm a_U) J^{n}_{i\pm U} + \psi_i(\pm a_U, \mp a_U) J^{n}_{i\mp U} = \int_{-a_U}^{+a_U} \psi_i(\pm a_U, u) Q_i(u) \, du \tag{12}$$

where

$\phi_{i\pm U}$ = surface flux at ($u = \pm a_U$)

$$J^{n}_{i\pm U} = -D_i \frac{d\phi_i(u)}{dn_U}\Big|_{u=\pm a_U} \tag{12-1}$$

There remains considerable mathematical manipulation before the final nodal code used in a core analysis computer program is attained. This will not be developed here. Use can be made of Moon's Ph.D. thesis[29]. Suffice to say that accurate

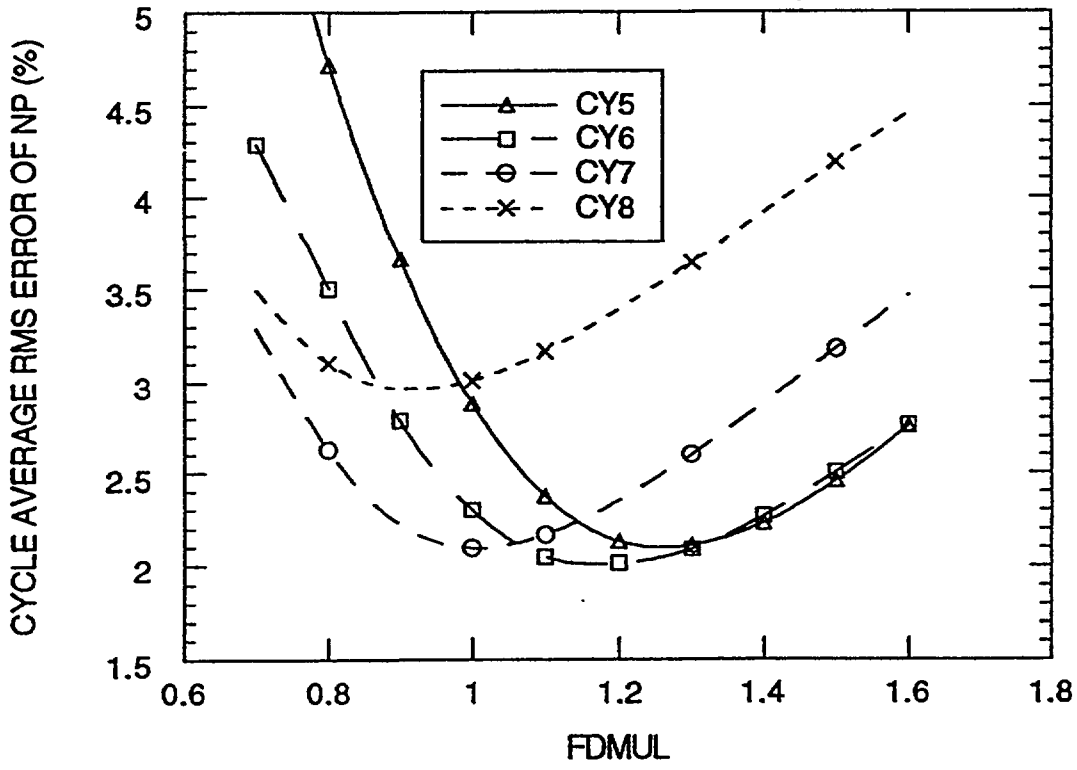

FIG. 5. FDMUL effect on cycle average RMS error of normalized power.

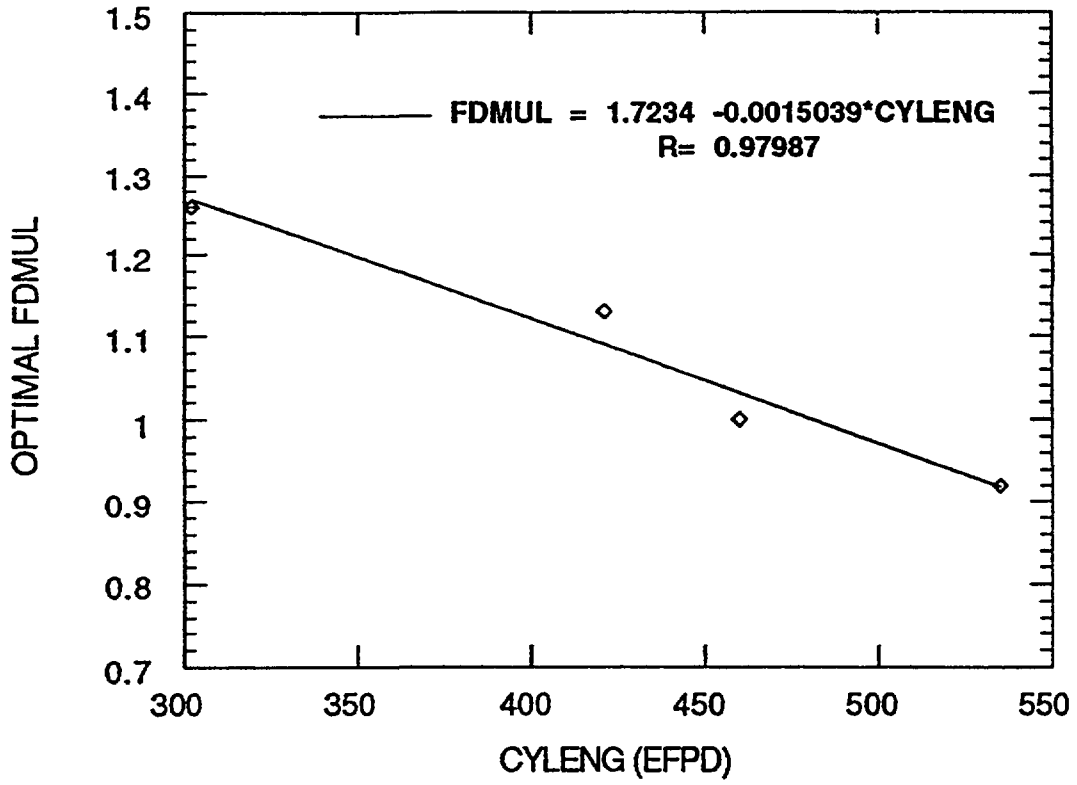

FIG. 6. NP optimal FDMUL as a function of cycle length.

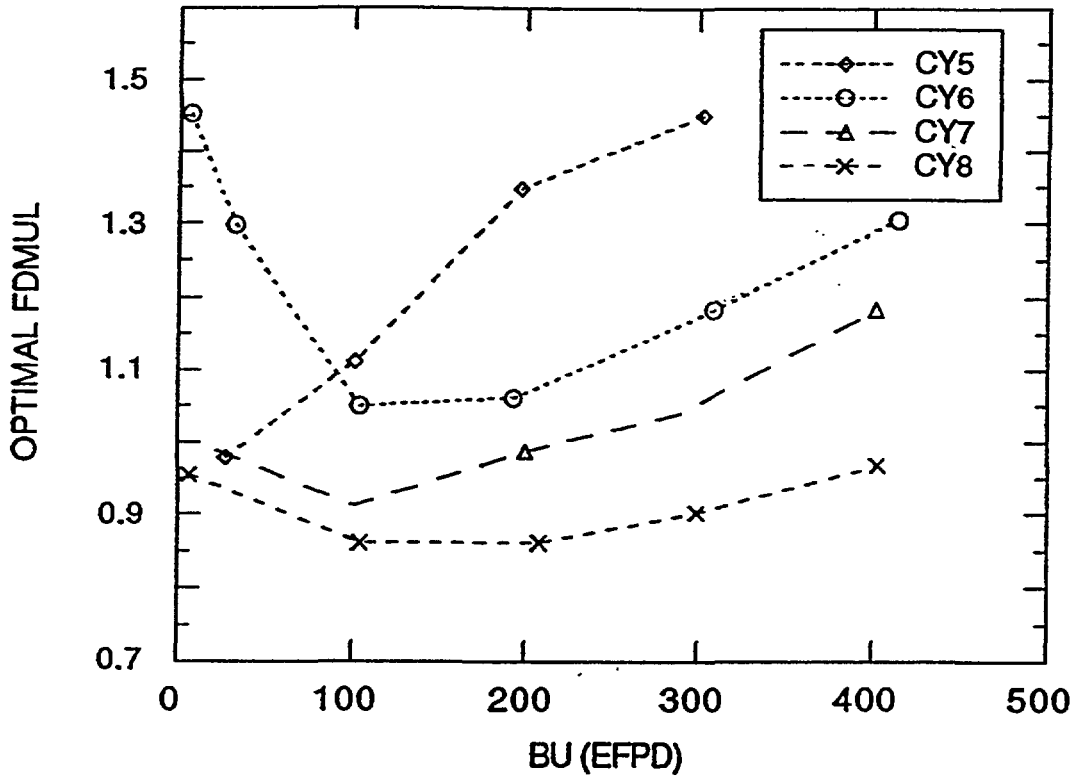

FIG. 7. Burnup dependent FDMUL for cycles 5–8.

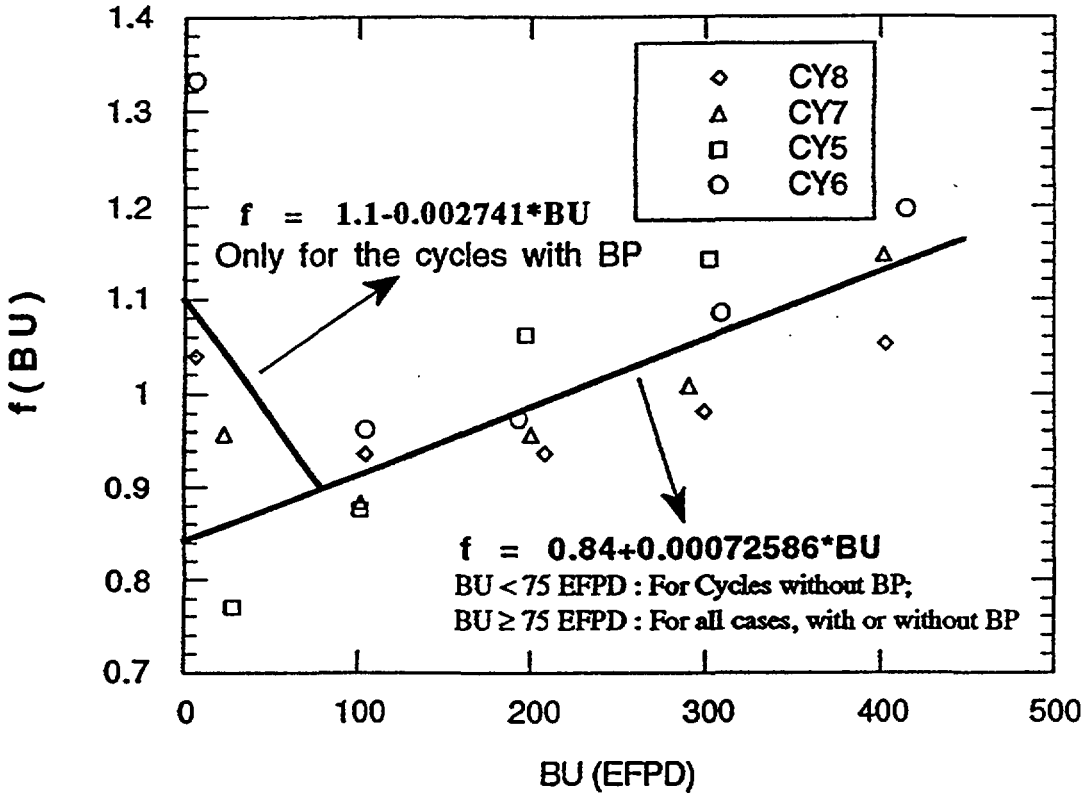

FIG. 8. Burnup dependent scaling factor f(BU) for FDMUL.

FAST DIFFUSION COEFFICIENT

$$D_1$$

$$D_1 \text{ (correct)} = f_1 \text{ (L)} * f_2 \text{ (BU)} * D_1 \text{ (LEOPARD)}$$

$$f_1 \text{ (L)} = (1{,}7234 - 0.0015039 * L)$$

$$f_2 \text{ (BU)} = a + b * BU$$

$$D_1 \text{ (correct)} = FDMUL * D_1 \text{ (LEOPARD)}$$

$$D_1 \text{ (correct)} = (1{,}7234\text{-}0.0015039L) * (a + b * BU) * D_1 \text{ (LEOP)}$$

FIG. 9. Fast diffusion coefficient correction.

calculations of the nodal fluxes are attained. Also derived are the surface fluxes and currents. These latter parameters together with the pin-power distribution calculated by a code like CPM-2 or WIMSD4 can be used to reconstruct the pin-power distribution.

One of the problems with modern nodal codes is that they are based on diffusion theory. This is valid everywhere in the core, but not at the core-reflector interface. Thus some parameter modification must be made to obtain good practical results. This is achieved in ADMARC by modifying the fast diffusion coefficient D_1. The method used to modify D_1 becomes especially important for long lifetime cores, i.e. cycle lengths > 400EFPD[30]. For a core with a hard spectrum of long cycle length, a significant amount of fast neutrons will hit the baffle and be reflected back into the core. Using two parameters, core cycle length and core average burnup, to represent the core spectrum hardness, the multiplication factor f_d used to modify fast diffusion coefficient can be expressed in the following form:

$$f_d \text{ (cy,bu)} = f_c \text{ (cy)} \cdot f_b \text{ (bu)} \tag{13}$$

where cy is the cycle length and bu is the core average burnup during the cycle, and components f_c and f_b take the forms

$$f_c \text{ (cy)} = \alpha - \beta \cdot cy \tag{14}$$

$$f_b \text{ (bu)} = \gamma + \delta \cdot bu \tag{15}$$

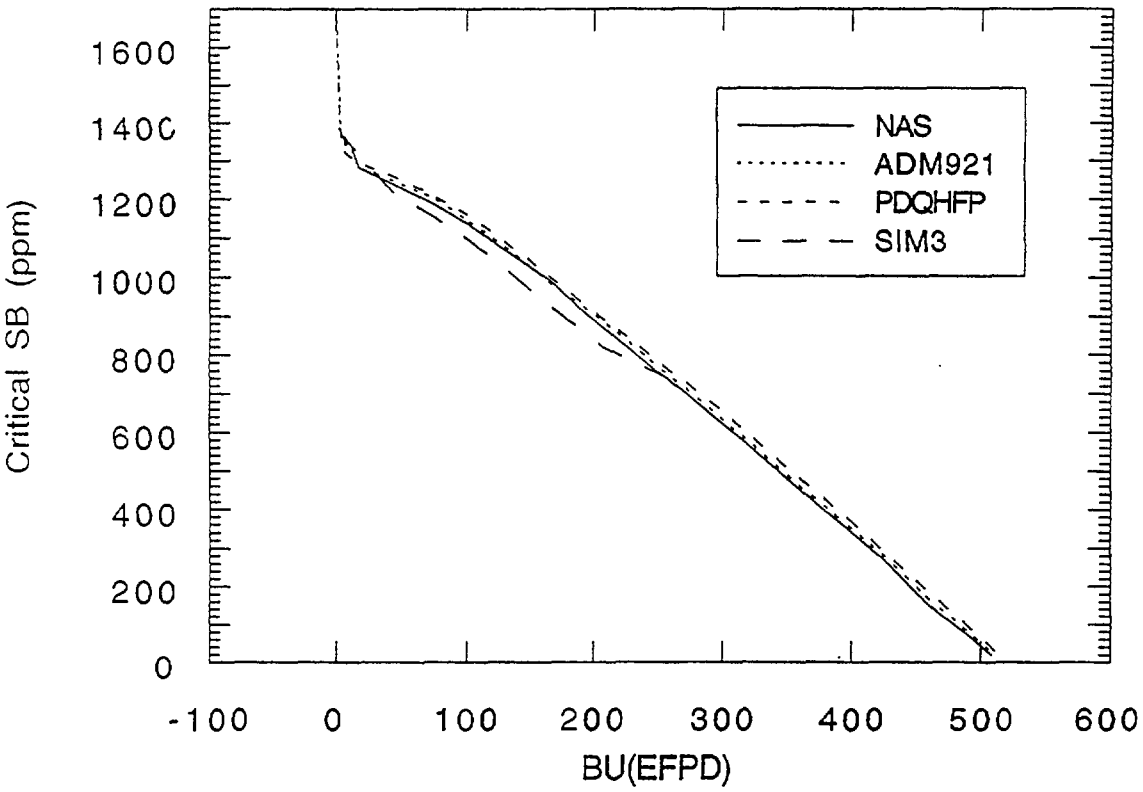

FIG. 10. TMI-1 cycle 8 critical boron letdown curves.

2-D NORMALIZED POWER DISTRIBUTION COMPARISION
BETWEEN NAS AND PFMP
EXPOSURE = 6. EFPD

	8	9	10	11	12	13	14	15.
	1.187	1.263	1.052	1.141	0.984	1.209	1.000	0.387
H	1.152	1.349	1.091	1.166	1.051	1.261	1.015	0.382
	-2.949	6.809	3.707	2.191	6.809	4.301	1.500	-1.292
		1.211	1.300	1.009	1.205	1.165	1.134	0.392
K		1.280	1.307	1.027	1.224	1.196	1.125	0.389
		5.698	0.538	1.784	1.577	2.661	-0.794	-0.765
			1.033	1.215	0.949	1.261	1.070	0.339
L			1.026	1.188	0.966	1.249	1.038	0.355
			-0.678	-2.222	1.791	-0.952	-2.991	4.720
				0.982	1.233	1.163	0.944	
M				0.992	1.228	1.139	0.938	
				1.018	-0.406	-2.064	-0.636	
					1.196	1.095	0.412	
N					1.186	1.024	0.399	
					-0.836	-6.484	-3.155	
						0.588		NAS
O						0.579		PFMP
						-1.531		%DIFF

RMS ERROR = 3.166

FIG. 11. TMI-1 cycle 8 NP comparison between measured and calculated values.

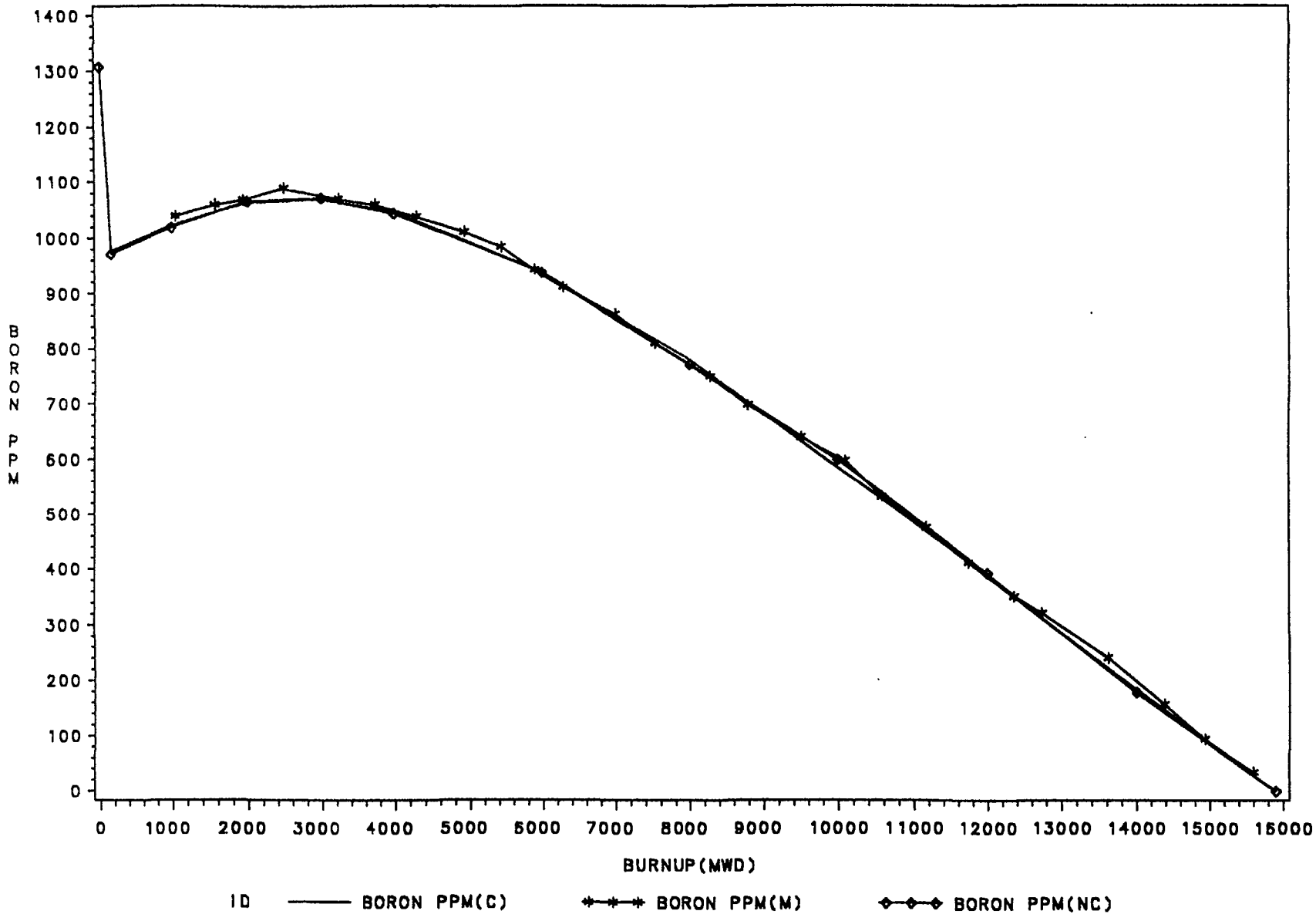

FIG. 12. Beaver Valley-1 cycle 7 boron letdown curve.

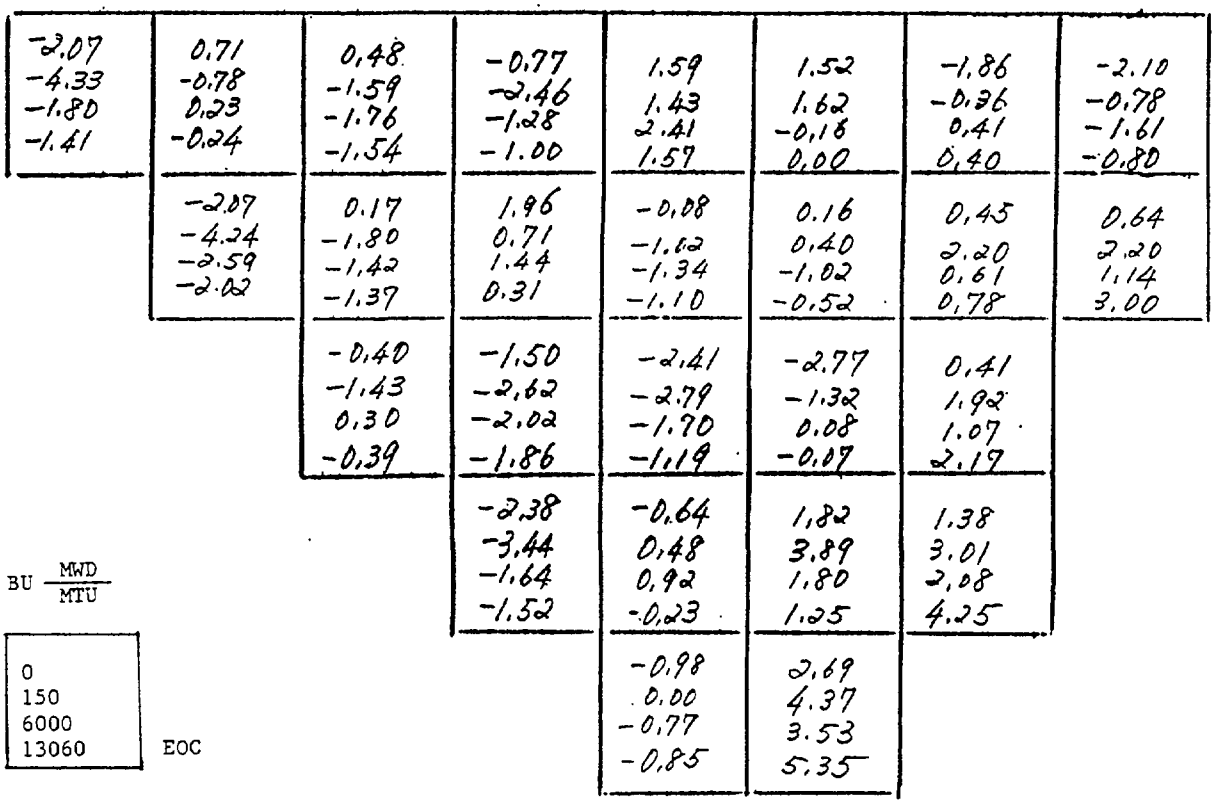

FIG. 13. Relative errors of normalized powers – Beaver Valley Unit 2, cycle 2.

The constants α, β, γ and δ are determined to minimize the difference between the calculated and the measured data. α, β, γ are all positive while δ may have different values depending on the range of burnup and whether or not the core is loaded with burnable poison. Figure 5 and Figure 6 show the effect of fd called FDMUL as a function of cycle length. Figure 7 is a plot of FDMUL that minimizes the cycle averaged rms error of NP. Figure 8 provides optimum FDMUL's as a function of burnup at each burnup step. Integrating these data produces the results shown in Figure 9 which are the equations for f_b. See Ref. 29 for more explanations.

Thus, by making the PSUI-LEOPARD calculation consistent with CPM-2 and by using the above D_1 calculation in ADMARC, accurate core calculations are performed by these two codes and referred to as the Penn State Fuel Management Package (PFMP). Fig 10 shows a comparison of ADMARC results for the soluble boron let down curve and Fig. 11 a comparison of normalized power distribution with measured data (NAS) and other codes for the TMI-1 core. Figs. 12 and 13 show the same comparisons for the Beaver Valley PWR's, respectively. ADMARC has been consistently providing 2D results that most of the time are as accurate as PDQ-7.

III. Optimization Techniques Used to Reload Cores

Six recent papers[31] [32] [33] [34] [35] [36] have attempted to develop automatic optimized practical reloads for PWRs using codes sufficiently accurate

to select potential reload configurations. These papers will now be briefly summarized; however, information from the latter reference,[36] is used extensively in this section. In fact, the majority of the work reported on this section has been recently performed by D. Bai, Li Zhian, and the author, because this is the most advanced work available for publishing that uses deterministic optimization techniques.

Morita et al.[31] uses a program that generates an analysis of a large number of loading patterns involving a four step process. The key to this method is that it solves a neutron diffusion equation in the backward direction to effect the user's specified target power distribution. The best candidates are chosen from the large number of patterns generated and then analyzed in the conventional manner. Kim et al.[32] developed a practical method using the SIMULATE-E code adapted to the CYBER-205 computer that can be used in the optimization of the PWR reload core design. Their analysis involved the use of Haling power distribution, and a discrete number of BP rods is used as control variables in this optimization problem. It is a fairly complex technique requiring several stages for solution. D.J. Kropaczek et al. [33] have developed a method to select optimum configurations using Monte Carlo techniques. They employed a so called Simulated Annealing method to restrict the number of acceptable loading patterns by a control parameter analogous to the solid temperature annealing process. They use the maximum end of cycle k_{eff} to choose the superior patterns. Although they establish a series of constraints, the potential number of trial configurations are enormous. i.e., of the order of 225,000 loading patterns. In general, this is basically an exhaustive search of all possible patterns that fall within the constraints. Galperin and Kimhy[34] have developed a knowledge-base heuristic search program using the LISP language to automatically reload a PWR. The code generates reload pattern based on depth first search in a tree pattern process. The final leaf nodes are evaluated using a nodal code. They have now developed a sophisticated package OPCON designed to enhance the performance of a fuel management engineer. This program is especially powerful when used to search for alternate reload patterns starting from one basic reload pattern. Using the rules in the program, OPCON automatically develops some superior reload patterns and evaluates them for future use. Suh and Levine [35] developed optimized automatic reload programs for pressurized water reactors using simple direct optimization techniques. Levine et al. [36] improved on Suh and Levine's method by minimizing power peak during depletion at a burnup step where the peak power is a maximum. These methods have been adopted with various modifications and introduction of new techniques to effect a practical design that automatically determined superior reload configurations for the TMI-1 and the Beaver Valley PWR's. In addition, these techniques have been incorporated within a single code to run the complete program automatically.[38] The TMI-1 and its various subroutines is explained in detail in Dan Bai's PhD thesis[30]. Li Zhian PhD thesis developed the various techniques and incorporated them into an expert system code for the Beaver Valley PWRs.[38]

In developing and applying these techniques, it is important to note that fundamental to all optimization techniques are the following requirements:

(1) The optimum core depends on the end-of-cycle (EOC) state. This final state should have the maximum allowed k_{eff} for the fissionable

material in the core at EOC. Any movement of fuel in the EOC core will either reduce the k_{eff} or violate the constraints.

(2) The core reload always begins with the beginning-of-cycle (BOC) and a path must be found to move from the BOC configuration to the optimum EOC configuration without violating constraints.

(3) The number of variables involved in trying to develop an optimum BOC configuration which leads to an optimum EOC core is considerable; however, the EOC core will by its very nature require it to be a low leakage core. In addition, the fuel assemblies with maximum k_∞ will be moved into the highest neutron importance positions without violating constraints. To obtain this path from BOC to EOC burnable poisons must be employed.

(4) Ahn and Levine[39] have shown that the core should be loaded with BP's to maintain a constant (Haling) power distribution through most of the cycle, but the optimum design changes the power distribution near EOC to reduce the leakage. Thus, the optimum power distribution follows close to a Haling power distribution.

(5) The availability of burnable poisons limits the ability to develop BOC configurations which will lead to optimum EOC configurations. As a consequence, several serial optimization calculations must be made before a BOC configuration can be determined using practical burnable poisons i.e., BPs that can be procured from the fuel manufacturer, that will lead to the best possible EOC core configuration.

(6) The inventory of fissionable fuel should be carefully monitored, because the total energy produced during the cycle and the metal-to-water ratio of the fuel assemblies determine the ^{235}U consumed during the cycle. Since the metal-to-water ratio of the fuel assemblies remains essentially unchanged from cycle to cycle, the ^{235}U consumed during the cycle is independent of the fuel management loading schemes and/or BOC core configuration. The ^{235}U required at EOC to maintain the core critical is, however, strongly affected by the loading strategies used for the core.

The above requirements can be used to automatically optimize a PWR. The first step is to use the Priority Loading Scheme (PLS) to load the PWR with selected used fuel and fresh fuel. The PLS is developed using either the OPHAL code [35] or the Space-Covering Approach and Modified-Wolfe (SCAM-W) algorithm[40]. In step 2 the loaded core is depleted using a Haling power distribution or by stepwise depletion to establish the core lifetime. This is achieved by setting the enrichment of the fresh fuel appropriately. A Haling calculation is determined for this core configuration and most likely will show some fuel assemblies having normalized powers or power densities that exceed allowed values. This is because the PLS is developed for ideal fuel assemblies which are different from the actual fuel assemblies. Hence some of the fresh fuel may have normalized powers that are too high. In Step 3, the reload configuration is

changed by (1) splitting the fresh fuel enrichment, (2) shuffling the fuel, or (3) by a combination of the two methods to reduce the NP's to allowed values. It should be mentioned that the maximum allowed Haling power peak must be less than the allowed value because practical BP designs cannot produce a Haling power distribution. Thus, the core NPs will actually exceed the Haling NP's by some small but significant values.

At the end of step 3, the process has established an optimum Haling core design that is BP solvable. A BP solvable core is one in which practical BPs can be inserted to make it a feasible core. The Haling calculation assumes inherently a rather sophisticated BP design, but this BP design does not appear in the calculation. Hence BPs must now be placed into the core in an optimum manner which is achieved in step 4.

There are now two different methods used to accomplish this in step 4 automatically; both methods calculate the required BP distribution using a power shape driven solution to the two group diffusion equation. This method takes the non-BP core at any particular burnup step and determines the optimum BP distribution which will produce the desired power shape.[38] Here the diffusion equation is used similar to that of Chao et al[40] to determine a BP composition at each burnup step to effect close to a radial Haling power distribution. The two group equations are solved with a fast source term that is forced to follow the Haling power distribution. This solution gives a first iteration value to Φ_1 and the thermal leakage $\int_V \nabla \cdot J_2 \, dV$. Rearranging the fast group equation as

$$\Sigma_{a2} = \frac{\frac{1}{V}\int_V \nabla \cdot J_2 \, dV - \Sigma_{c12} \ \phi_1}{\phi_2^*} \tag{16}$$

a first iteration value of Σ_{a2} is obtained where

$$\phi_2^* = \frac{p(r) - K_1 \Sigma_{f1} \ \phi_1}{K_2 \Sigma_{f2}} \tag{17}$$

and

p(r) = power density

K_g = energy conversion constant watt-sec./fission

and g = 1,2 for groups 1 and 2, respectively.

All other constants in the above equations are standard. The new value of Σ_{a2} is used in a repeat of the above calculations and the process continued until the calculations converge.

Thus this method develops at each burnup step a BP distribution that will match as close as possible the desired power shape. This allows developing an

artificial BP depletion curve for each BP position which in almost all cases will not match a practical BP. Once this is achieved, the problem then is to choose a practical set of BPs that minimize the maximum power peak during depletion. During any depletion calculation there is one burnup step in which the peak power is a maximum for the total cycle. Both methods use this burnup step to choose the final BP solution, but they differ how they use the data at this burnup step. Each method is discussed further in this report in the sections dealing with finding the BP solution for the actual cores.

In step 5, the core is depleted in a normal manner using the BP reload distribution developed in Step 4. If the power distribution produces some fuel assemblies with normalized powers that are too high, step 6 allows some readjusting of the core configuration to bring these values into compliance with the constraints. At the end of step 6, the core design is complete and step 7 presents a summary description of the design. The code actually produces two separate designs. In step 3, two core configurations are selected, one by first splitting the fresh fuel enrichments and then shuffling if necessary, and the other configuration is produced by reversing the procedure.

The above methods have been incorporated into a main program, one of which employs an expert system code to automate all of the calculations and decisions made during the optimization process. The other is incorporated into a single main Fortran program. In this section, techniques used to optimize cycle 9 of TMI Unit I and cycle 2 of Beaver Valley Unit II are described and results of the analysis are presented.[38]

III.1 Optimizing the TMI-1 Cycle 9 Core Reload

The above steps have been used to automatically design the TMI-1 Cycle 9 core reload; however, the following design specifications had to be included in this process;

(1) A cycle length of 600 EFPD +0/-25 EFPD;

(2) The number of fresh fuel assemblies $\leq$ 80:

(3) A maximum allowed enrichment of 4.4 wt% ^{235}U for the fresh fuel assemblies;

(4) The fresh fuel with burnable poisons (BPs), cannot be placed in positions where there are control rods;

(5) The maximum allowed BP is 2.8% boron in the matrix;

(6) For Cycle 9, twenty fresh fuel assemblies should be used that have enrichments of 3.63%, four of which should have BP's of 1.7 wt% B_4C;

(7) The Maximum RPD or NP cannot exceed 1.375;

(8) The assembly averaged RPD or NP at the boundary assemblies must be below 1.10 because the peak/average ratios in those assemblies are high;

(9) Minimize the neutron fluences at the pressure vessel welds, i.e., at 0 and 11 degrees of the core horizontal axis.

(10) Minimize the number of different fresh fuel enrichments.

The above design specifications establish many constraints on the manner in which the TMI-1 core can be loaded to effect a 600 EFPD. Nevertheless, there remains significant flexibility in the design so that optimization techniques can be used to determine the final design.

The first step in the optimization procedure is to use the Priority Loading Scheme to load the core with available used fuel assemblies and fresh fuel. Either the OPHAL code or the SCAM-W algorithm results can be used here. Table 1 shows the OPHAL BOC k_∞ and the k_∞ of the fuel assemblies available for reload in the core.

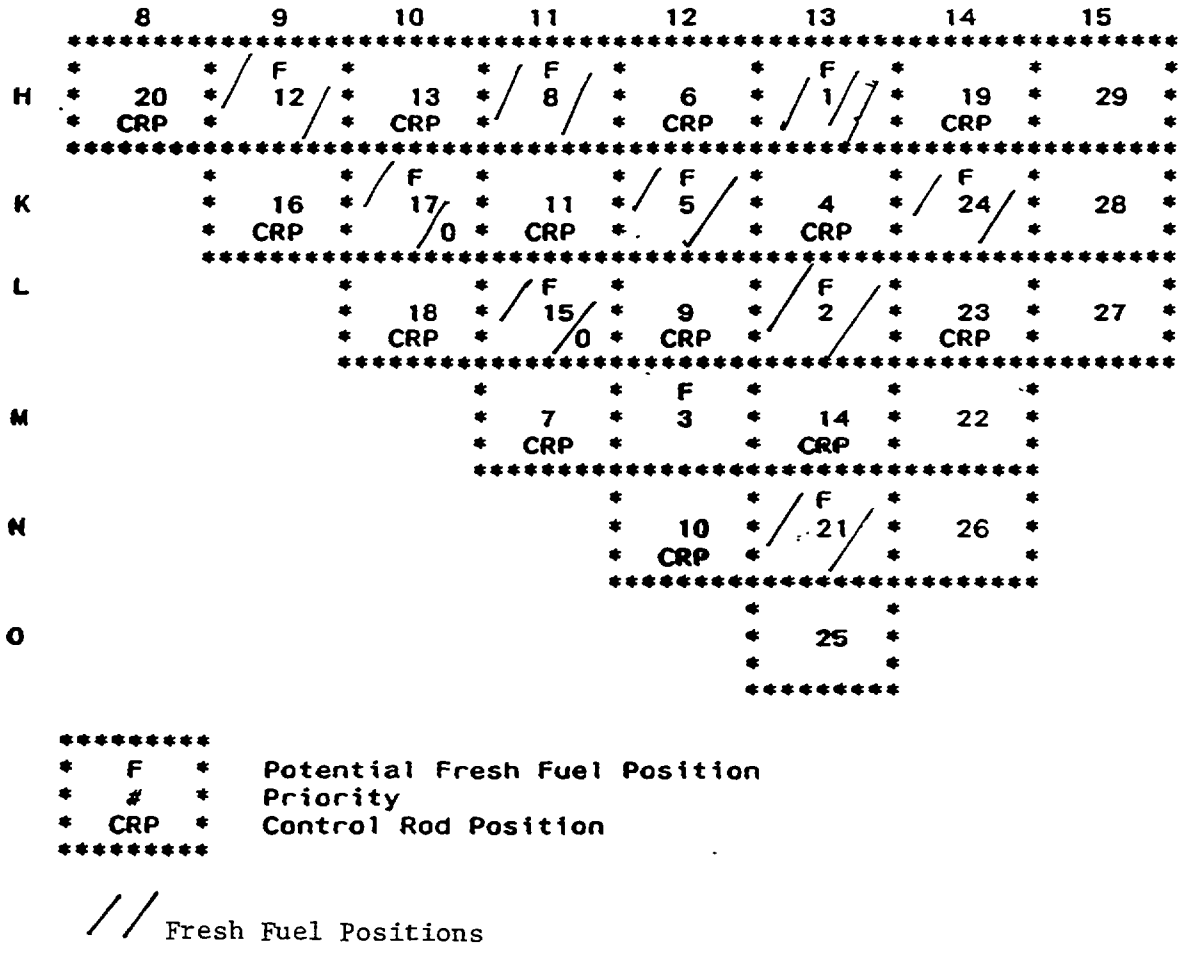

FIG. 14. TMI-1 core loading information (priority, control rods, fresh fuel positions).

Table 1
Priority Scheme

Priority	Core Position	OPHAL k_∞	Actual k_∞
1	6	1.378	1.398
2	19	1.374	1.398
3	23	1.374	1.398
4	13	1.359	1.172
5	12	1.355	1.398
6	5	1.343	1.158
7	22	1.342	1.158
8	4	1.342	1.398
9	18	1.341	1.149
10	26	1.331	1.125
11	11	1.329	1.127
12	2	1.328	1.398
13	3	1.328	1.125
14	24	1.327	1.124
15	17	1.321	1.398
16	9	1.317	1.125
17	10	1.316	1.398
18	16	1.312	1.123
19	7	1.299	1.123
20	1	1.275	1.098
21	27	1.256	1.398
22	25	1.256	1.398
23	20	1.256	1.123
24	14	1.256	1.100
25	29	1.242	1.107
26	28	1.003	1.398
27	21	1.003	1.058
28	15	1.003	1.038
29	8	1.003	1.013

The next step is to use the Haling power distribution to establish the core lifetime. In doing this, it is important to obtain a core lifetime slightly greater than 600 EFPD. This is because the Haling power calculation slightly over estimates the actual lifetime of the TMI-1 core.

For a TMI-1 core lifetime of 600 EFPD, analysis has shown that a minimum of 76 fresh fuel assemblies are required to meet core lifetimes whereas, 80 fresh fuel assemblies is the maximum allowed in the core to meet design constraints. Figure 14 shows core positions where control rods are located and where the 80 fuel assemblies can be placed. As a consequence, first estimates of the enrichment can be made in a matter of seconds of computer time by adjusting the enrichments in the fresh fuel. For Table 1 an enrichment of 4.3% is used for which k_∞ (BOC) = 1.398. The related core parameters for the core when loaded with available fuel assemblies are shown in Fig. 15. The k_∞ (ACT) refers to the core with the critical soluble boron.

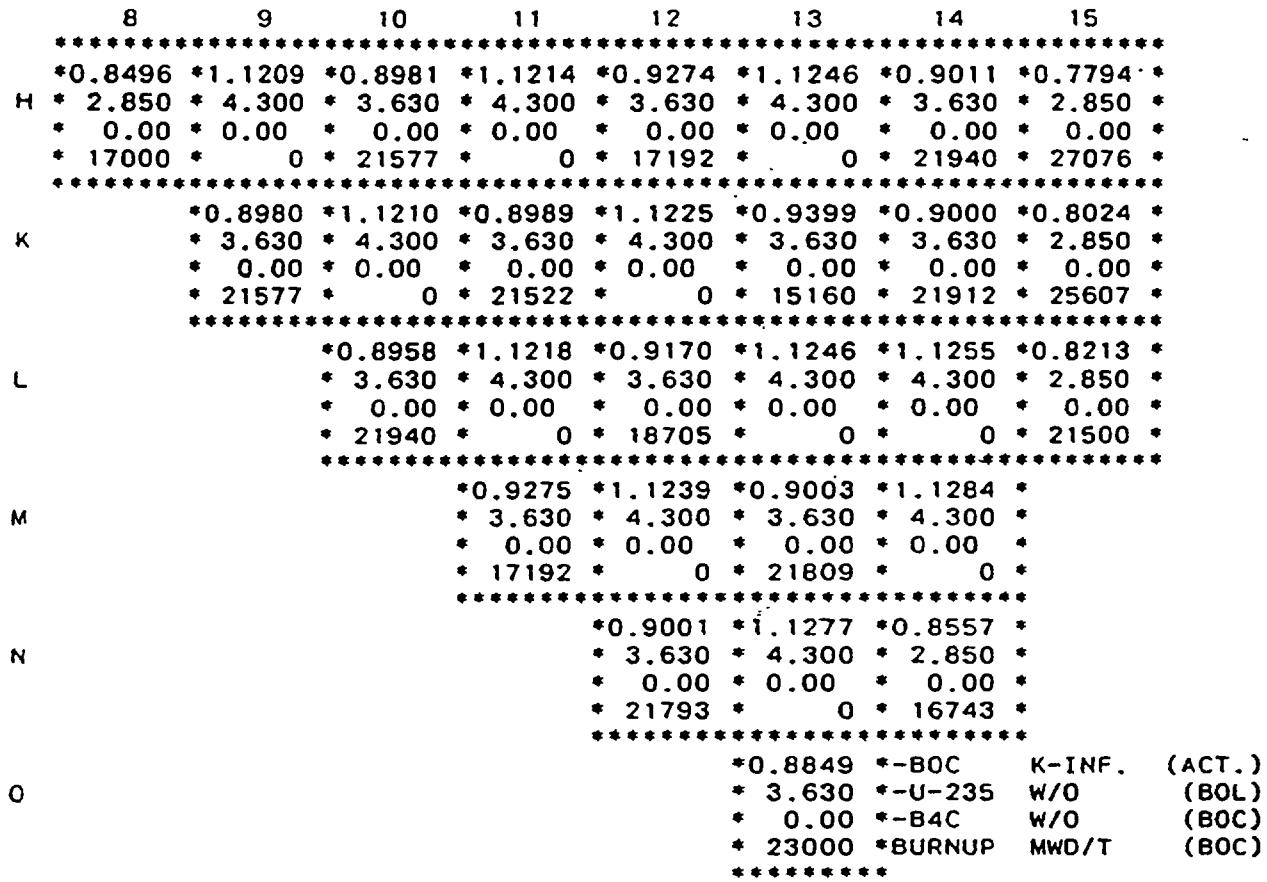

FIG. 15. TMI-1 loading of core with available fuel assemblies.

The required new fresh fuel enrichment is quickly determined by depleting the core with a Haling power distribution for a few increased enrichments. The new enrichment then produces a Haling power distribution which is significantly different from the optimum OPHAL Haling results as shown in Fig. 16. The Haling power distribution establishes the flattest possible power distribution for a particular core configuration. Once real BP's are placed in the core, the maximum NP's will normally be exceeded in the actual case as shown in Fig. 16. Since a maximum NP of 1.37 has been set by fuel management guidelines to prevent pin power peaks from being exceeded, the maximum NP allowed for a Haling power distribution must be below this by approximately 5%. Hence, an NP of 1.32 is the maximum allowed in a Haling calculation and this value has been exceeded by certain positions when available fuel assemblies are placed into the core as shown in Fig. 16. Fuel assemblies in core positions 2, 4, 10, 12, and 17 gave NP's greater than 1.32 and hence, must have their enrichment reduced because it is not possible to shuffle the fresh fuel with used fuel in the core. This requires splitting the fuel enrichment of the fresh fuel reload causing the other fresh fuel positions to have their enrichment increased to meet the core lifetime requirements of 600 EFPD.

For this core, 20 fresh fuel assemblies with enrichments of 3.63% must be placed into the core because they had been previously manufactured. Using this constraint and others as previously given, a BP solvable core is generated which meets the NP max = 1.32. This is shown in Fig. 17. Also in Fig. 17 is the Haling power distribution produced before the 20 fresh fuel 3.63% enriched fuel assemblies are inserted.

The next step is to allocate the BP's. For this core, the peak NP's occur near the middle of the cycle where the BP's are well depleted. As stated there are two methods developed for performing this task automatically to minimize the power peak during depletion. For the TMI-1, an iterative technique is used in

```
        8        9       10       11       12       13       14       15

    ********************************************************************
    *   1    *   2    *   3    *   4    *   5    *   6    *   7    *   8    *
    * 1.275 * 1.328 * 1.328 * 1.342 * 1.343 * 1.378 * 1.299 * 1.003 *
H   * 1.098 * 1.398 * 1.125 * 1.398 * 1.158 * 1.398 * 1.123 * 1.098 *
    * 1.156 * 1.250 * 1.272 * 1.330 * 1.353 * 1.350 * 0.902 * 0.318 *
    * 1.064 * 1.353 * 1.142 * 1.363 * 1.162 * 1.230 * 0.768 * 0.345 *
    ********************************************************************
            *   9    *   10   *   11   *   12   *   13   *   14   *   15   *
            * 1.317 * 1.316 * 1.329 * 1.355 * 1.359 * 1.256 * 1.003 *
K           * 1.125 * 1.398 * 1.127 * 1.398 * 1.172 * 1.100 * 1.035 *
            * 1.235 * 1.245 * 1.300 * 1.369 * 1.290 * 0.832 * 0.301 *
            * 1.138 * 1.360 * 1.144 * 1.345 * 1.081 * 0.798 * 0.369 *
            ********************************************************************
                    *   16   *   17   *   18   *   19   *   20   *   21   *
L                   * 1.312 * 1.321 * 1.341 * 1.374 * 1.341 * 1.003 *
                    * 1.124 * 1.398 * 1.149 * 1.398 * 1.124 * 1.059 *
                    * 1.239 * 1.277 * 1.320 * 1.271 * 0.768 * 0.244 *
                    * 1.141 * 1.361 * 1.139 * 1.254 * 1.026 * 0.366 *
                    ********************************************************
                            *   22   *   23   *   24   *   25   *
                            * 1.342 * 1.374 * 1.327 * 1.256 *
M                           * 1.162 * 1.398 * 1.125 * 1.398 *
                            * 1.324 * 1.344 * 1.067 * 0.620 *
                            * 1.164 * 1.293 * 0.976 * 0.875 *
                            ************************************
                                    *   26     27       28
                                    * 1.331 * 1.256 * 1.003 *
N                                   * 1.125 * 1.398 * 1.398 *
                                    * 1.123 * 0.781 * 0.298 *
                                      0.995   0.983   0.458
                                    ************************
                                            *   29   *
                                            * 1.242 *
O                                           * 1.109 *
                                            * 0.455 *
                                            * 0.527 *
                                            *********

    *********
    *   #   *   Core Position
    * x.xxx *   OPHAL k-infinity
    * y.yyy *   Actual k-infinity
    * u.uuu *   Haling NP from OPHAL
    * v.vvv *   Actual Haling NP
    *********
```

FIG. 16. Comparison of OPHAL and actual results.

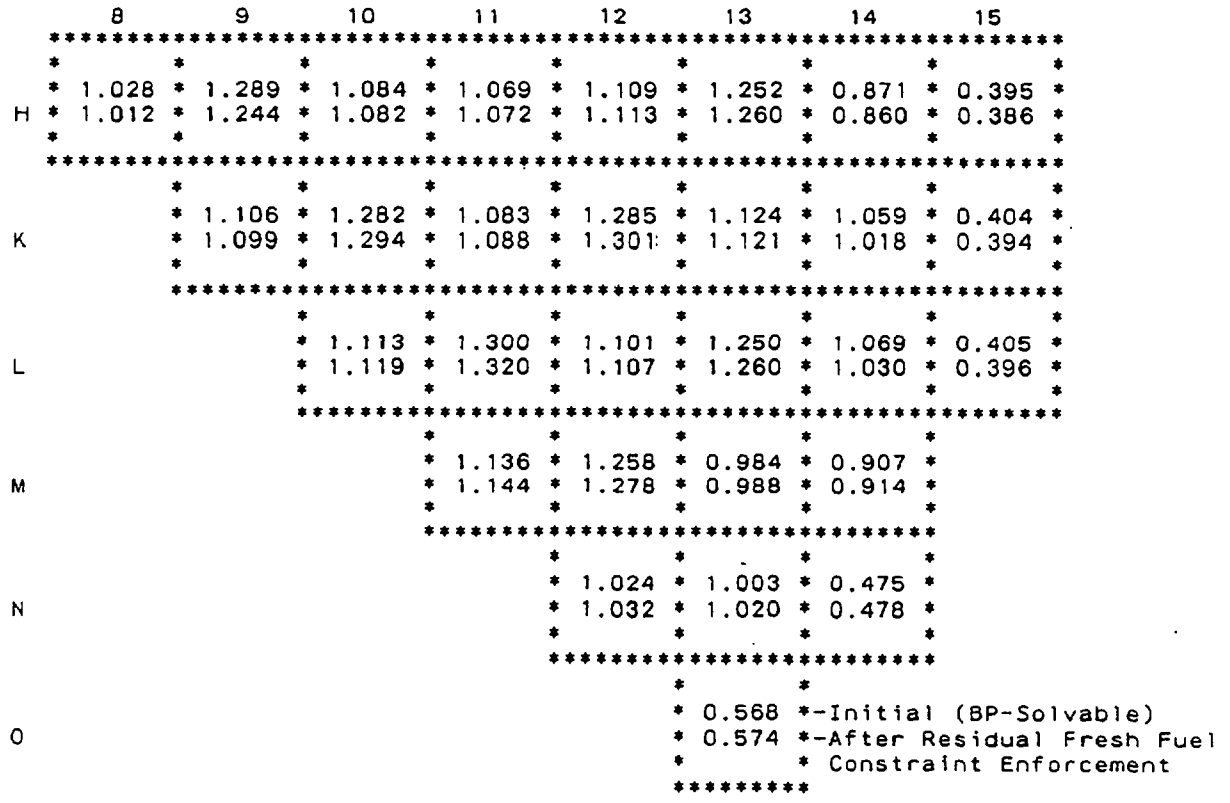

FIG. 17. Cycle 9 haling power distribution in the TMI-1 cycle optimization process.

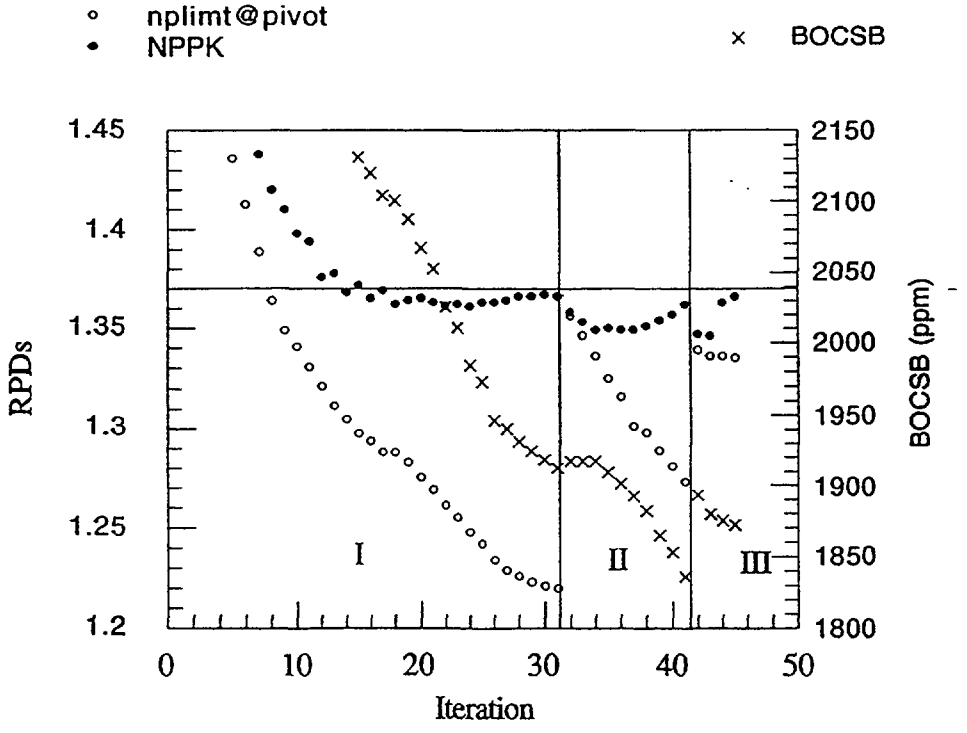

FIG. 18. TMI-1 cycle 9 BP iteration process.

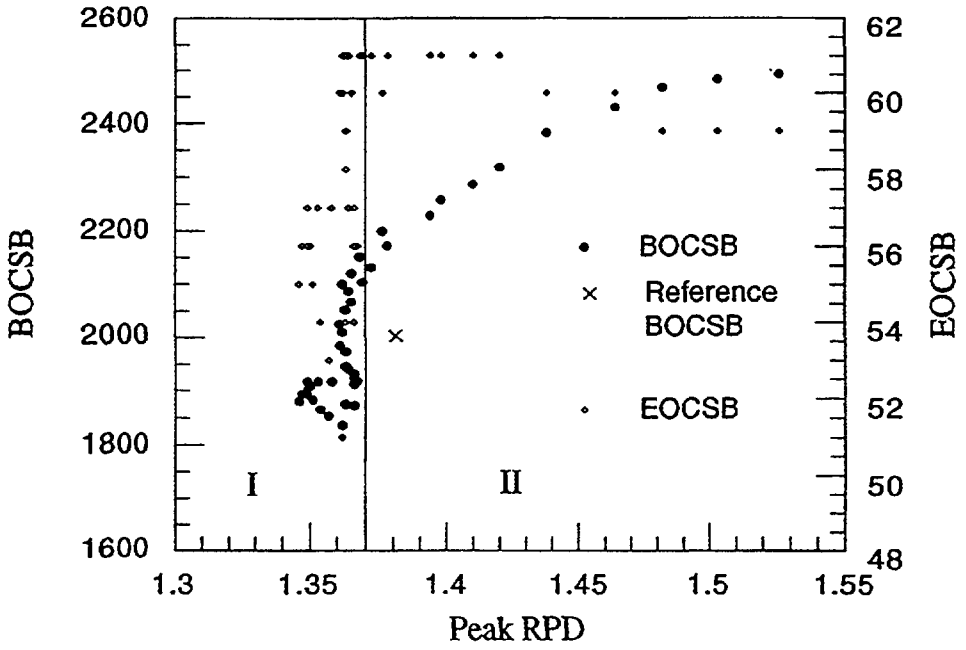

FIG. 19. TMI-1 cycle 9 BP placement optimization results.

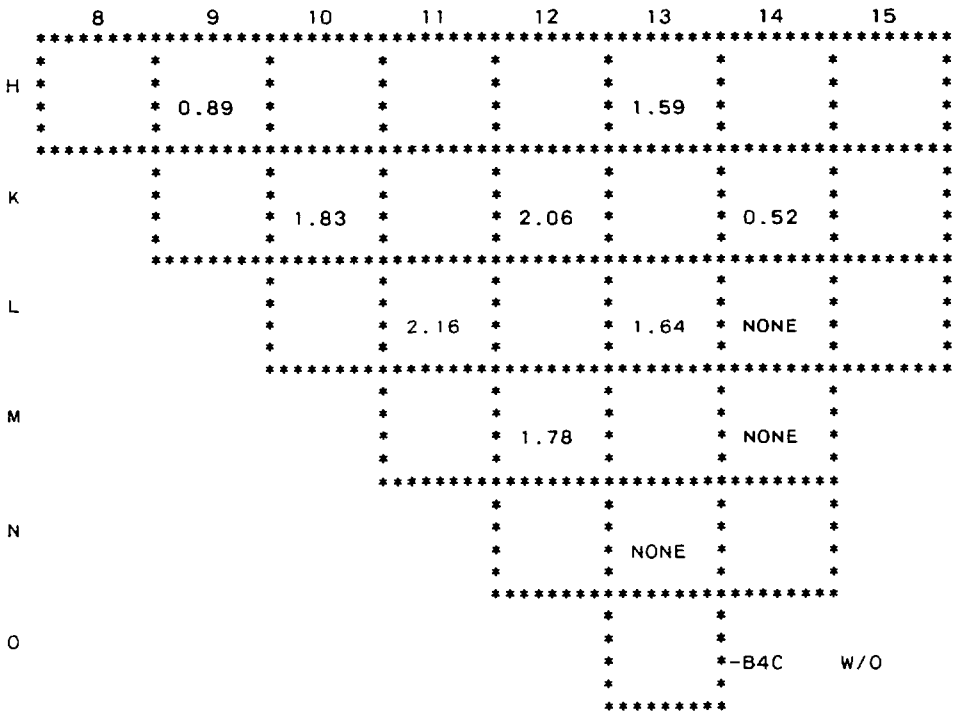

FIG. 20. TMI-1 cycle 9 optimal BP loading pattern (OPT2).

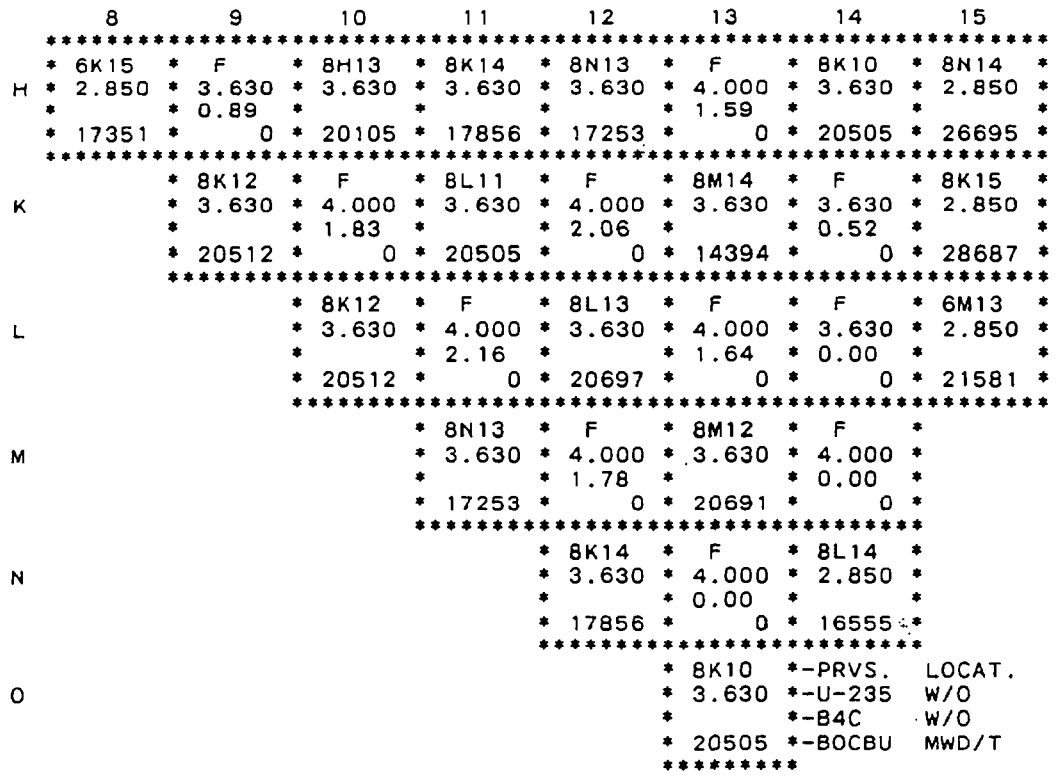

FIG. 21. TMI-1 cycle 9 optimal loading pattern (OPT2).

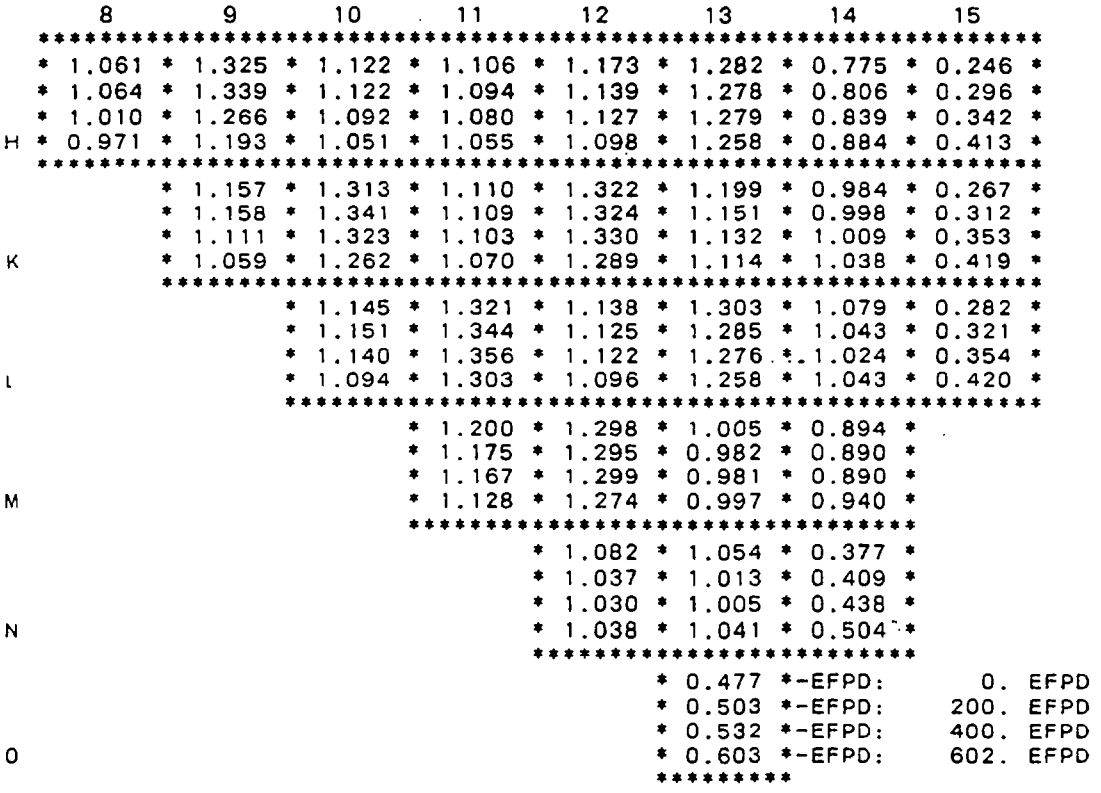

FIG. 22. TMI-1cycle 9 optimal pattern (OPT2) power distribution.

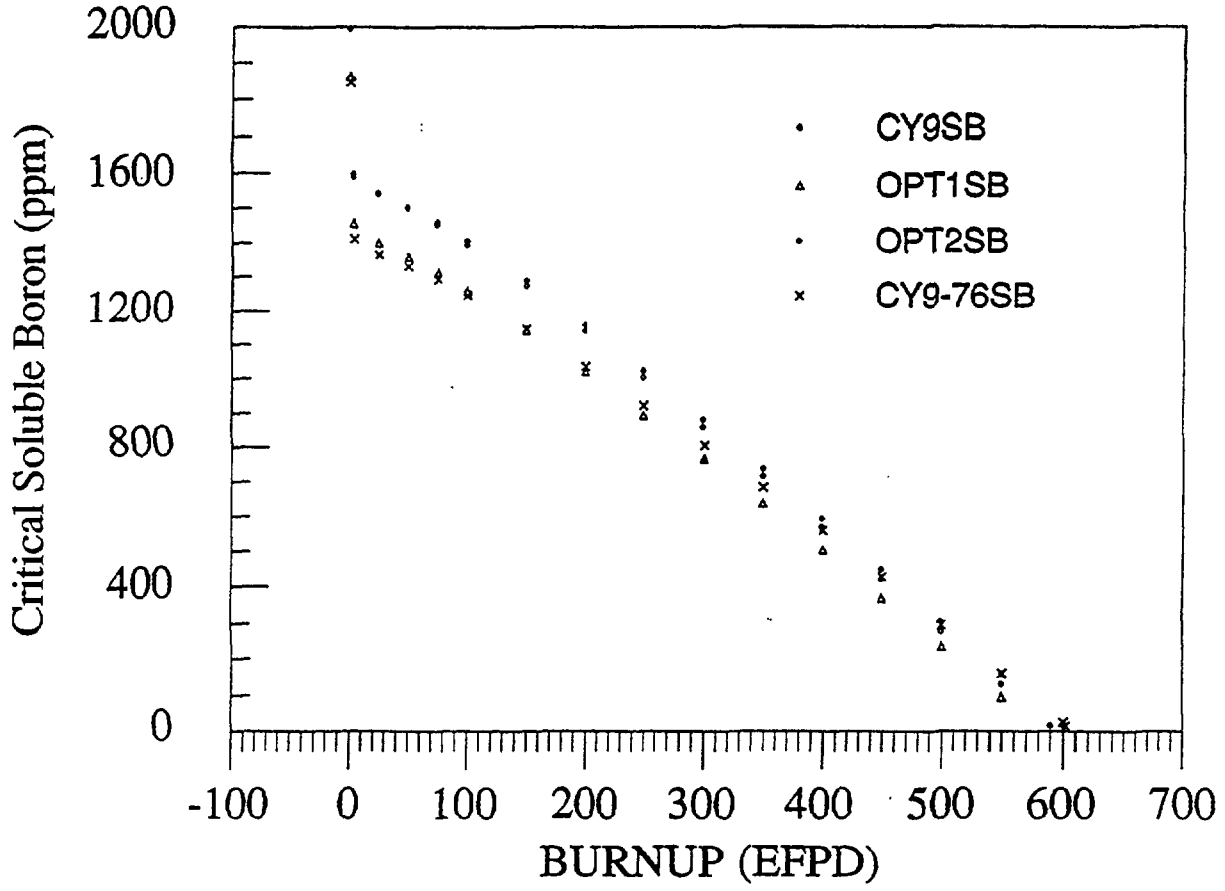

FIG. 23. TMI-1 cycle 9 critical soluble boron letdown curves of different loading patterns.

which the analysis reduces the maximum power peak (NP or RPD) successively until convergence is reached. The basic method of using the power shape driven solution to the diffusion equation to determine the correct BP distribution at a burnup step is performed during each iteration.

The first step determines the maximum NP at BOC with no BP's in the core, which is very high. A power shape is chosen that reduces the maximum power peak by a few percent. The BP distribution which produces this power shape is then found and used to deplete the core to several EPFD. The new NP_{max} is reduced again and the BP found to produce this power shape. This process continues as shown in region I of Fig. 18 until the process converges.

At this point the core is depleted beyond the point where the NP_{max} for the complete cycle is determined. The new burnup step is chosen as a pivotal position for the power shape changed at this burnup step to reduce the NP_{max} as shown in II Fig. 18. It can be observed for this case that all of the BP designs have NP_{max} that are below the maximum allowed and, therefore, are acceptable. However, the program chooses a third pivotal burnup step, the one which produces a maximum power peak for the new BP distribution. The program then selects automatically the BP distribution that gives a low NP_{max} and BOC soluble boron concentration. Fig. 19 shows better how this is accomplished. The point

at NP_{max} = 1.345 and BOCSB = 1875 ppm soluble boron is the first optimum point. Other points near this may also be chosen for solution. Fig. 20 shows the BP distribution producing the optimum solution and Fig. 21 shows the final loading. Fig. 22 shows the NP distribution at a few selected burnup steps during depletion and at EOC. Finally, Fig. 23 shows the boron let down curve for the optimum core, OPT2 SB, plus three other configurations shown for comparison.

III.2 Optimizing the BV-1 Cycle 10 Core Reload[38]

The process used to automatically optimize the Beaver Valley Unit 1 Cycle 10 reload follows a similar procedure to that performed for the TMI-1 Cycle 9. The constraints for the Beaver Valley core are listed below and they are different from the TMI-1 cycle 9 constraints.

(1)	Cycle Length	420 + 20 EFPD;
(2)	Maximum Fresh Fuel Enrichment	4.0%;
(3)	Maximum Peaking NP	1.395;
(4)	Maximum Number of IFBA	9000;

Table 2 - Ranking Tables Of The Loading Positions

Rank	$k_\infty(EOC)$	Position	Number Of Fuel Assemblies
1	1.1902	23	8
2	1.1792	14	8
3	1.1685	22	8
4	1.1590	19	8
5	1.1466	7	4
6	1.1412	16	4
7	1.1346	11	8
8	1.1329	2	4
9	1.0957	5	4
10	1.0542	3	4
11	1.0517	13	8
12	1.0428	6	4
13	1.0296	10	8
14	1.0027	20	8
15	0.9920	25	4
16	0.9915	12	8
17	0.9367	15	8
18	0.9331	8	4
19	0.9318	17	8
20	0.9288	26	8
21	0.9187	24	8
22	0.9072	18	8
23	0.9062	1	1
24	0.8993	21	4
25	0.8981	9	4
26	0.8884	4	4

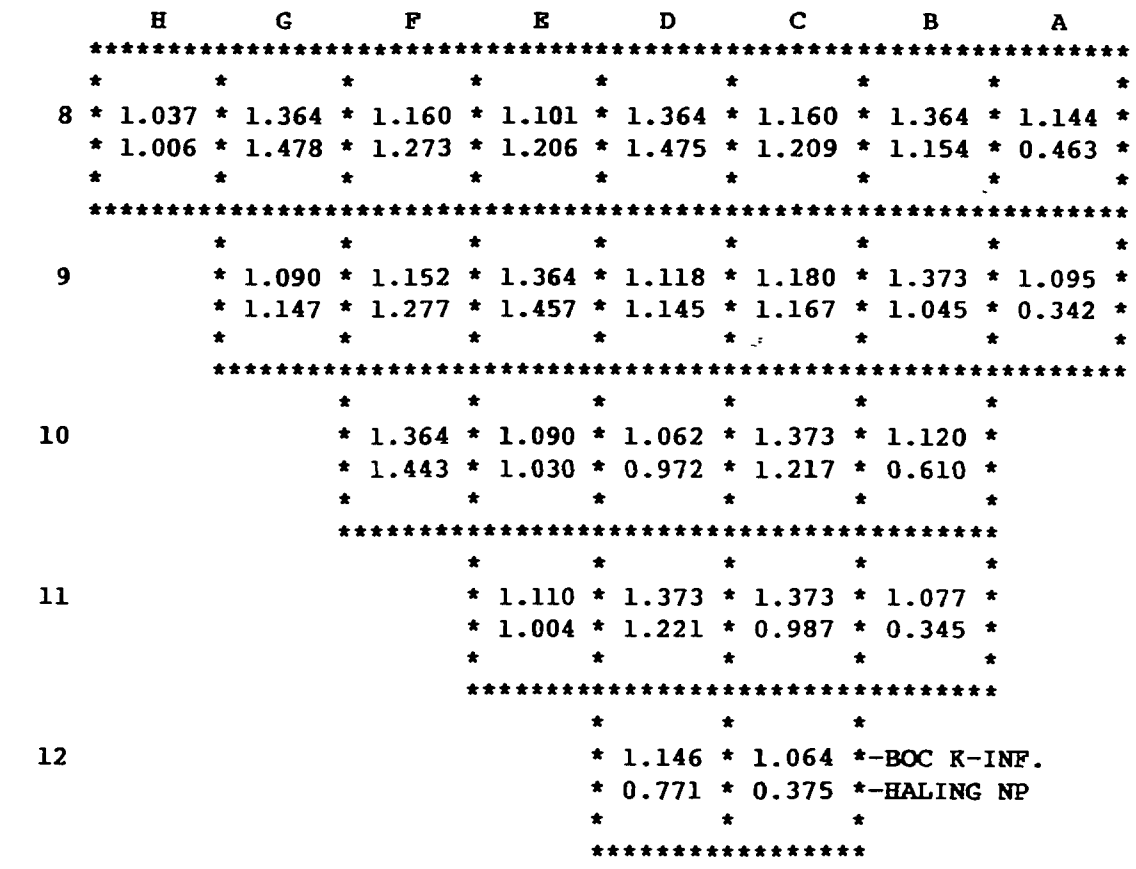

FIG. 24. Beaver Valley Unit 1, cycle 10 – BOC k_∞ and haling power distribution of the first trial loading.

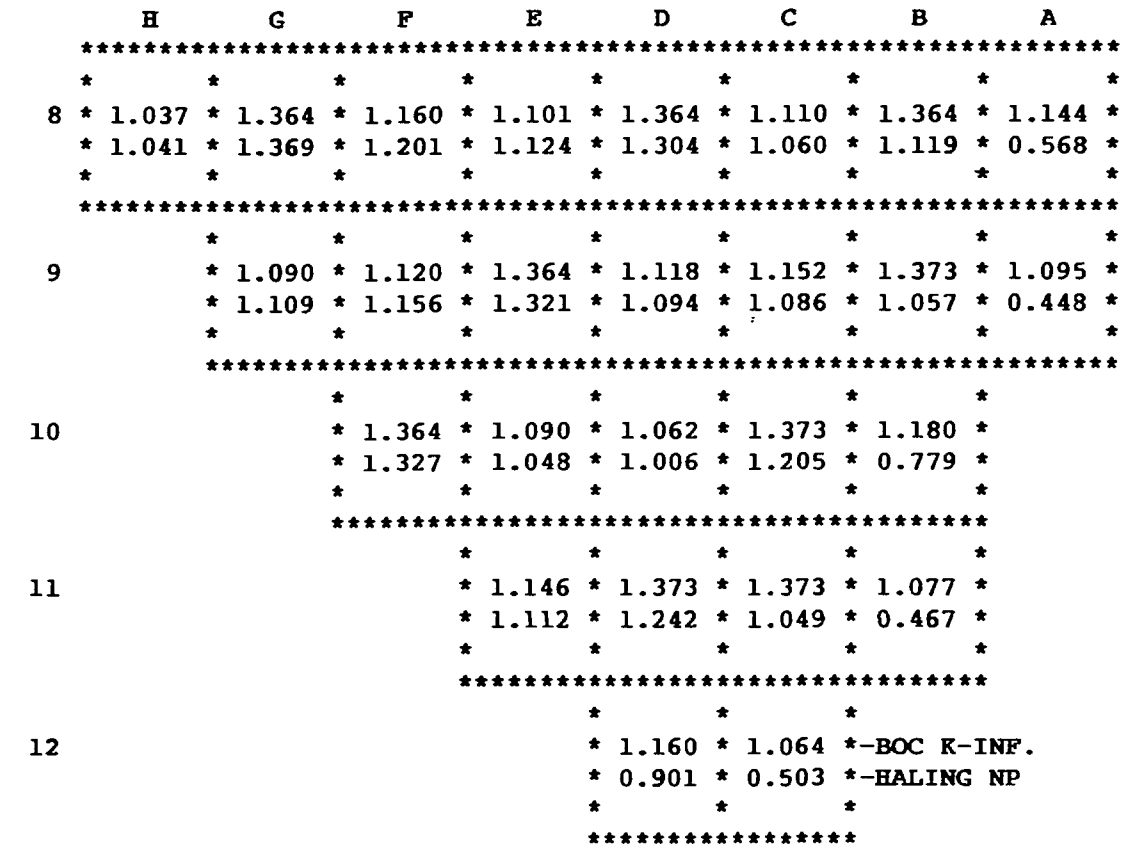

FIG. 25. Beaver Valley Unit 1, cycle 10 – BOC k_∞ and haling NP distribution of the final loading with BPs.

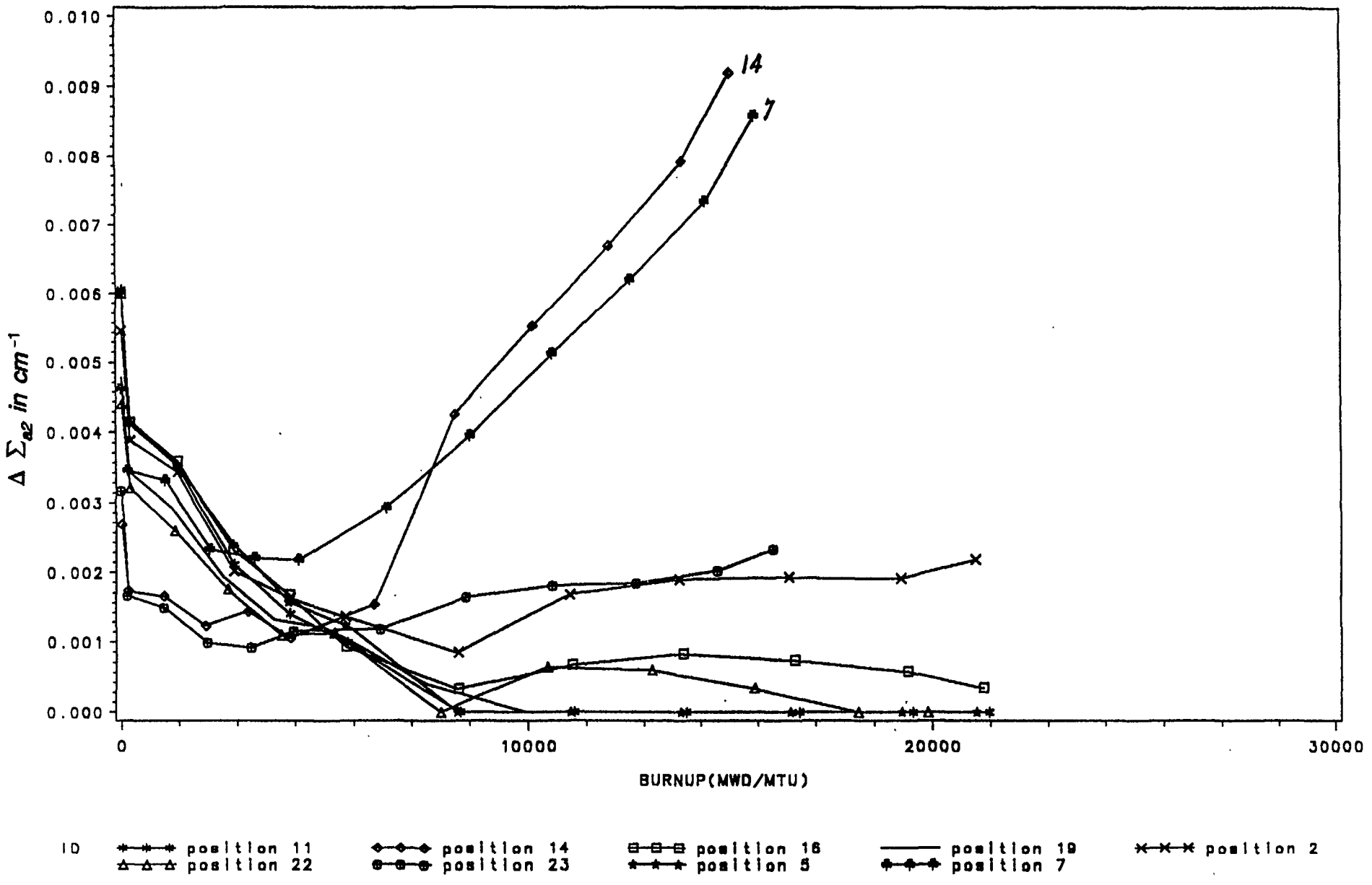

FIG. 26. Beaver Valley Unit 1, cycle 10 – depletion profile to attain haling power distribution.

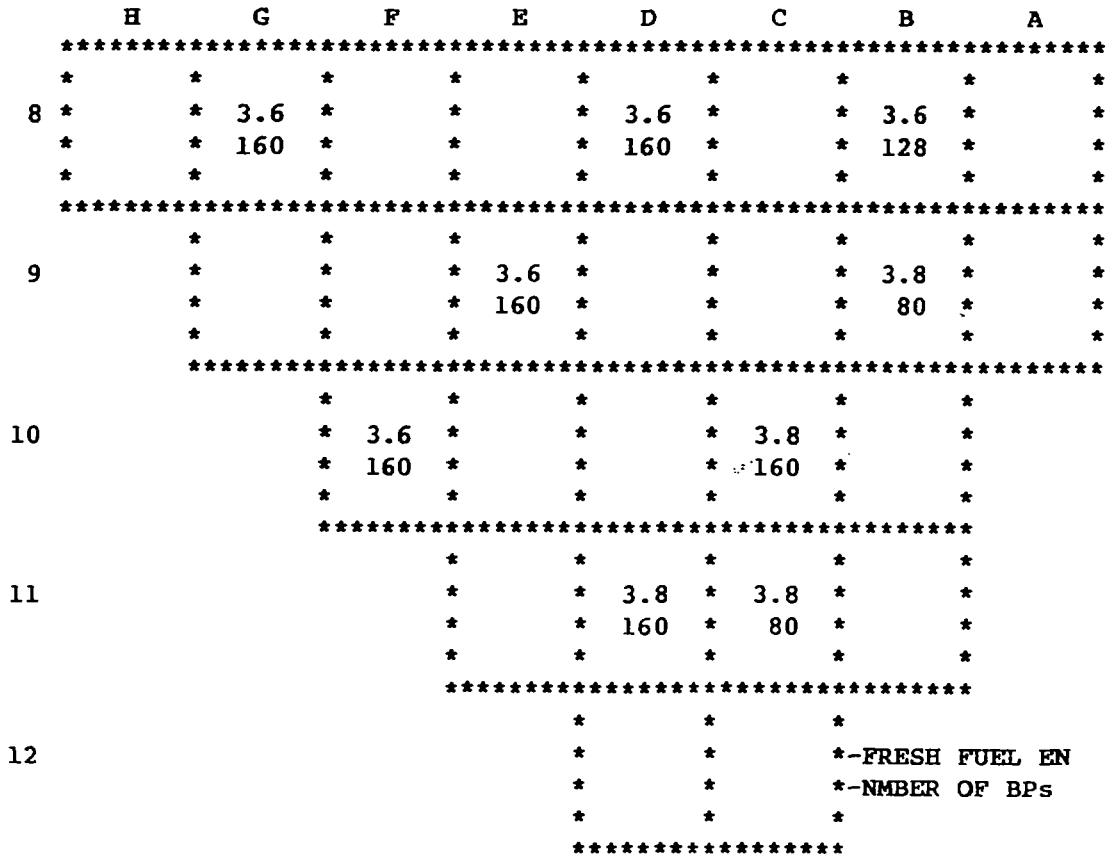

FIG. 27. Beaver Valley Unit 1, cycle 10 – fresh fuel enrichment and IFBA burnable poison loading.

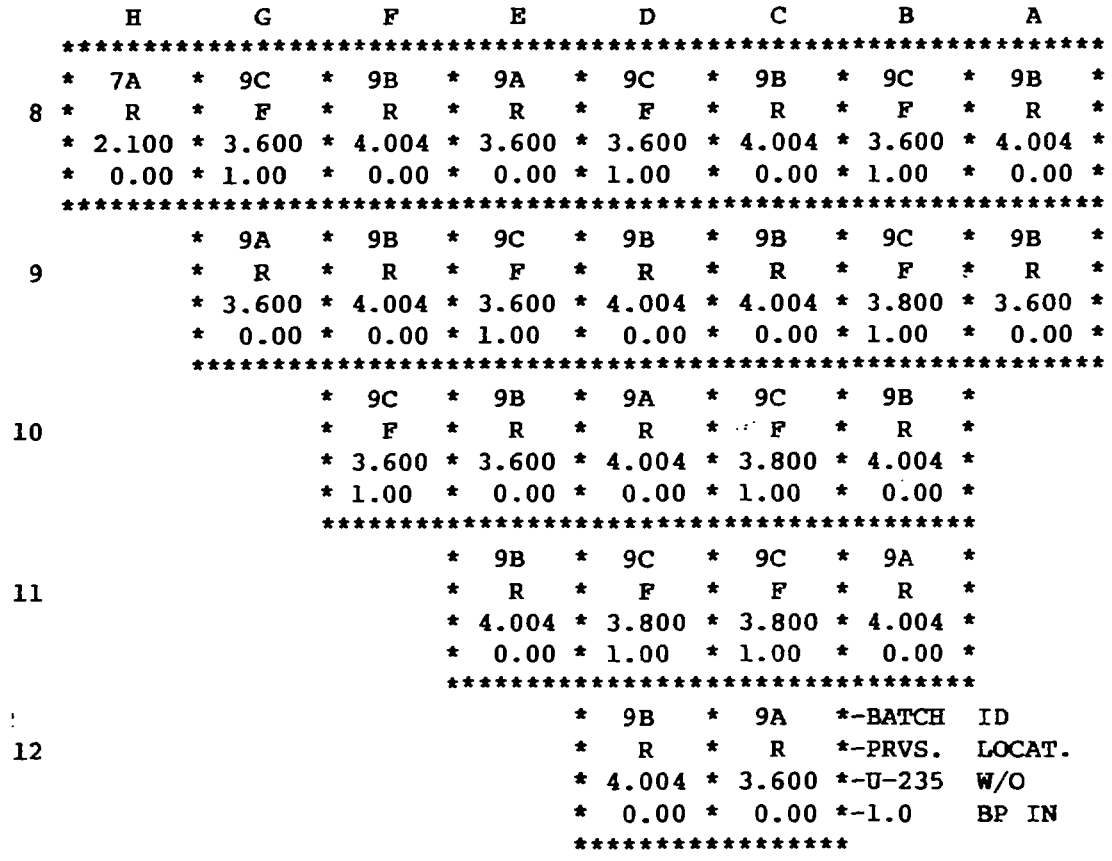

FIG. 28. Beaver Valley.

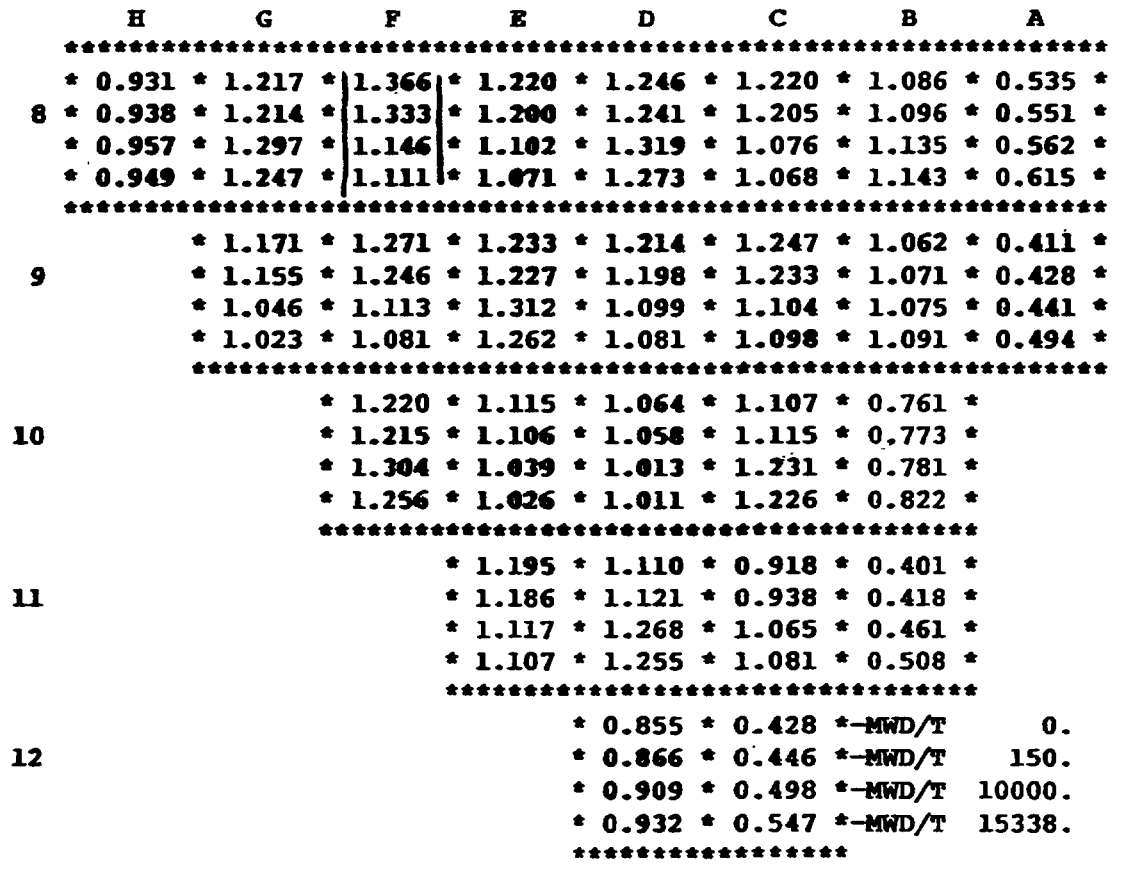

FIG. 29. Beaver Valley Unit 1, cycle 10 – assembly power distribution at typical burnup steps.

(5) Maximum Number of Fresh Fuel Enrichment Splits 3;

(6) Minimum Number of Fuel Assemblies per Split 16;

(7) All Once Burned Assemblies Must Be Reused.

The other differences are as follows:

(1) The Priority Scheme shown in Table 2 is developed from the SCAM-W algorithm.

(2) The BP optimization scheme is different in that the Haling power distribution is always the power shape used to develop the BP distribution, since it automatically minimizes the NP_{max} for the cycle. Every fresh fuel assembly can have IFBA BPs; hence, every fresh fuel assembly is chosen for potential BPs.

The first step is to load the core according to the priority scheme and establish the fresh fuel enrichment and hot spots in the Haling power distribution. Fig. 24 shows the resulting k_∞ distribution and its Haling power distribution. Note that, as expected, this Haling power distribution has very

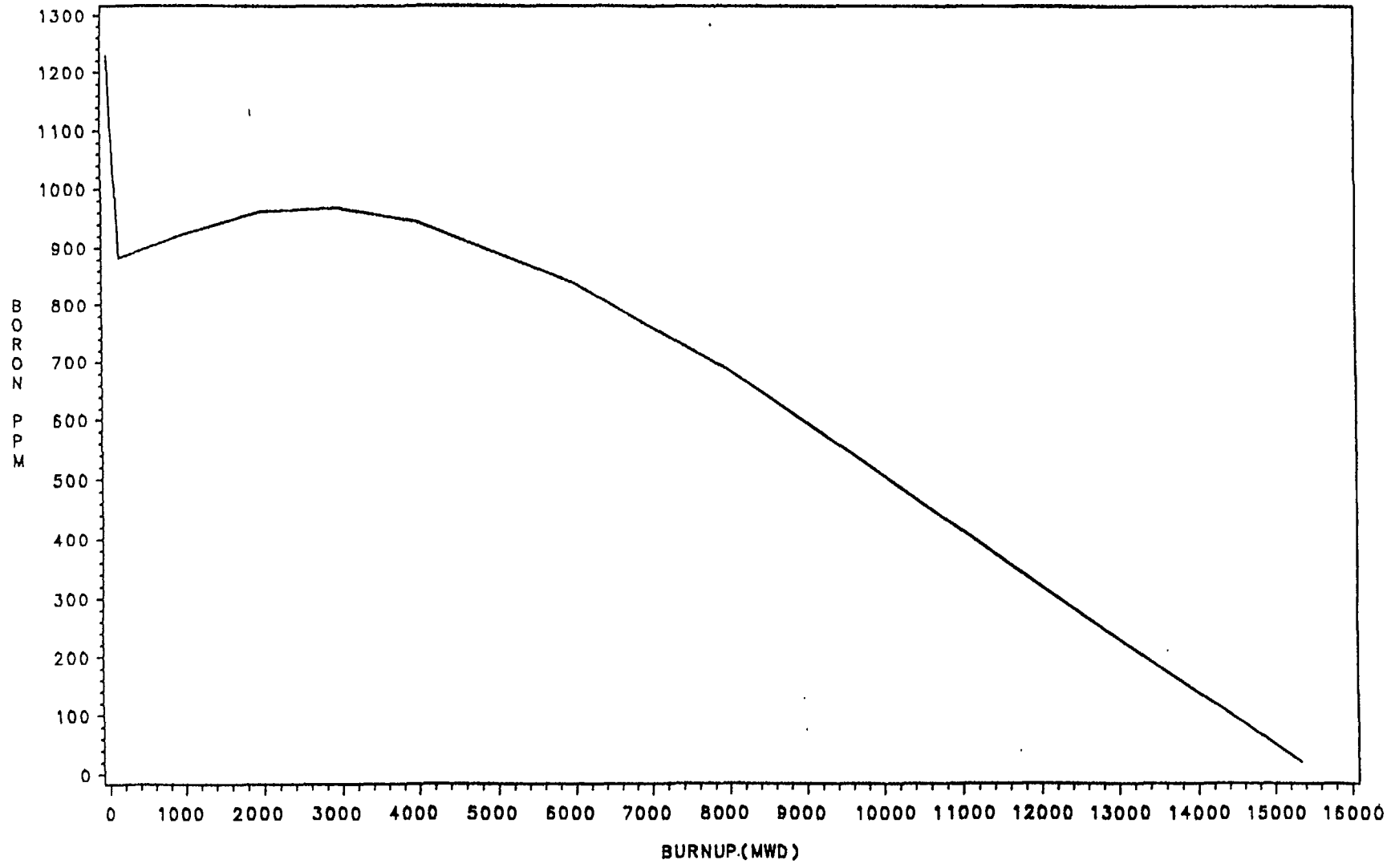

FIG. 30. Beaver Valley Unit 1, cycle 10 – boron letdown curve for optimal core loading design.

high NPs at positions G-8, D-8, H-9, and E-10. The BP solvable core is shown in Fig.-25 where the NPs have been greatly reduced at the hot spots. The method used to find the BP distribution from this core is different. This first step here is to determine the BP depletion trajectories which would create a true Haling power distribution during depletion. The answer to this is shown in Fig. 26 as determined by the power shape driven solution to the diffusion equation which produces the Σ_{a2}'s.

It can be observed that two positions, 7 and 14, produce trajectories that are impossible to achieve with practical BPs. Hence, the maximum allowed BPs are placed here that will deplete at the end of the cycle. The code then checks to determine at each burnup step the $\Delta\Sigma_{a2}$ required to produce the Haling power distribution at that step. Thus, the fresh fuel are assigned IFBA's that produce the Haling power distribution at each burnup step. At this point, the IFBA's that match the required Σ_{a2} at the burnup step having the maximum NP are chosen for the core. The code then tests to determine if this solution is acceptable by depleting the core and checking at each burnup step if the allowed NP_{max} is violated. If so, the IFBA content is increased by a preset amount and the solution checked again. Fig. 27 shows the final BP and fresh fuel enrichment solution and Fig. 28 shows the final reload design. By doing this, an acceptable optimum solution was obtained. Fig. 29 shows the power distribution as a function of burnup for a few selected burnup steps and at EOC, and Fig. 30 shows the soluble boron let down curve for this burnup core.

It should be noted that the NP_{max} 1.372 which occurred during depletion did not occur at any of the burnup steps in Fig. 29.

IV. Expert System Codes

Both the TMI-1 and the Beaver Valley optimization schemes have been integrated into a single main computer code to perform the complete reload analysis using simple input data. The only input required for these codes are:

(1) Core lifetime
(2) Number of fresh fuel assemblies and their enrichment
(3) Available used fuel assemblies
(4) Constraint Values
(5) BP design: WABA, PYREX, or IFBA (Beaver Valley PWRs only)

Other input data have default values which may be changed by the user and such data are plant specific. These main codes then perform all of the tasks necessary to produce optimum reloads and prints out the design together with a summary of the important characteristics of the core during a cycle depletion, e.g. soluble boron let down curve, NPs as a function of burnup. Thus, these codes can now be used by inexperienced engineers to produce superior or optimum reloads. Presently, there are separate main codes for the TMI-1 reactor and the BV reactors. The TMI-1 main code is incorporated in a complete FORTRAN language system whereas the Beaver Valley main code is more of the expert system code type. The Beaver Valley main code is incorporated in a C-language environment

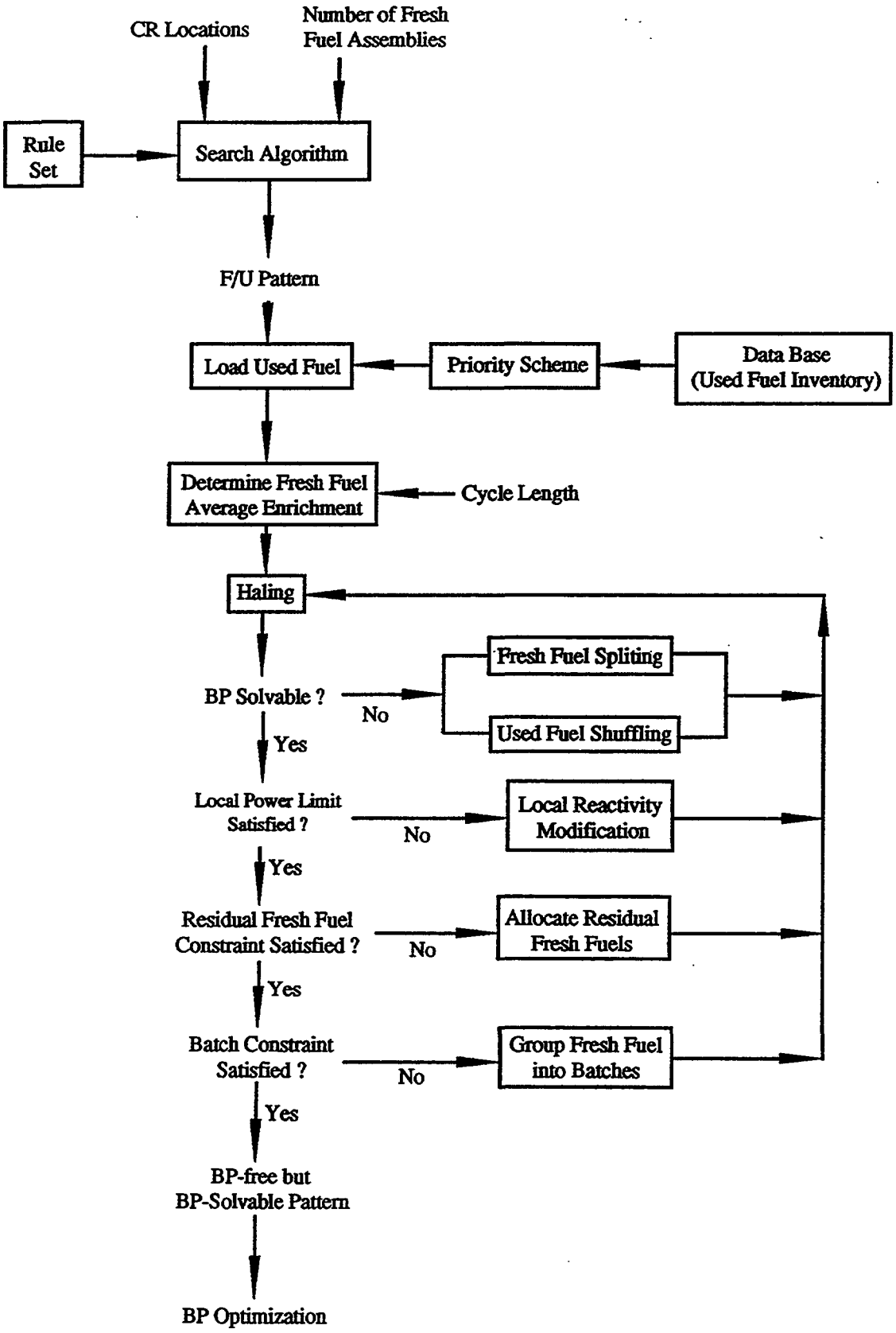

FIG. 31. Main computer program for reloading TMI-1 in FORTRAN language.

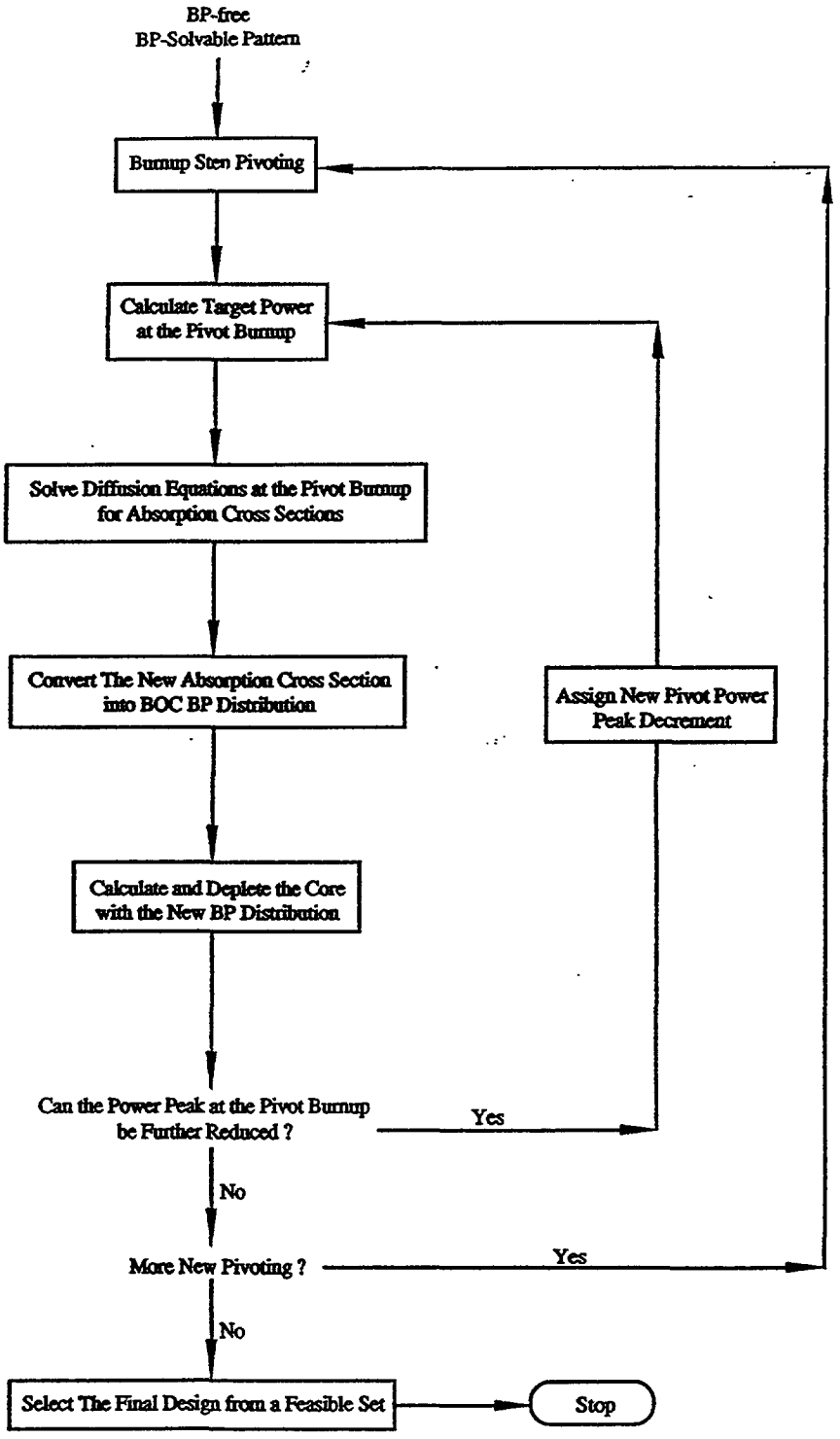
BP-free
BP-Solvable Pattern
Burnup Step Pivoting
Calculate Target Power at the Pivot Burnup
Solve Diffusion Equations at the Pivot Burnup for Absorption Cross Sections
Convert The New Absorption Cross Section into BOC BP Distribution
Assign New Pivot Power Peak Decrement
Calculate and Deplete the Core with the New BP Distribution
Can the Power Peak at the Pivot Burnup be Further Reduced ?
Yes
No
More New Pivoting ?
Yes
No
Select The Final Design from a Feasible Set
Stop

FIG. 31. (cont.)

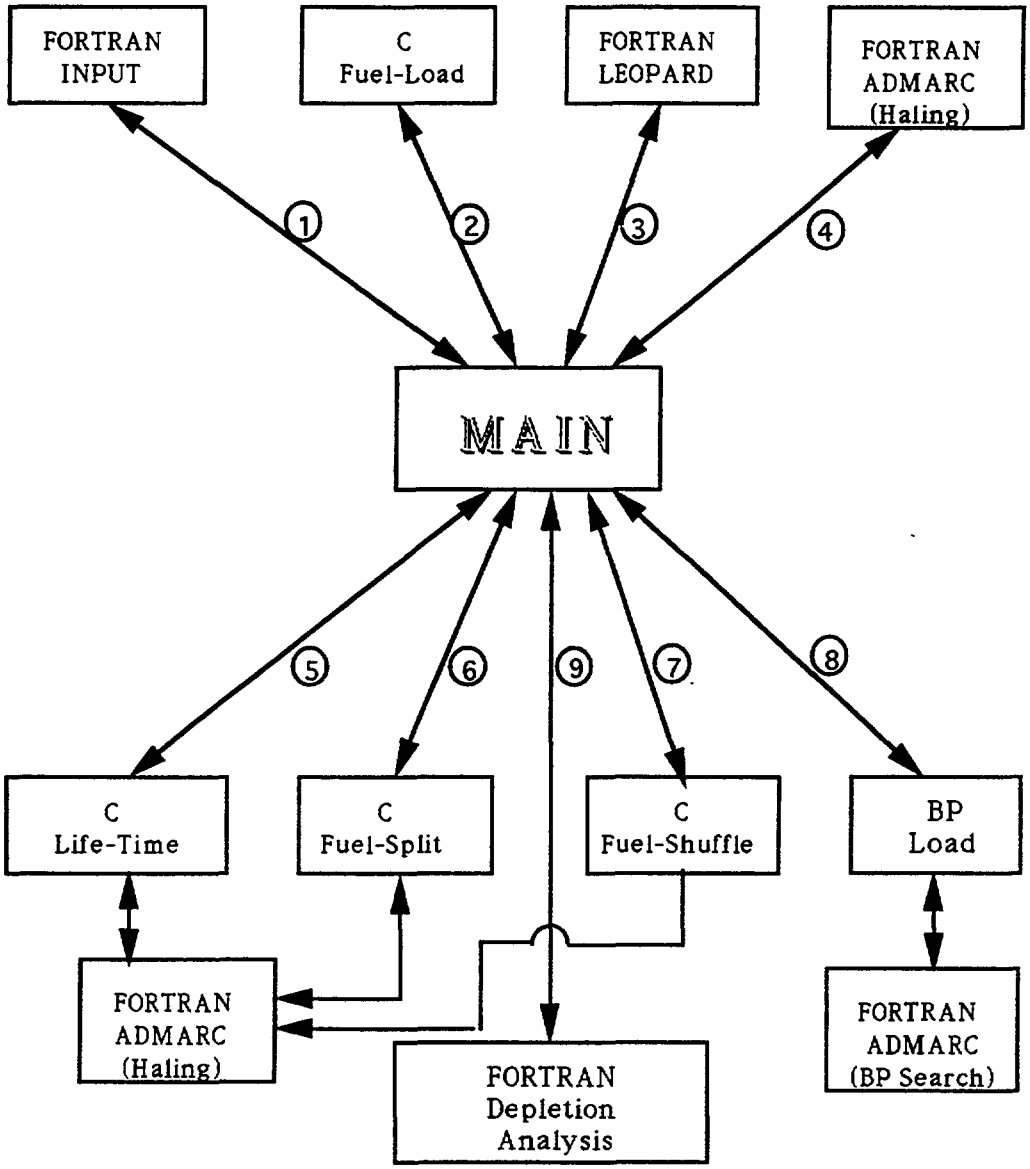

FIG. 32. Block diagram of the automatic optimal PWR reload design computer code sytem.

wherein the if-then rules can be handled very conveniently. In any case, both codes use the same overall procedure as described in this report. Both codes use the same logic to alter the initial core configuration loaded according to the PLS. This first core may produce hot spots at certain locations and the code changes the core configuration based on the following general rules:

(1) If a location is a control rod location, then only IFBA's are allowed;

(2) Any fresh fuel in the inner regions should have BP in it. An inner region location is any location that is not in the periphery and is not directly next to the periphery.

(3) No two fresh fuel assemblies are allowed to be adjacent to each other unless at least one of them is not in the inner regions.

(4) If possible, always avoid allocating fresh fuel assemblies in the periphery region.

(5) If possible, avoid use of fresh fuel assemblies in the region that is next to the periphery where fast neutron fluence is required to be minimized.

(6) The total number of fresh fuel assemblies should be equal to the required value.

Another major difference between the two main codes is the way the core data is stored.. The TMI-1 code uses a data base oriented interaction fuel management system to maintain the in-core history of the TMI-1 power plant.[42] This sub-system has a user-friendly human-machine interface that enables it to interact separately with the main plant data, gather information for future use, calculate, store or retrieve cross-section data for use by ADMARC. During the optimization process this sub-system is used to automatically provide required information and cross-section data for the fuel assemblies used in the core and then run ADMARC. ADMARC has been enlarged with additional subroutines that perform the various techniques to assign fresh fuel loading split the fuel enrichment, establish BP loading. These subroutines permit this main code to be written completely in FORTRAN language as shown in Fig. 31.

The BV expert system code has its main program written in C-language and calls all of the FORTRAN subroutines to perform the reactor physics calculations. The if-then rules are used to load the core according to the priority schemes and for BP placement. This expert system code is shown in Fig. 32.

The final output of both expert system codes is a summary report of the two core optimum loading configurations.

V. The Boiling Water and CANDU Reactors

Optimization of the reload core for the Boiling Water Reactor can follow the same methods used for the PWR. However, the compulational time for performing such calculations take much longer because of the void production and the need for control rod adjustment for criticality. A simple method was employed by Kim et al to increase discharge-burnup for a multicycle reload design.[43] A one dimensional code "HUDDLE" was used to optimize the BWR using basically the methods described in this report. This was achieved by assuming an average void distribution in each region and cylindricizing the core into homogenized rings.

The results of this work showed how to load the core ringwise to optimize the reload configuration. Single fuel element placement could not be determined. However, the analysis did show additional reduction in fuel costs could be achieved. To do this one must optimize several cycles and include maximizing discharge burnup as one of the objectives. The result showed a slight increase in the fuel cost of the current cycle, but overall lower costs for subsequent cycles. These costs savings are small compared to what can be achieved from going to a standard core design to an optimized design.

The CANDU reactor is loaded on-line and, therefore, presents a completely different problem when optimizing its fuel cycle. Here, if the enrichment of the fresh fuel is fixed, e.g., natural fuel, optimization would require that all of the fuel be pushed through the core at a rate that maximizes its discharge burnup. Thus, fuel at the periphery would be moved through the core at a slower rate than those internal to the core. Where optimization may be applied is to the enrichment of the fresh fuel. Here the objective function would require minimizing the fuel cost in terms of cost per unit electrical energy, e.g., mills per kw hr. One thousand mills equals a dollar and the kw hr is the electrical energy produced by the core.

VI. Conclusion

For the first time two separate single codes have been developed which use minimal information to perform all the tasks needed to produce acceptable and optimum reload cores. Thus, an inexperienced engineer can use these codes to produce optimum or superior reloads.

The basic technique is to use a priority scheme to load the fuel and then modify this loading to reduce the hot spots. This is achieved quickly by using the Haling Power distribution. Two methods are used to alter the core; one first splits the fuel enrichment and then shuffles the core if necessary to obtain the best design; and in the other method, the core fuel is first shuffled and then fuel enrichment split if necessary to obtain the final configuration. In both cases the final configuration is called a BP solvable core.

The next step places the BPs in the core so as to minimize the maximum power peak during depletion. This is achieved successfully by two different methods showing that there may be many acceptable solutions. The final step is to deplete the core and provide a summary of the results.

Important to these analyses are the uses of accurate and fast nucleonic codes to evaluate the various core configurations. These fast codes have been made accurate by benchmarking the cross-section codes to the CPM-2 code, and by using a 2D 2 group modern nodal code benchmarked to measured core data. These codes are modified to be plant specific.

The expert system and the FORTRAN main codes are now being studied to determine where improvements can be made in the future. It presently appears that (1) the BP placements should be studied to determine if some solutions are better than others. It is believed that the best solution is among the solutions obtained by these analyses; (2) the changes made to the fuel reload configuration obtained by the priority scheme obtain solutions that are in the family of optimum configurations. The reload solutions found by this method presented here may not be the true global optimum, but this true global optimum should not be far away. That is the true global optimum should not produce six or seven effective full power days more than the solution obtained by this method. Nevertheless, some new ways of checking the priority scheme with the actual fuel are planned in the future to determine the range of optimum lifetime configuration that might be obtained.

As experience is gained with these two codes it is planned to determine the best of each method and combine them into an improved main code. This main code should then exhibit the superior attributes of both of the present two codes. Even so, the present codes each should allow nuclear engineers to determine the acceptable final reload configurations in less than one day of analyses saving considerable manpower and computer funds in the process. In addition, should problems occur that require a quick change in the reload design, this code should be capable of helping the nuclear engineers find an acceptable solution within a few hours of effort on their part.

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CODE DESCRIPTION OF LWRs

(Session 2)

AKHILESH — A 2-D CORE SIMULATOR CODE FOR FUEL CYCLE ANALYSIS OF PWRs/VVERs

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Abstract

Reload pattern optimisation studies require a fast and reliable calculational model to predict the parameters like fuel cycle length, power peak factors, reactivity coefficients of fuel and moderator temperatures etc. In this paper we describe the code AKHILESH which can treat PWR or VVER type of cores by an one group, two dimensional diffusion model using albedo type boundary condition at core-reflector interface. The IAEA VVER-1000 MWe benchmark reactor (the Russian Kalinin plant) has been chosen for testing the code AKHILESH. The fuel assembly homogenised parameters as a function of burnup, boron etc were obtained by EXCEL code. For the fresh core (beginning of cycle-1) two 3-D reference solutions were obtained in two groups with explicit treatment of reflector and control regions. A finite element code FINERC was used in the first method while in the second one a finite difference code TRIHEX-3D was used with fine triangular mesh divisions. The albedo parameters were tuned in AKHILESH code such that the power distribution with six triangular divisions per assembly is comparable with the reference solutions. In AKHILESH space dependent feedback effects due to Doppler, xenon, coolant temperature etc are considered. The code was used for core followup simulations of six fuel cycles of the above IAEA VVER benchmark. AKHILESH code has the potential of being used as a level-2 code for rapid survey type studies of a large number of reload plans.

INTRODUCTION

Recently we have undertaken the complete fuel cycle analyses up to cycle 6 of the VVER-1000 benchmark as part of the activities of the IAEA CRP on in-core fuel management

code package validation for LWRs [1]. Assembly burnup code EXCEL [2] based on supercell concept has been used for performing lattice calculations. For core calculations, the one group, two dimensional PWR fuel cycle analysis code AKHILESH [3] was modified for hexagonal assemblies. This document describes the salient features of the calculational models of EXCEL and AKHILESH codes and the results of the VVER-1000 benchmark analysis. All the technical specifications for the VVER-1000 benchmark can be found in Ref. 1 and are not repeated here.

CALCULATION MODEL OF EXCEL

The basic nuclear data is 69 group WIMS library of U.K. The cross sections are condensed to 28 groups with the spectrum of a typical VVER lattice pincell. This 28 group library has been used for all the VVER lattice analyses presented here.

The EXCEL code treats the fuel-assembly unit cell as if it were part of an infinite lattice. The fuel pins in an assembly are categorised into several fuel pincell types depending on the layer count from the water gap and its location in the layer. Appropriate Dancoff factors and resonance self-shielded cross sections are evaluated. The pincells are analysed in 28 groups by interface current ($J\pm$) formalism.

The infinite lattice assumption of the pincell calculations is corrected by performing a cylindrical supercell calculation in an artificial 1-D geometry wherein a single water rod is considered at the centre, surrounded by several paste ring regions of various pincells.

A mini-supercell calculation is done for other heterogeneities in 28 groups by $J\pm$ formalism where the heterogeneity with its fine structure is at the centre, surrounded by rings of homogeneous fuel pastes of the neighbouring pincells.

Using the appropriate supercell 28 group spectra, the fuel pincell and other material cross sections are condensed to a few groups, typically 5.

The fuel assembly is then analysed in the proper 2-D geometry by solving few group diffusion equation in hexagonal geometry. We obtain K_∞, power distribution and flux and volume weighted one or two group homogenised assembly parameters from this calculation.

The equations of depletion/build-up of fuel isotopes are solved for a number of fuel pins, termed as the burnup zone. The calculation proceeds in the same manner for subsequent burnup steps.

CALCULATION MODEL OF AKHILESH

AKHILESH is a one group, two dimensional diffusion theory code. AKHILESH solves nodal type equations for the neutron source with coupling kernel being obtained from finite differencing of one group diffusion equations. Core-reflector interface is treated by an albedo type boundary condition. It uses the input database in the form of K_∞ and M^2 and their perturbations as generated by EXCEL. Axial leakage is accounted by buckling. The code can consider one mesh or six triangular meshes per hexagonal assembly.

The nominal fuel temperature (T_f) was assumed to be 830^0C and nominal coolant temperature (T_c) was 302^0C. The one group burnup dependent lattice parameters K_∞ and M^2 for the nominal conditions were evaluated for four values of boron histories viz, 0, 600, 1200 and 1800 ppm for ten fuel types used in the six fuel cycles. The above K_∞ and M^2 values are subject to many perturbations due to various physical phenomena. The major ones are space (or power) dependent Doppler, xenon, coolant temperature feedback effects.

Mean temperature of fuel in a mesh is deduced by assuming linear variation of $(T_f - T_c)$ with the power in the mesh relative to nominal average value. $\Delta K/K$ due to fuel temperature is then applied by interpolation of the tabular values evaluated as a function of burnup for each fuel type.

The $\Delta K/K$ due to xenon is evaluated as a function of burnup, boron and power rating for each fuel type. The $\Delta K/K$ is interpolated as function of relative mesh power from,

$$\frac{\Delta K}{K} = \frac{P\ (1+X_2)\ X_1}{P + X_2} - X_1$$

where X_1 is the $\Delta K/K$ due to xenon at relative average power (P=1.0). X_1 is evaluated as a function of burnup and boron for each fuel type. X_2 is evaluated for different heat rating values from EXCEL for fresh fuel without boron.

The mean coolant temperature in a mesh is assumed to be the addition of the inlet temperature and $\Delta T/2$ where ΔT is proportional to the power in that mesh. Both K_∞ and M^2 are perturbed due to T_c. These $\Delta K/K$ values are evaluated as a function of boron and burnup for each fuel type.

The presence of control (RCCAs) in a fuel assembly is also accounted by burnup dependent worths of these RCCAs. Since AKHILESH is a 2-D code fractional control is treated by weighted reduction of $\Delta K/K$ in an approximate manner.

The change in reactivity due to the difference in the instantaneous critical boron at a given cycle burnup and the average boron (history) that was present up to that burnup is also considered.

The flow of calculations in the code AKHILESH is as follows. We start with the initial burnup profile of a given cycle. An estimated boron concentration, flat guess for source distribution and unit eigenvalue are assumed initially. The innermost loop updates the eigenvalue and source distribution. Power dependent feedbacks are applied after every fifth iteration. After convergence of source and eigenvalue the boron concentration is varied such that a required K-effective or eigenvalue can be met. When the critical boron is evaluated the burnup profile is built up for the given burnup step and the calculations are repeated as before. Finally the cycle calculations are terminated when all burnup steps are completed. The burnup profile for next cycle can be reconstituted with the help of new loading

	18	19	
	1166	834	(1)
	1160	830	(2)
	1167	831	(3)

14	15	16	17
895	956	948	828
896	959	955	824
891	960	972	824

8	9	10	11	12	13
1213	929	958	1335	1243	703
1207	934	962	1327	1241	700
1221	947	961	1305	1244	691

1	2	3	4	5	6	7
876	928	956	1168	1034	1086	1090
876	934	964	1159	1039	1096	1086
896	959	989	1151	1029	1091	1088

(1) FEM 3-D 2G 6 TRI/HEX (Q-Q) - FINERC
(2) FDM 3-D 2G 54 TRI/HEX - TRIHEX-3D
(3) FDM 2-D 1G 6 TRI/HEX - AKHILESH

Fig.1 VVER-1000 MWe IAEA BENCHMARK - BEGINNING OF CYCLE-1 COMPARISON OF ASSEMBLY POWER DISTRIBUTION

map. A boron history profile is also followed up and reconstituted for next cycle which is evaluated in a similar way.

RESULTS OF VVER-1000 MWe REACTOR ANALYSIS

The lattice results of EXCEL and core results with one mesh per assembly were presented in an IAEA RCM [4]. The core results with six triangular meshes per assembly are presented here.

Text cont. on p. 82.

Table-I

VVER-1000 Core Fuel Cycle Analysis by AKHILESH - Cycle - 1

Cycle Burnup FPD	% Full Power	T-inlet ^{0}C	Critical Boron ppm	RPPF
0.00	48	283.4	1291	1.285
6.40	35	282.5	1314	1.278
11.50	50	283.4	1247	1.271
19.20	53	283.7	1213	1.266
27.50	52	283.6	1201	1.271
39.40	74	285.0	1109	1.283
47.30	46	283.7	1161	1.284
50.80	68	284.7	1088	1.286
58.10	55	283.8	1095	1.283
58.80	68	284.7	1058	1.282
70.50	17	281.1	1211	1.261
72.00	57	283.9	1040	1.278
74.50	62	284.2	1016	1.281
84.60	79	285.5	932	1.276
103.20	80	285.5	848	1.265
123.50	79	285.5	755	1.252
130.40	66	284.6	759	1.251
132.70	77	285.3	717	1.246
142.20	79	285.5	669	1.243
154.80	58	285.9	663	1.231
165.70	93	286.4	522	1.231
180.20	103	287.2	431	1.229
197.80	34	282.3	547	1.213
200.00	103	287.2	332	1.217
235.30	101	287.1	165	1.208

Table-II

VVER-1000 Core Fuel Cycle Analysis by AKHILESH - Cycle - 2

Cycle Burnup FPD	% Full Power	T-inlet ^{0}C	Critical Boron ppm	RPPF
0.00	100	287.0	1097	1.414
10.00	100	287.0	1064	1.432
13.60	97	286.7	1052	1.425
20.00	100	287.0	1012	1.405
25.00	69	284.8	1074	1.408
26.10	89	286.1	1009	1.391
27.00	100	287.0	977	1.378
35.90	100	287.0	934	1.356
68.30	100	287.0	773	1.278
78.10	74	285.1	793	1.269
82.30	100	287.0	708	1.253
100.10	95	286.6	639	1.235
110.90	100	287.0	578	1.223
121.80	100	287.0	528	1.214
155.10	100	287.0	372	1.190
170.30	100	287.0	301	1.183
200.00	73	284.8	237	1.181
205.30	90	286.2	163	1.170
214.80	65	284.5	193	1.172
217.50	90	286.2	107	1.165
238.00	90	286.2	19	1.163
243.40	90	286.2	-5	1.162
265.80	90	286.2	-103	1.156

Table-III

VVER-1000 Core Fuel Cycle Analysis by AKHILESH - Cycle - 3

Cycle Burnup FPD	% Full Power	T-inlet ^0C	Critical Boron ppm	RPPF
0.00	80	285.5	993	1.322
11.00	101	287.1	876	1.311
19.90	101	287.1	830	1.307
37.00	60	284.1	853	1.311
65.20	100	287.0	607	1.298
86.20	100	287.0	509	1.285
89.30	14	280.0	862	1.284
90.20	72	284.9	563	1.282
92.20	99	286.9	483	1.283
95.20	100	287.0	467	1.280
97.20	97	286.8	466	1.278
117.70	100	287.0	362	1.267
120.40	97	286.7	356	1.265
147.00	100	287.0	224	1.249
155.80	100	287.0	183	1.244
160.70	93	286.4	177	1.242
163.70	100	287.0	147	1.240
173.70	100	287.0	100	1.234
182.80	101	287.1	56	1.231
188.60	101	287.1	29	1.229
197.50	81	285.5	41	1.223
200.00	95	286.6	-8	1.224
208.80	99	286.9	-58	1.220
211.70	95	296.6	-54	1.221
214.60	101	287.1	-88	1.218
217.40	93	286.4	-81	1.216
221.80	87	286.0	-85	1.215
222.30	62	284.2	-9	1.212

Table-IV

VVER-1000 Core Fuel Cycle Analysis by AKHILESH - Cycle - 4

Cycle Burnup FPD	% Full Power	T-inlet ^{0}C	Critical Boron ppm	RPPF
0.00	11	270.0	1422	1.327
1.20	61	284.1	1083	1.305
2.20	100	287.0	971	1.301
4.70	101	287.1	953	1.306
7.40	101	287.1	939	1.303
13.50	101	287.1	913	1.291
23.50	101	287.1	868	1.269
39.50	100	287.0	803	1.243
52.90	97	286.7	755	1.234
76.60	99	286.9	654	1.262
87.50	101	287.1	603	1.272
93.50	99	287.0	581	1.276
115.60	101	287.1	487	1.294
131.50	98	286.8	431	1.300
160.40	100	287.0	314	1.309
178.00	100	287.0	246	1.314
197.40	97	286.7	178	1.318
207.40	100	287.0	133	1.319
217.40	100	287.0	95	1.319
228.30	101	287.1	52	1.319
238.30	100	287.0	16	1.319
242.30	101	287.0	-1	1.319
250.30	100	287.0	-29	1.319

Table-V

VVER-1000 Core Fuel Cycle Analysis by AKHILESH - Cycle - 5

Cycle Burnup FPD	% Full Power	T-inlet ^{0}C	Critical Boron ppm	RPPF
0.00	9	280.0	1629	1.410
5.39	96	286.6	1229	1.425
17.07	85	285.7	1196	1.420
19.02	98	286.8	1153	1.410
25.94	98	286.8	1125	1.393
40.69	99	286.8	1061	1.360
52.51	99	286.9	1011	1.334
63.46	100	287.0	964	1.309
71.07	100	287.0	933	1.294
79.65	99	287.0	899	1.292
87.33	99	286.9	866	1.290
100.00	100	286.9	814	1.292
102.21	99	286.9	804	1.291
122.21	99	286.8	719	1.286
123.00	99	286.8	718	1.288
140.21	99	286.8	643	1.283
141.10	99	287.0	641	1.284
144.00	100	287.0	622	1.279
200.00	100	287.0	400	1.270
236.00	100	287.0	258	1.264

Table-VI

VVER-1000 Core Fuel Cycle Analysis by AKHILESH - Cycle - 6

Cycle Burnup FPD	% Full Power	T-inlet ^{0}C	Critical Boron ppm	RPPF
0.00	83	285.7	1399	1.262
2.00	83	285.7	1389	1.265
5.00	97	286.7	1336	1.260
10.00	99	286.9	1311	1.256
12.00	100	287.0	1304	1.260
19.00	99	286.8	1271	1.249
23.00	95	286.5	1262	1.242
26.00	100	287.0	1237	1.235
29.00	99	286.9	1227	1.233
32.00	99	286.9	1214	1.230
39.00	96	286.6	1186	1.216
68.00	99	286.8	1056	1.226
95.00	99	286.9	933	1.248
115.00	97	286.7	851	1.258
117.00	100	287.0	835	1.260
165.00	100	287.0	628	1.279
176.00	100	287.0	585	1.277
183.00	100	287.0	554	1.280
189.00	99	286.9	530	1.283
205.00	100	287.0	462	1.282

Table-VII Assemblywise Power Distribution - Cycle - 1

Burnup	Assembly Power																		
FPD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
0.0	0.921	0.974	0.998	1.144	1.041	1.085	1.069	1.215	0.962	0.977	1.285	1.207	0.676	0.917	0.970	0.965	0.830	1.147	0.839
6.4	0.931	0.984	1.012	1.175	1.047	1.080	1.056	1.225	0.976	0.986	1.278	1.194	0.668	0.924	0.970	0.960	0.821	1.139	0.831
11.5	0.955	1.007	1.029	1.175	1.051	1.076	1.039	1.241	0.991	0.996	1.271	1.179	0.657	0.938	0.976	0.957	0.810	1.132	0.822
19.2	0.974	1.023	1.039	1.160	1.053	1.074	1.029	1.249	0.999	1.001	1.266	1.169	0.651	0.947	0.983	0.958	0.805	1.129	0.818
27.5	0.995	1.045	1.065	1.211	1.062	1.064	1.007	1.271	1.023	1.015	1.256	1.147	0.637	0.960	0.983	0.948	0.788	1.114	0.802
39.4	1.021	1.067	1.084	1.220	1.067	1.057	0.986	1.283	1.042	1.028	1.245	1.127	0.625	0.978	0.991	0.946	0.776	1.105	0.793
47.3	1.021	1.068	1.084	1.222	1.067	1.056	0.986	1.284	1.042	1.027	1.244	1.127	0.626	0.977	0.990	0.946	0.777	1.105	0.794
50.8	1.030	1.075	1.092	1.230	1.070	1.054	0.977	1.286	1.051	1.034	1.239	1.118	0.621	0.985	0.994	0.945	0.772	1.101	0.789
58.1	1.033	1.077	1.089	1.209	1.068	1.055	0.978	1.283	1.048	1.032	1.238	1.119	0.623	0.987	0.997	0.948	0.774	1.104	0.792
58.8	1.036	1.078	1.091	1.209	1.069	1.054	0.975	1.282	1.051	1.035	1.236	1.116	0.621	0.990	0.999	0.949	0.773	1.103	0.791
70.5	1.014	1.056	1.065	1.160	1.059	1.064	0.999	1.261	1.025	1.018	1.245	1.138	0.637	0.974	0.997	0.959	0.791	1.118	0.808
72.0	1.035	1.077	1.092	1.219	1.071	1.052	0.971	1.278	1.054	1.038	1.232	1.111	0.621	0.994	1.001	0.949	0.772	1.101	0.791
74.5	1.040	1.082	1.096	1.220	1.071	1.051	0.968	1.281	1.057	1.040	1.230	1.108	0.619	0.996	1.002	0.949	0.770	1.100	0.790
84.6	1.046	1.085	1.098	1.218	1.073	1.049	0.962	1.276	1.061	1.044	1.223	1.101	0.618	1.004	1.007	0.951	0.769	1.099	0.790
103.2	1.045	1.081	1.095	1.215	1.072	1.046	0.958	1.265	1.063	1.047	1.216	1.096	0.620	1.010	1.012	0.955	0.772	1.099	0.794
123.5	1.044	1.077	1.089	1.197	1.070	1.047	0.960	1.252	1.059	1.046	1.211	1.095	0.626	1.013	1.017	0.961	0.778	1.102	0.800
130.4	1.042	1.076	1.090	1.206	1.070	1.045	0.959	1.251	1.061	1.047	1.209	1.093	0.626	1.013	1.016	0.960	0.778	1.100	0.800
132.7	1.043	1.075	1.087	1.192	1.069	1.047	0.960	1.246	1.059	1.047	1.208	1.094	0.628	1.014	1.019	0.963	0.780	1.102	0.803
142.2	1.043	1.074	1.088	1.201	1.070	1.044	0.956	1.243	1.061	1.049	1.204	1.090	0.628	1.018	1.020	0.963	0.780	1.101	0.803
154.8	1.036	1.066	1.076	1.173	1.065	1.047	0.965	1.231	1.051	1.043	1.204	1.096	0.636	1.014	1.022	0.970	0.789	1.107	0.812
165.7	1.040	1.070	1.084	1.199	1.069	1.042	0.954	1.231	1.062	1.051	1.197	1.085	0.632	1.023	1.024	0.968	0.785	1.100	0.808
180.2	1.043	1.072	1.088	1.214	1.070	1.038	0.950	1.229	1.067	1.055	1.191	1.079	0.633	1.026	1.024	0.966	0.784	1.096	0.807
197.8	1.029	1.057	1.067	1.162	1.061	1.046	0.970	1.213	1.045	1.040	1.196	1.097	0.649	1.013	1.023	0.976	0.801	1.108	0.823
200.0	1.039	1.066	1.080	1.192	1.067	1.040	0.953	1.217	1.061	1.052	1.188	1.081	0.640	1.026	1.028	0.972	0.791	1.099	0.814
235.3	1.040	1.065	1.077	1.180	1.065	1.040	0.955	1.208	1.059	1.051	1.181	1.079	0.648	1.027	1.029	0.976	0.797	1.098	0.819

Table-VIII Assemblywise Power Distribution - Cycle - 2

Burnup	Assembly Power																		
FPD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
0.0	0.953	1.386	1.414	1.173	1.103	0.795	0.894	1.288	1.376	1.265	0.947	1.020	0.571	1.173	1.009	0.741	0.694	1.085	0.725
10.0	1.114	1.432	1.424	1.173	1.101	0.793	0.885	1.304	1.376	1.261	0.945	1.011	0.566	1.170	1.006	0.739	0.688	1.075	0.719
13.6	1.110	1.425	1.418	1.170	1.100	0.796	0.889	1.298	1.371	1.259	0.947	1.014	0.569	1.168	1.007	0.742	0.691	1.078	0.722
20.0	1.096	1.405	1.400	1.164	1.099	0.801	0.897	1.283	1.358	1.253	0.950	1.022	0.576	1.162	1.008	0.748	0.699	1.084	0.729
25.0	1.098	1.408	1.406	1.175	1.103	0.799	0.891	1.288	1.363	1.258	0.951	1.017	0.571	1.167	1.010	0.745	0.694	1.080	0.724
26.1	1.087	1.391	1.389	1.160	1.099	0.805	0.902	1.272	1.349	1.250	0.953	1.027	0.580	1.159	1.010	0.752	0.704	1.088	0.734
27.0	1.077	1.378	1.378	1.157	1.098	0.809	0.909	1.262	1.341	1.247	0.954	1.033	0.585	1.155	1.010	0.756	0.709	1.093	0.739
35.9	1.058	1.350	1.356	1.155	1.098	0.816	0.920	1.241	1.324	1.240	0.958	1.044	0.594	1.148	1.012	0.764	0.719	1.100	0.748
68.3	1.002	1.264	1.278	1.107	1.088	0.844	0.964	1.171	1.260	1.209	0.971	1.086	0.630	1.120	1.018	0.796	0.760	1.132	0.787
78.1	0.996	1.256	1.269	1.096	1.088	0.847	0.968	1.164	1.253	1.206	0.974	1.090	0.634	1.118	1.020	0.800	0.764	1.135	0.791
82.3	0.986	1.239	1.253	1.089	1.085	0.854	0.977	1.150	1.240	1.199	0.976	1.098	0.642	1.111	1.020	0.808	0.773	1.140	0.799
100.1	0.974	1.218	1.235	1.087	1.084	0.862	0.983	1.134	1.226	1.192	0.980	1.103	0.650	1.104	1.021	0.817	0.782	1.143	0.807
110.9	0.966	1.203	1.223	1.085	1.084	0.867	0.987	1.124	1.216	1.187	0.983	1.107	0.655	1.100	1.022	0.823	0.788	1.145	0.813
121.8	0.961	1.193	1.214	1.084	1.084	0.871	0.990	1.116	1.209	1.184	0.985	1.109	0.659	1.097	1.023	0.827	0.793	1.146	0.817
155.1	0.952	1.169	1.190	1.071	1.082	0.883	0.996	1.097	1.188	1.173	0.992	1.114	0.671	1.089	1.027	0.842	0.806	1.149	0.829
170.3	0.950	1.162	1.183	1.071	1.082	0.887	0.997	1.092	1.182	1.170	0.994	1.113	0.675	1.087	1.028	0.847	0.810	1.148	0.832
200.0	0.955	1.161	1.181	1.072	1.085	0.892	0.990	1.093	1.180	1.169	0.998	1.107	0.675	1.089	1.031	0.851	0.809	1.142	0.831
205.3	0.950	1.151	1.170	1.060	1.082	0.897	0.996	1.084	1.170	1.163	1.000	1.112	0.681	1.084	1.031	0.858	0.816	1.146	0.838
214.8	0.954	1.156	1.172	1.051	1.083	0.897	0.993	1.088	1.172	1.164	1.002	1.110	0.679	1.087	1.034	0.858	0.814	1.144	0.836
217.5	0.948	1.146	1.164	1.053	1.082	0.901	0.997	1.079	1.165	1.160	1.002	1.113	0.684	1.082	1.033	0.862	0.819	1.146	0.841
238.0	0.949	1.142	1.162	1.064	1.083	0.904	0.993	1.078	1.163	1.160	1.004	1.109	0.684	1.082	1.034	0.866	0.819	1.142	0.840
243.4	0.950	1.141	1.160	1.062	1.083	0.905	0.993	1.077	1.162	1.158	1.005	1.109	0.685	1.082	1.034	0.867	0.820	1.142	0.841
265.8	0.951	1.136	1.154	1.059	1.083	0.909	0.993	1.073	1.156	1.155	1.007	1.108	0.688	1.080	1.036	0.873	0.823	1.142	0.844

Table-IX Assemblywise Power Distribution - Cycle - 3

Burnup	Assembly Power																		
FPD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
0.0	0.803	1.026	1.066	0.955	0.947	0.964	0.978	0.952	1.208	1.178	1.164	1.181	0.437	1.028	1.152	0.887	0.796	1.322	0.898
11.0	0.816	1.039	1.075	0.960	0.949	0.962	0.973	0.961	1.214	1.179	1.161	1.174	0.438	1.030	1.149	0.885	0.795	1.311	0.894
19.9	0.821	1.043	1.076	0.961	0.949	0.962	0.974	0.964	1.213	1.177	1.159	1.173	0.440	1.030	1.147	0.886	0.797	1.307	0.894
37.0	0.829	1.046	1.066	0.908	0.947	0.972	0.982	0.963	1.196	1.167	1.163	1.180	0.447	1.028	1.151	0.895	0.805	1.311	0.901
65.2	0.831	1.041	1.067	0.957	0.950	0.969	0.989	0.961	1.190	1.160	1.150	1.180	0.456	1.019	1.134	0.896	0.817	1.298	0.908
86.2	0.841	1.047	1.071	0.966	0.954	0.971	0.989	0.967	1.189	1.158	1.146	1.175	0.460	1.020	1.129	0.897	0.819	1.285	0.907
89.3	0.857	1.069	1.082	0.924	0.956	0.973	0.977	0.983	1.198	1.165	1.155	1.167	0.451	1.033	1.143	0.895	0.804	1.284	0.894
90.2	0.848	1.055	1.076	0.961	0.955	0.971	0.986	0.973	1.191	1.159	1.147	1.172	0.459	1.023	1.130	0.896	0.817	1.282	0.904
92.2	0.845	1.050	1.072	0.962	0.954	0.971	0.989	0.969	1.188	1.156	1.146	1.174	0.462	1.020	1.129	0.897	0.821	1.283	0.907
95.2	0.847	1.051	1.073	0.967	0.955	0.971	0.988	0.971	1.189	1.156	1.145	1.172	0.462	1.020	1.128	0.897	0.821	1.280	0.906
97.2	0.849	1.053	1.075	0.971	0.956	0.970	0.987	0.973	1.190	1.157	1.144	1.171	0.462	1.021	1.128	0.896	0.820	1.278	0.905
117.7	0.861	1.063	1.080	0.973	0.959	0.971	0.986	0.980	1.189	1.156	1.142	1.165	0.466	1.022	1.125	0.897	0.821	1.267	0.903
120.4	0.863	1.064	1.081	0.973	0.959	0.971	0.985	0.981	1.189	1.156	1.141	1.164	0.466	1.023	1.125	0.897	0.821	1.265	0.902
147.0	0.879	1.077	1.089	0.981	0.964	0.971	0.981	0.991	1.191	1.155	1.138	1.155	0.471	1.025	1.120	0.898	0.822	1.249	0.898
155.8	0.885	1.081	1.092	0.986	0.965	0.971	0.979	0.996	1.192	1.156	1.136	1.151	0.472	1.027	1.119	0.898	0.821	1.244	0.896
160.7	0.888	1.084	1.093	0.982	0.966	0.972	0.978	0.998	1.192	1.155	1.136	1.150	0.473	1.027	1.119	0.898	0.821	1.242	0.895
163.7	0.889	1.084	1.095	0.988	0.967	0.971	0.977	0.999	1.193	1.155	1.136	1.148	0.473	1.027	1.118	0.898	0.821	1.240	0.895
173.7	0.894	1.088	1.097	0.990	0.968	0.971	0.976	1.002	1.193	1.155	1.134	1.145	0.475	1.028	1.118	0.898	0.821	1.234	0.893
182.8	0.899	1.091	1.099	0.990	0.969	0.972	0.975	1.005	1.193	1.155	1.133	1.143	0.476	1.029	1.117	0.899	0.822	1.231	0.892
188.6	0.902	1.093	1.100	0.990	0.970	0.972	0.974	1.006	1.192	1.154	1.133	1.142	0.477	1.030	1.117	0.899	0.822	1.229	0.892
197.5	0.909	1.100	1.105	0.994	0.973	0.972	0.970	1.012	1.195	1.156	1.132	1.137	0.477	1.032	1.117	0.898	0.819	1.223	0.888
200.0	0.908	1.098	1.103	0.992	0.972	0.972	0.972	1.010	1.193	1.154	1.132	1.138	0.478	1.031	1.116	0.899	0.821	1.224	0.890
208.8	0.911	1.099	1.104	0.995	0.973	0.973	0.971	1.012	1.192	1.154	1.131	1.136	0.480	1.031	1.115	0.900	0.822	1.220	0.889
211.7	0.911	1.098	1.102	0.989	0.973	0.974	0.972	1.011	1.189	1.152	1.132	1.137	0.482	1.031	1.116	0.901	0.824	1.221	0.891
214.6	0.913	1.101	1.105	0.997	0.974	0.973	0.970	1.013	1.192	1.154	1.131	1.135	0.481	1.032	1.115	0.900	0.822	1.218	0.888
217.4	0.916	1.103	1.107	0.998	0.975	0.973	0.969	1.015	1.193	1.154	1.131	1.133	0.481	1.033	1.115	0.900	0.821	1.216	0.887
221.8	0.919	1.105	1.108	0.997	0.976	0.973	0.969	1.017	1.193	1.154	1.130	1.132	0.482	1.033	1.115	0.900	0.821	1.215	0.886
222.3	0.923	1.111	1.113	1.001	0.977	0.972	0.964	1.022	1.198	1.157	1.131	1.128	0.479	1.037	1.116	0.898	0.816	1.212	0.881

Table-X Assemblywise Power Distribution - Cycle - 4

Burnup	Assembly Power																		
FPD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
0.0	1.064	0.876	1.045	0.847	1.164	1.236	1.111	1.052	0.864	0.860	1.007	1.158	0.734	0.953	0.884	1.327	0.976	1.145	0.985
1.2	1.305	1.063	1.261	1.090	1.259	1.168	0.944	1.261	1.031	0.967	0.998	1.028	0.604	1.072	0.908	1.187	0.811	1.112	0.841
2.2	1.301	1.056	1.257	1.098	1.261	1.167	0.946	1.255	1.029	0.966	0.997	1.028	0.605	1.070	0.907	1.189	0.812	1.113	0.845
4.7	1.306	1.062	1.264	1.108	1.262	1.166	0.942	1.262	1.035	0.970	0.997	1.025	0.602	1.074	0.907	1.182	0.807	1.109	0.839
7.4	1.303	1.061	1.264	1.114	1.262	1.164	0.941	1.262	1.036	0.971	0.996	1.024	0.602	1.074	0.907	1.181	0.807	1.108	0.839
13.5	1.291	1.053	1.255	1.111	1.258	1.164	0.946	1.251	1.032	0.969	0.997	1.026	0.606	1.072	0.908	1.186	0.812	1.110	0.844
23.5	1.269	1.036	1.234	1.100	1.253	1.164	0.955	1.230	1.020	0.965	0.998	1.032	0.614	1.065	0.910	1.197	0.823	1.116	0.856
39.5	1.227	1.003	1.196	1.080	1.243	1.164	0.973	1.190	0.999	0.956	0.999	1.043	0.630	1.052	0.913	1.217	0.844	1.128	0.878
52.9	1.195	0.980	1.167	1.064	1.234	1.164	0.987	1.160	0.982	0.949	1.000	1.051	0.642	1.043	0.916	1.233	0.860	1.137	0.895
76.6	1.144	0.940	1.120	1.036	1.216	1.161	1.012	1.111	0.954	0.936	1.000	1.063	0.665	1.024	0.918	1.262	0.890	1.151	0.926
87.5	1.123	0.924	1.101	1.024	1.207	1.159	1.023	1.091	0.942	0.930	1.000	1.068	0.675	1.016	0.918	1.272	0.903	1.156	0.940
93.5	1.114	0.918	1.093	1.020	1.202	1.158	1.028	1.083	0.938	0.928	0.999	1.070	0.680	1.013	0.919	1.276	0.909	1.157	0.946
115.6	1.083	0.896	1.064	0.997	1.184	1.153	1.045	1.055	0.920	0.918	0.997	1.078	0.698	1.001	0.919	1.294	0.931	1.165	0.968
131.5	1.067	0.887	1.052	0.991	1.173	1.149	1.053	1.043	0.914	0.914	0.996	1.079	0.706	0.997	0.920	1.300	0.941	1.166	0.978
160.4	1.045	0.875	1.034	0.978	1.154	1.140	1.066	1.024	0.905	0.908	0.992	1.082	0.723	0.989	0.921	1.309	0.958	1.166	0.994
178.0	1.034	0.869	1.024	0.972	1.143	1.134	1.072	1.015	0.900	0.905	0.991	1.082	0.731	0.985	0.922	1.314	0.967	1.166	1.003
197.4	1.024	0.864	1.015	0.964	1.132	1.129	1.079	1.006	0.896	0.903	0.989	1.083	0.740	0.981	0.923	1.318	0.976	1.165	1.012
207.4	1.020	0.862	1.011	0.961	1.127	1.126	1.083	1.002	0.894	0.901	0.988	1.083	0.745	0.979	0.922	1.319	0.981	1.164	1.016
217.4	1.017	0.861	1.009	0.962	1.124	1.123	1.085	1.000	0.893	0.901	0.986	1.082	0.748	0.978	0.922	1.319	0.984	1.162	1.018
228.3	1.015	0.860	1.006	0.961	1.119	1.120	1.088	0.997	0.893	0.900	0.985	1.081	0.753	0.977	0.922	1.319	0.988	1.160	1.021
238.3	1.012	0.859	1.003	0.958	1.115	1.118	1.090	0.994	0.891	0.899	0.984	1.081	0.757	0.975	0.922	1.319	0.991	1.159	1.024
242.3	1.010	0.859	1.002	0.958	1.114	1.116	1.091	0.993	0.891	0.899	0.984	1.081	0.758	0.975	0.922	1.319	0.992	1.158	1.025
250.3	1.010	0.859	1.001	0.957	1.111	1.114	1.093	0.992	0.891	0.899	0.983	1.080	0.761	0.974	0.922	1.319	0.994	1.157	1.026

Table-XI Assemblywise Power Distribution - Cycle - 5

Burnup	Assembly Power																	
FPD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
0.0	1.019	1.101	1.410	1.159	0.993	1.170	0.644	1.404	1.011	1.287	1.011	1.097	0.514	0.982	1.263	1.122	0.712	0.852	0.616
5.4	1.038	1.116	1.425	1.299	1.005	1.145	0.637	1.404	1.034	1.301	0.997	1.070	0.510	0.983	1.238	1.097	0.701	0.848	0.611
17.1	1.042	1.117	1.420	1.295	1.006	1.145	0.640	1.400	1.034	1.295	0.999	1.072	0.511	0.985	1.237	1.097	0.702	0.851	0.614
19.0	1.037	1.110	1.410	1.286	1.005	1.149	0.644	1.390	1.029	1.294	1.000	1.076	0.515	0.982	1.237	1.102	0.707	0.854	0.618
25.9	1.026	1.097	1.393	1.278	1.005	1.155	0.648	1.373	1.023	1.295	1.002	1.080	0.519	0.981	1.237	1.108	0.712	0.857	0.622
40.7	1.002	1.071	1.360	1.266	1.006	1.167	0.658	1.338	1.010	1.294	1.007	1.089	0.529	0.978	1.234	1.120	0.725	0.864	0.631
52.5	0.982	1.049	1.334	1.259	1.007	1.177	0.665	1.310	1.000	1.294	1.010	1.095	0.538	0.975	1.232	1.131	0.736	0.869	0.638
63.5	0.963	1.029	1.309	1.248	1.008	1.187	0.672	1.284	0.990	1.295	1.013	1.102	0.546	0.972	1.230	1.141	0.746	0.873	0.645
71.1	0.952	1.017	1.292	1.237	1.008	1.194	0.677	1.267	0.983	1.294	1.015	1.106	0.552	0.970	1.228	1.148	0.753	0.877	0.649
79.6	0.941	1.005	1.277	1.230	1.008	1.200	0.682	1.251	0.977	1.292	1.017	1.110	0.558	0.967	1.226	1.155	0.761	0.880	0.654
87.3	0.934	0.996	1.263	1.218	1.006	1.205	0.687	1.237	0.971	1.290	1.018	1.114	0.564	0.965	1.224	1.162	0.768	0.883	0.659
100.0	0.922	0.982	1.247	1.219	1.007	1.210	0.692	1.220	0.965	1.292	1.019	1.116	0.571	0.963	1.220	1.167	0.777	0.885	0.664
102.2	0.919	0.978	1.240	1.207	1.006	1.213	0.694	1.214	0.962	1.291	1.020	1.119	0.574	0.962	1.220	1.170	0.780	0.887	0.666
122.2	0.900	0.957	1.211	1.190	1.005	1.224	0.705	1.184	0.949	1.286	1.022	1.126	0.588	0.956	1.214	1.184	0.797	0.893	0.676
123.0	0.899	0.956	1.212	1.198	1.005	1.223	0.704	1.184	0.950	1.288	1.021	1.124	0.587	0.957	1.213	1.183	0.796	0.892	0.676
140.2	0.887	0.941	1.189	1.177	1.003	1.231	0.713	1.162	0.940	1.283	1.023	1.131	0.600	0.952	1.208	1.193	0.811	0.897	0.684
141.1	0.886	0.941	1.190	1.185	1.004	1.230	0.712	1.162	0.941	1.284	1.022	1.130	0.599	0.953	1.208	1.191	0.810	0.896	0.683
144.0	0.883	0.937	1.179	1.154	1.000	1.236	0.717	1.155	0.934	1.279	1.024	1.137	0.605	0.950	1.209	1.199	0.817	0.900	0.688
200.0	0.869	0.916	1.138	1.127	0.996	1.247	0.736	1.115	0.918	1.270	1.024	1.141	0.633	0.942	1.192	1.215	0.847	0.908	0.707
236.0	0.865	0.908	1.120	1.115	0.994	1.250	0.745	1.097	0.913	1.264	1.022	1.141	0.649	0.939	1.182	1.220	0.864	0.912	0.716

Table-XII Assemblywise Power Distribution - Cycle - 6

Burnup	Assembly Power																		
FPD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
0.0	1.175	1.018	1.044	1.053	0.804	1.127	0.989	1.262	1.036	1.108	1.001	1.212	0.610	1.020	1.091	1.175	0.785	1.098	0.770
2.0	1.177	1.021	1.047	1.054	0.805	1.126	0.988	1.265	1.038	1.109	1.000	1.210	0.609	1.022	1.091	1.172	0.784	1.097	0.769
5.0	1.173	1.017	1.043	1.051	0.806	1.126	0.989	1.260	1.035	1.107	1.000	1.212	0.612	1.019	1.090	1.176	0.787	1.098	0.773
10.0	1.168	1.014	1.043	1.061	0.807	1.123	0.987	1.256	1.035	1.108	1.000	1.209	0.612	1.019	1.089	1.177	0.788	1.098	0.775
12.0	1.169	1.016	1.048	1.084	0.810	1.119	0.982	1.260	1.041	1.112	0.998	1.203	0.609	1.021	1.088	1.172	0.785	1.094	0.772
19.0	1.163	1.010	1.040	1.068	0.809	1.119	0.984	1.249	1.034	1.106	0.999	1.206	0.614	1.018	1.088	1.179	0.792	1.098	0.780
23.0	1.158	1.006	1.034	1.054	0.808	1.121	0.986	1.242	1.029	1.103	1.000	1.208	0.617	1.016	1.089	1.185	0.797	1.102	0.785
26.0	1.154	1.001	1.030	1.051	0.808	1.120	0.987	1.235	1.025	1.100	1.000	1.209	0.619	1.014	1.089	1.190	0.801	1.104	0.789
29.0	1.151	0.999	1.030	1.060	0.809	1.118	0.985	1.233	1.026	1.101	1.000	1.207	0.619	1.014	1.088	1.190	0.801	1.103	0.790
32.0	1.149	0.997	1.028	1.058	0.809	1.117	0.985	1.230	1.024	1.099	1.000	1.206	0.621	1.013	1.088	1.192	0.804	1.104	0.793
39.0	1.140	0.989	1.016	1.030	0.806	1.119	0.990	1.216	1.014	1.092	1.001	1.211	0.628	1.008	1.089	1.203	0.814	1.111	0.804
68.0	1.111	0.965	0.997	1.034	0.806	1.108	0.984	1.181	0.999	1.080	0.999	1.204	0.640	0.998	1.086	1.225	0.836	1.121	0.833
95.0	1.081	0.941	0.973	1.006	0.803	1.103	0.986	1.146	0.979	1.066	0.999	1.205	0.656	0.987	1.084	1.248	0.863	1.133	0.863
115.0	1.064	0.929	0.963	1.007	0.804	1.097	0.984	1.130	0.972	1.059	0.997	1.202	0.665	0.981	1.081	1.258	0.876	1.135	0.878
117.0	1.061	0.927	0.961	1.005	0.803	1.097	0.985	1.126	0.970	1.058	0.997	1.202	0.666	0.980	1.080	1.260	0.878	1.136	0.881
165.0	1.027	0.905	0.939	0.980	0.803	1.089	0.985	1.092	0.951	1.043	0.995	1.197	0.687	0.968	1.076	1.279	0.908	1.146	0.916
176.0	1.024	0.905	0.943	1.000	0.807	1.083	0.979	1.093	0.955	1.046	0.993	1.189	0.687	0.970	1.073	1.277	0.908	1.142	0.917
183.0	1.022	0.904	0.940	0.989	0.806	1.083	0.979	1.089	0.952	1.043	0.993	1.189	0.691	0.968	1.073	1.280	0.913	1.143	0.922
189.0	1.019	0.902	0.936	0.977	0.805	1.083	0.981	1.085	0.948	1.040	0.993	1.190	0.694	0.967	1.073	1.283	0.917	1.145	0.927
205.0	1.017	0.903	0.938	0.986	0.808	1.078	0.976	1.083	0.950	1.040	0.991	1.182	0.696	0.967	1.070	1.282	0.921	1.143	0.932

In order to tune the albedo boundary conditions at the core reflector interface, the core loading of beginning of cycle-1 was chosen. Two group 3-D calculations were done by both finite element and fine mesh finite difference methods with the codes FINERC [5] and TRIHEX-3D [6] respectively. The water-stainless steel layers surrounding the core were treated explicitly. A boron concentration of 1300 ppm was considered. In FEM, 6 triangles per hexagon, 22 cm axial mesh and quadratic polynomial in radial and axial directions were used. For FDM 54 triangles per hexagon and 11 cm axial mesh was considered. Partial control was present in one assembly. The assembly power distribution of the above reference solutions were used to tune albedoes and adjust $\Delta K/K$ for control in AKHILESH. No feedback effects were considered in the above calculations. The assembly numbers and the power distributions in 30^0 symmetric part of the core are given in Fig.1. The deviations in power distribution are found to be ≈2% which is deemed as reasonable.

The estimated critical boron concentration (C_B) and the radial peak power factor (RPPF) at the actual operating conditions as a function of cycle burnup are given in Tables-I to VI for cycle 1 to cycle 6. Tables-VII to XII give the assemblywise power distribution for the six cycles. It was observed that the power distribution calculated with six triangles per hexagon is significantly different from those reported earlier with one mesh per assembly. Since the new code has been tested against FEM and fine mesh FDM these power distributions are expected to be closer to reality.

CONCLUSIONS

The one group 2-D AKHILESH code has been used for the prediction of critical boron and assembly power distribution for six fuel cycles of VVER-1000 MWe reactor benchmark. Use of six triangles per hexagon has improved the accuracy of power distribution substantially with respect to reference evaluations. The calculational time is of the order of a few minutes only for entire cycle. Thus AKHILESH code can be used successfully as a level-2 calculational tool for survey

type analyses. A 3-D model is necessary for better estimate of radial as well as axial peak factors. Two group nodal techniques could be developed for more accurate (level-3) followup simulations.

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THE CORFU PROGRAM FOR BWR RELOAD DESIGN

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Abstract

A program CORFU intended to serve as a BWR core reload designer's assistant is being developed at the Technical Research Centre of Finland. The program creates a preliminary loading scheme, suggests fuel bundle moves and control rod patterns in order to optimize the loading and makes some final adjustments to an already acceptable loading pattern.

1 INTRODUCTION

The problem of finding a new core configuration for a power reactor at the end of a cycle of operation has an almost innumerable number of (at least theoretically) possible solutions. Even if most of these solutions for one reason or another can be disregarded immediately, the reload planning still usually involves extensive analysis and comparison of many different alternatives. The core of a BWR typically contains some 400 - 800 fuel assemblies, of which 20 - 40 % will have to be replaced by fresh fuel at refuellings taking place at intervals of 1, 1.5 or even 2 years. At the same time, the core is thoroughly rearranged, i. e. most of the remaining assemblies are usually moved to new locations in order to obtain an optimum loading pattern. Previously burnt fuel assemblies, which have been taken out of the reactor earlier, may also be reinserted into the core. For a boiling water reactor, the control rod patterns to be used during the cycle of operation are closely connected with the reloading scheme and they should also be optimized at the same time as this scheme is developed. The final goal is to find a core configuration and control rod patterns that allow the reactor to stay critical at full power for as long as possible using a certain amount of fresh fuel while

satisfying any existing thermal and safety restrictions concerning for instance heat flux, dryout and shutdown margins.

2 THE CORFU PROGRAM

The core designer's work is largely a trial and error operation, where a number of loading patterns have to be studied more or less thoroughly using a core simulator program as the main tool to determine how the burnup cycle would turn out, if a particular core design plan is realized. The search for improvements is based on the results of these and other calculations. This process might

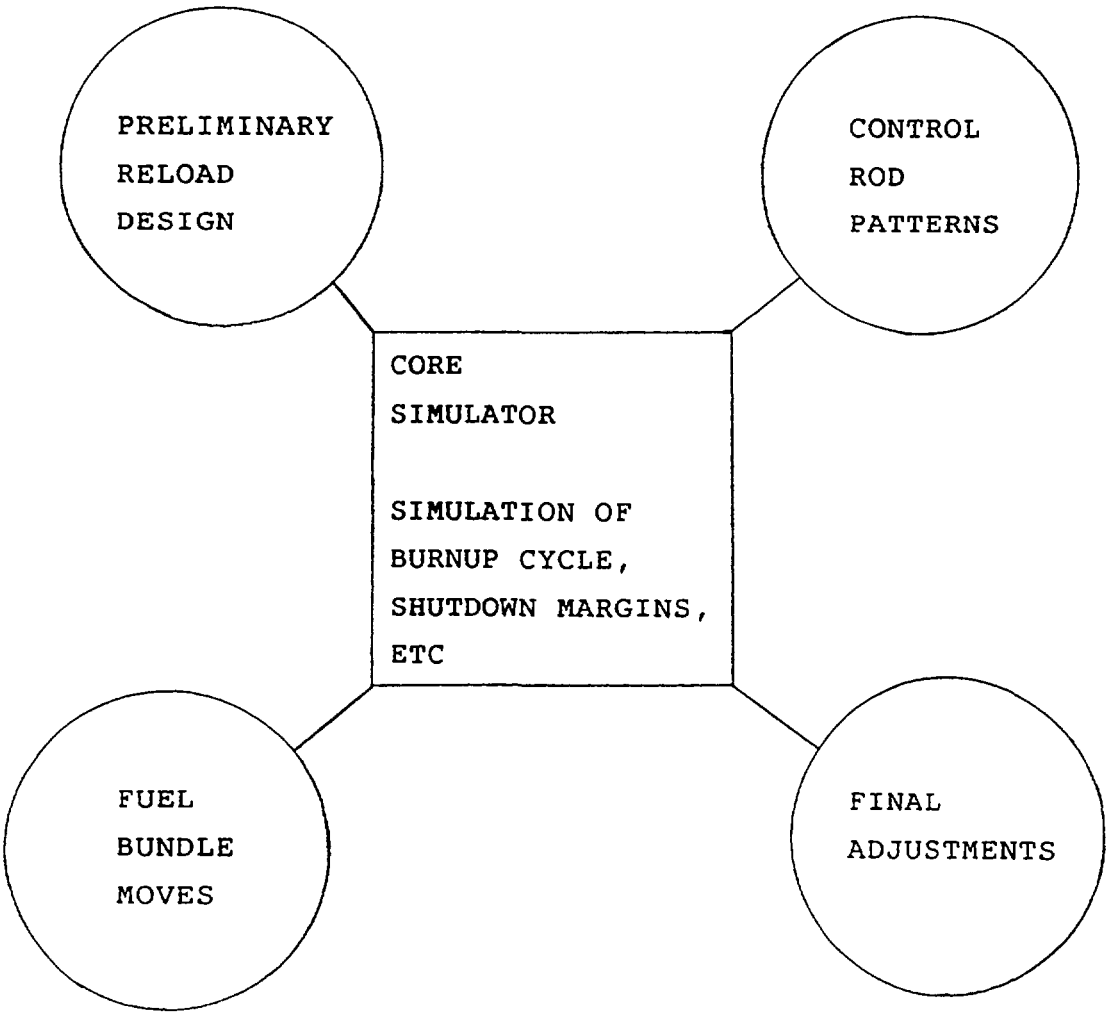


Figure 1. The main modules of CORFU.

be considerably more efficient if the core designer were assisted by an appropriate computer program, which would make it possible to achieve better results in the same time or comparable results in a shorter time than through purely "manual" planning. Such an assistant, called CORFU (from COre ReFUelling) is being developed at the Nuclear Engineering Laboratory of the Technical Research Centre of Finland.

The main modules of the program are shown in Figure 1. They perform the following tasks:

a) to create a preliminary loading pattern to be used as a starting-point for the subsequent work,
b) to suggest suitable control rod patterns at different burnup levels during the cycle of operation,
c) to suggest fuel bundle moves in order to improve the loading scheme being studied, and
d) to make some final adjustments.

These main modules will be briefly discussed in the following sections of the paper.

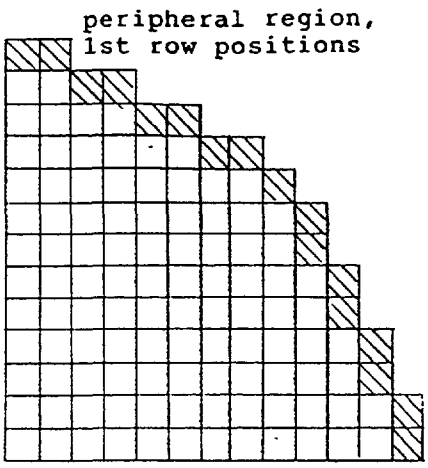
3 PRELIMINARY RELOAD DESIGN

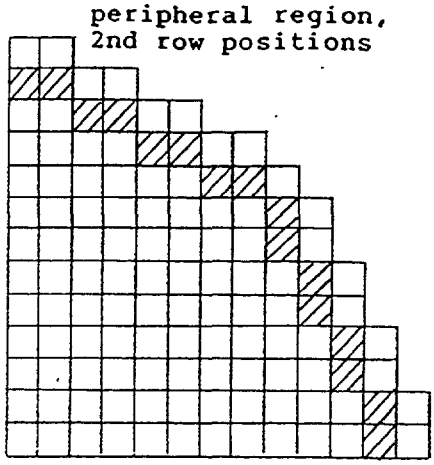
It is, of course, not forbidden for the core designer to find the best possible loading pattern in the very first attempt. In practice, this is not very likely to happen. Nevertheless, the first preliminary scheme that is used as a starting-point for the planning should, of course, preferably be as good as possible. Most of the positions to be used for fresh and once-burnt assemblies can be chosen beforehand, and later studies will probably cause only minor changes in this respect. If there are fresh fuel bundles of different enrichments and burnable absorber contents, one might find out that some of them are better suited for certain locations than the other ones, but this doesn't affect the fresh positions, just the distribution of fresh fuel types amongst them. Once-burnt bundles showing different average burnups and reactivities or only different

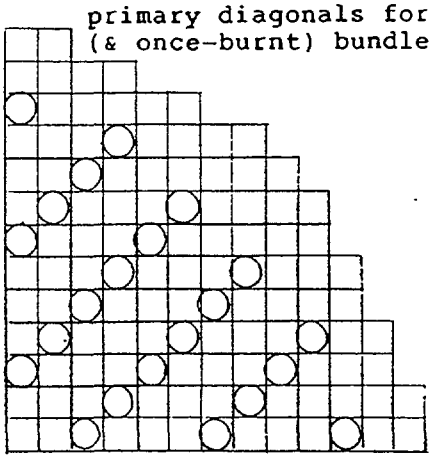
axial burnup distributions may also have to be interchanged within the group of positions chosen for such bundles. The core boundary, which in this case means the two outermost "rows" of bundles, is built up at this early stage according to the low leakage principle. A third group of positions that require special treatment are the ones that belong to control rod modules, deep (for adjusting reactivity) or shallow (for forming the axial power distribution). Fresh fuel is not loaded in the control cells, i. e. immediately adjacent to control rods that are going to be used during the coming cycle of operation. This is important in order to avoid a distorted burnup distribution in the fresh fuel bundles with their burnable absorber rods and high power peaks above the top of the control rod as it is finally withdrawn towards the end of the cycle. In cells that contain a control rod inserted deep in the core for a rather long time, not only fresh but also once-burnt and maybe even older high-reactivity bundles should be avoided. For any other "non-special" location it is initially only required that unnecessary bundle moves should be avoided and necessary moves might just as well be as short as possible.

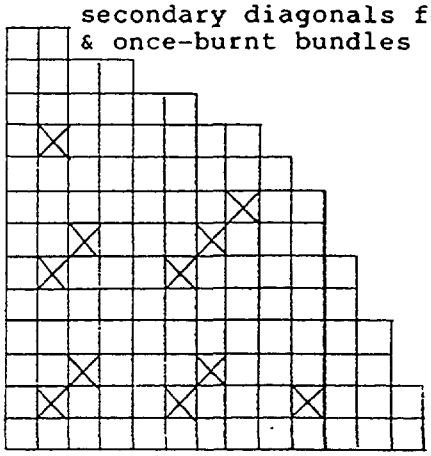
The program uses fixed sets of locations, based upon the loading principle (different variants of single or double diagonal loading etc), for the fresh fuel bundles. These sets naturally depend on the core geometry, but they have to be evaluated only once for a given reactor. "Fresh positions" not needed for fresh fuel are used as once-burnt locations instead and the rest of the once-burnt bundles are placed in other suitable positions, preferably on "free diagonals", i. e. not close to fresh fuel, but also as fresh bundle neighbours in areas not too far away from the core boundary or control rods in use. The more fresh and once-burnt fuel assemblies there are in the reactor, the more difficult will it often be to find good positions for them all.

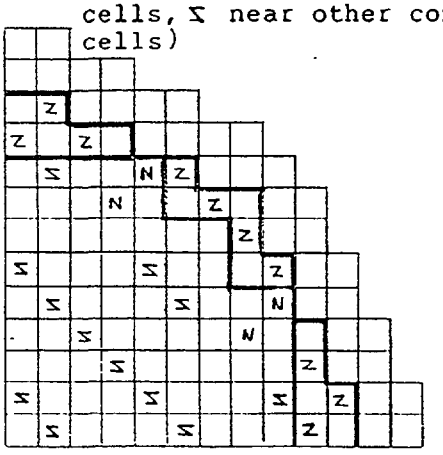
An example of how the core can be divided into different categories of positions (from the reload designer's point of view) is shown in Figure 2.

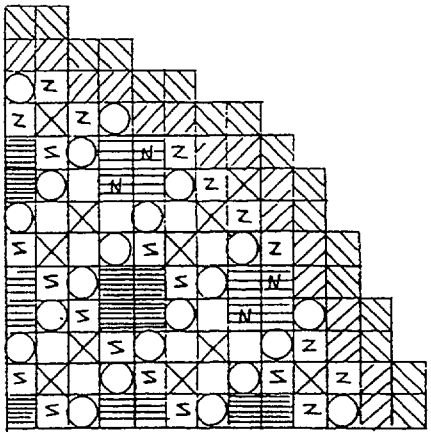












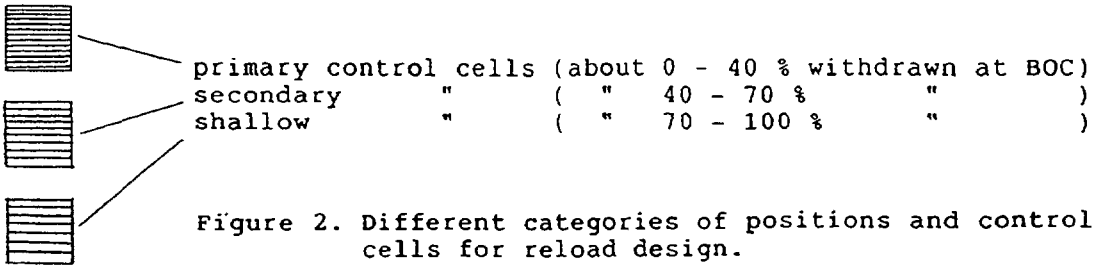


Figure 2. Different categories of positions and control cells for reload design.

4 CONTROL ROD PATTERNS

The control rods to be used during the cycle of operation are selected in advance. Thus the program immediately knows the locations of both the deep and the shallow rods, and it's only the precise insertions of these rods (or rather rod groups) that have to be determined. A test calculation has to be performed at each burnup level using some plausible control rod patterns, which then are corrected by the program. In the test calculation, the deep rods might be completely inserted into and the shallow rods completely withdrawn from the core, but the results will probably be better if a more accurate initial guess can be made.

The control rods have two important functions, i. e. to make it possible to reach the right k_{eff} value (theoretically = 1, but for a certain data-code combination usually differing from 1 by a small, hopefully known, amount, which can vary with burnup) and to shape the power distribution. For the power distribution to be acceptable, the thermal margins have to be large enough, e. g. the surface heat flux or the linear heat generation rate must not exceed a certain limit and the critical power ratio must not fall below another limit value. This is still not enough, however; in order to achieve a good neutron economy, the power should be kept relatively high in the lower part of the core for as long as possible during the cycle, which requires good separation between the deep and shallow rods. As a rule of thumb, it can be said that the former should be as much and the latter as little inserted as possible. A high void fraction during operation will increase the reactivity of the fuel at higher burnups and burning the core in the lower regions will lift the power distributions upwards near EOC, where, on the contrary, a low void fraction is advantageous.

It is generally not very difficult to reach the correct k_{eff}, as the dependence of the multiplication factor on the insertion level of a control rod is fairly well-known. Thermal margins based upon heat flux, dryout margin etc

cannot always be that easily corrected. Nevertheless, the program makes an attempt to improve the initially guessed control rod pattern in this respect also, using knowledge of the influence exercised by a control rod on the power distribution in the core as a whole and especially in the neighbouring regions. If it proves impossible to find a control rod pattern that is good enough, the loading scheme must be changed, and if even the desired k_{eff} is difficult to achieve, then the amount of fresh fuel will probably have to be adjusted.

5 MOVES FOR IMPROVEMENT

After the results of a simulation for a certain loading scheme have been computed, the core positions with their allocated fuel bundles are arranged starting with the one having the smallest overall margin, i. e. the smallest dryout or shutdown margin or the largest surface heat flux at some occasion during the cycle. Other quantities than the three ones mentioned may of course be used, if they are of greater interest to the core designer. The different quantities are compared using coefficients that have to be given to the program. Thus the importance of a certain quantity of interest may be enhanced through a suitable choice of coefficients. The "margin" for a particular position n is determined by an expression of the following type:

$$[\text{Smallest Margin}]_n = \min_E \min \{c_{SHF}(SHF_{limit} - SHF_n(E)), c_{CPR}(CPR_n(E) - CPR_{limit}), c_{SDM}(SDM_n(E) - SDM_{limit})\}, \quad (1)$$

where SHF stands for surface heat flux, CPR for critical power ratio, SDM for shutdown margin and E for burnup. c_{SHF}, c_{CPR} and c_{SDM} are the coefficients to be provided by the user, and SHF_{limit}, CPR_{limit} and SDM_{limit} the limits that have to be satisfied by the loading.

The program tries to find more suitable bundles for each position (and its nearest neighbours) in turn, starting from

the one with the smallest margin as defined by Equation (1), until positions where the margin is already deemed good enough are reached. If the position being investigated is occupied by a fresh bundle, which is of course often the case, it can be improved directly only if there are fresh bundles with a larger amount of burnable absorber or lower enrichment available. Otherwise it will have to be enough to exchange the neighbour bundles for fuel of lower reactivity / higher burnup at the appropriate axial level. On the other hand, these moves mustn't be likely to reduce the margin at any other location in the core below the acceptable level; such moves will not be suggested by the program.

6 FINAL ADJUSTMENTS

The module for final adjustments to the loading scheme performs, to some extent, similar tasks as the first module, the one for guessing a preliminary loading pattern. Even if the initial guess satisfies a number of more or less important conditions, these conditions may again have been violated through the bundle moves carried out during the search for an optimum core design.

The most important thing is, of course, to obtain a burnup cycle of the requested length with sufficiently good thermal and other margins everywhere. The final adjustment module is not activated until such a core design has been found. The program then makes a number of checks, e. g. to make sure that the core boundary is as low-leaking as possible, that the burnup of bundles placed in control rod modules is sufficiently high and the control rod history sufficiently low (i. e. that these bundles have not previously been tortured in the vicinity of a control rod) and that any existing maximum burnup limits will not be exceeded anywhere in the core. CORFU also makes a comparison between the burnup or reactivity distributions of the old core at EOC and the new core at BOC to decide whether some of the intended bundle moves can be avoided altogether.

PARAMETRISATION FOR OPTIMISATION OF RELOAD PATTERNS FOR BOILING WATER REACTORS

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Abstract

Parametrisation of reload patterns for BWRs is attempted for the purpose of optimising them. This is done in two stages. The first stage involves an algorithm for constitution of a reload pattern out of a given set of fuel bundles. It is designed to constitute patterns with features of Low Leakage Loading (LLL) and checker board arrangement of fresh and exposed fuel bundles in the central region. The characteristics of the Reload Patterns can be manipulated by means of only two input parameters. The idea of the dependence the Haling power peaking and the cycle energy of the "Biparametric Reload Patterns (BRP)" on the two input parameters is presented through case studies. The idea of the ranges of these characteristic quantities and their mutual relation is given by the case studies both for general as well as the optimum BRPs. A priliminary optimisation procedure for biparametric reload patterns is presented. This method is augmented by a multi-parametric algorithm to reshuffling the radial exposure distribution interactively so as to exhaust any possibility of improvement in a given reload pattern. When tested against this procedure, the optimum biparametric reload patterns are seen to provide only a limited scope for improvement showing that they are very close to the optimum reload patterns. The extent of possible improvement is illustrated. The entire procedure is incorporated in a 2-dimentional code CORECOOK. The computor time needed for this task of optimisation of reload pattern is comparable to that required for one routine 3-D core followup calculation. This algorithm has been used for obtaining the reload patterns for all the five cores loaded in BWRs at Tarapur (India) since 1988.

1. INTRODUCTION

Design of a reload pattern involves decisions regarding assignments of fuel bundles to locations for fuel bundles in a core. A reload pattern for a reactor core which has n fuel locations involves n such decisions. Thus there are n decision variables for a reload pattern. The problem of optimisation of a reload pattern for such a core involves taking these many decisions so as to extremise a chosen objective function under the given constraint(s).

There can be several candidates for an objective function. It could for example be, minimisation of the core enrichment or neutron leakage. Alternatively, it could also be maximisation of the cycle energy or the Keff at the assumed End Of Cycle (EOC) or fuel discharge burnup. One may note that these extremisations of the respective objective functions are consistent with each other. All of them cater to the efficient fuel utilisation and reduction in the neutron fluence on the reactor vessal. Thus Suzuki and Kiyose (1971) have determined that minimum core enrichment can be obtained for the reload core configuration by maximising the EOC core Keff. Huang and Levine (1978) also have shown that "if the mechanical design of the fuel assembly remains fixed, the minimum core enrichment can be obtained for the reload core configuration by maximizing the EOC core Keff." It is very clear that higher value of Keff at a given core burnup, higher cycle energy, higher fuel discharge burnup and better fuel utilisation are all mutually consistent. The Low Leakage Loading (LLL) concept aimed at minimisation of neutron leakage is consistent with the maximisation of cycle energy (Downar and Sesonske, 1988). Thus the concepts discussed above viz. lower enrichments, higher cycle energy, higher fuel discharge burnup, higher fuel utilisation and LLL concept are all mutually consistant. According to Downar (1986) LLL also provides accelerated fuel depreciation and quick returns on the investment on fuel.

The Haling power peaking is another important quantity which is a function of a given reload pattern. Its value is required to be low in the interest of the fuel integrity in steady state operation. This requirement, however, is inconsistent with the mutually consistent functions and their

requirements considered previously. Thus we have two mutually inconsistant requirements. Therefore, one of the two has to be optimised while the other is to be constrained. Maximisation is consistent with applying a lower bound and vice versa. Thus one may maximise cycle energy or put a lower bound on it if one wants to constrain it. Similarly one may either minimise Haling power peaking or put an upper limit on it.

Given a set of values for the decision variables, a calculational model is required to evaluate the objective function as well as the functions to be constrained. In the case of a reload pattern we require the knowledge of the nuclear properties of each of the n fuel bundles and a model to calculate these functions. A Haling calculation is the model used for evaluating the cycle energy and the Haling power peaking for the given reload pattern.

If we assume that each of the n fuel bundles is different then we have n! possible reload patterns. If we assume that the all the fresh bundles are identical then this number reduces to n!/r! where r is the number of the fresh bundles. This is still enormously large a number. It is impracticable to constitute and evaluate so many reload patterns before selecting the optimum one out of them. The solution of this problem calls for a three-pronged approach viz. the use of standard optimisation methods, use of fast models for performing Haling calculations and reduction in the number of the decision variables.

The major portion of work put in solving this problem so far has been put in the use of optimisation methods. They involve heuristic methods as well as Non-linear programming and linear programming methods. Non-linear programming usually involves the use of second derivatives of the system equations along with some sort of searching technique. In this approach the original system and the constraint equations are preserved, but are solved by using algorithms that are much less efficient and proven than those for linear programming. Hoshino(1978) applied a non-linear programming method to the multi-cycle optimisation problem with apparent success (Downar and Sesonske, 1988). Linear programming uses first derivatives relating the decision variable to the optimisation function to improve the next decision. This is obtained from the results

of previous iterations. The principal problem encountered with linear programming is the inability to distinguish a local from the global optimum since the algorithm employs only the first derivatives of the objective functions. Therefore different starting points for the algorithm will sometimes result in different optimum solutions. Suzuki and Kiyose(1971) applied linear programming to determine the optimum refuling schemes for light water reactors described by very simple core models. The optimisation methods employing simulated annealing techniques significantly reduce the probability of getting trapped in a local minima and free one to start from any initial configuration. Kropaczek and Turinsky (1991) combine this optimisation technique with computationally efficient core physics model based on second-order accurate generalised perturbation theory. Perturbation theory is also used by Mingle (1975). Dynamic programming reduces an N dimensional problem to N one dimensional problems. It uses the "principal of optimality" (Bellman,1957) to guide the search process by noting that whatever the initial decision, the resulting decisions must constitute an optimal policy with regard to the state resulting from the initial decision. This technique was successfully used by Civita and Fornaciari (1972) for the multi-cycle optimization problem. Stout and Robinson (1973) also have used dynamic programming for this purpose. Variational methods are used to obtain the derivatives more easily. Integer programming methods are used since the decisions involved are regarding the assignment of fuel bundles to locations and both of them can be represented by integers.

Apart from how one constitutes the successive reload patterns, faster models for evaluating them are desirable. This can be achieved by performing Haling calculations in 2 instead of the 3-dimensions and also by using the symmetries of the reload patterns if any.

The number of possible reload patterns and the complexity of the problem increases rapidly with the number of the decision variables. Therefore reduction in this number is highly desirable. This makes application of the optimisation methods to the problem practical. The reduction in the number of decision variables has depended upon the scope of the fuel

management problem to be solved. Many authers have chosen to study less ambitious and simpler problems, for example, only a two or three region problem in optimisation of reload pattern (Downar,1987; Suzuki A. and Kiyose R., 1971; Suzuki A. and Kiyose R., 1971; Ahn D. H. and Levine S. H., 1985 Matoda H., 1971; Sauar T., 1971). This could be a study in itself to gain insight into the problem. It could also be the first of a multi-stage process to tackle the actual problem. The size of the problem of constituting one reload pattern can also be reduced by deciding to follow some kind of symmetry in the core. Thus in the traditionally followed Quadrant Symmetric Cores (QSC), the number of fuel locations reduces by a factor of four. In the Sector Symmetric Cores (SSC) suggested by Burte and Vaidya (1986) a "sector" contains more than one fourth the number of bundles in the core because here the fuel bundles appear on the axes of symmetry and they are counted in both the sectors sharing a symmetry axis. It is enough to solve the problem of reload pattern for such a symmetric part say a "quadrant" or a "sector" of the core. This solution can easily be expanded to the solution for the full core by the use of the assumed symmetry.

One can refer to many other works reported in this field. The notable among them are Motoda (1975), Wall and Fenech (1969), M. Melice (1969), Suh and Levine (1990), Morita et al (1986), Kim et al (1987), Ho and Sesonske (1982) A. Galperin and Y. Kimhy (1991). The problem is still considered to be a challenge (White and Avila 1990). Tahara et al, have as late as in 1991, proposed a "computor aided system for generating fuel shuffling configurations based on knowledge engineering". According to them "the conventional way of generating an appropriate fuel loading pattern is a method of trial and error, which is laborious and requires much computor time." Downar and Sesonske (1988) have the following to say in their review article in regard to recent methods to optimise the reload patterns which are recently modified by the use of Low Leakage Loading strategies and also of the use of burnable poison. "While overcoming some of the shortcomings of previous techniques, these methods are still being tested and have yet to receive widespread acceptance for actual core reload design."

2. THE APPROACH

The emphasis here is on reducing the decision variables without compromising the complexities of problem of optimisation. We attempt to achieve this by incorporating the traditional guidelines.

2.1. GUIDELINES ON OPTIMISATION

Normally the locations for fuel bundles are arranged periodically with constant pitch. This means that the fuel bundles are uniformly distributed over the core. in other words "fuel density" is uniform all over the core.

The optimisation then is to be achieved by exposure distribution. The relevant guidelines are in the form of the out-in and in-out schemes. They refer to the direction in which the exposures increase as one goes inwards ie. away from the periphery or outwards ie. as one goes towards the periphery respectively. The former generally leads to power flattening at the expense of cycle energy and the latter, vice versa. An application of a combination of these two guidelines respectively in the central (or non-peripheral) and peripheral region leads to what is called the Low Leakage Loading (LLL) guideline. Power density in the peripheral region is expected to be quite low in view of the leakage of neutrons outside the core. In the non-peripheral region however, it tends to peak. LLL is aimed at flattening the power where it tends to peak by increasing the exposure towards the centre and it is aimed instead, at reducing the leakage of neutrons outside and thus contribute to cycle energy by increasing the exposures towards the periphery.

2.2. CASE OF TWO CATAGORIES OF BUNDLES-CHECKER BOARD

Guidelines are always to be used with discrimination. The guidelines on the radial distribution of fuel bundles in the form of the in-out, out-in schemes are no exception. Loading all the fresh bundles whether in the central or the peripheral

region depending upon the in-out or the out-in scheme may lead to undesirably large power peaking or large neutron leakage. As a part of such a discrimination it is advisable to catagorise the fuel bundles suitably depending upon their important nuclear properties and intersperse them with each other to soften the effects of these properties. Thus the power densities of bundles from the catagory with higher reactivity will be reduced by loading fuel bundles from the catagory with lower reactivity near them. The power densities in the latter ones will be increased due to the former bundles. This is desirable.

We have seen, irrespective of the catagorisation of the fuel bundles, that they are distributed uniformly with constant pitch. It is desirable to apply similar consideration to distribution to the fuel bundles of each catagory separately also.

The nuclear properties mainly depend upon the fuel exposure apart from their design. It is common to catagorise the fuel bundles as per the number of cycles they have seen. However, the group of fresh bundles stands clearly apart on the exposure histogram of the fuel bundles to be loaded in a core. Therefore, the division of the fuel bundles into the catagories of the fresh and the exposed fuel bundles is most justified. We can recognise that the traditionally recommended checker board arrangement of fresh and exposed fuel bundles indeed distributes the bundles of each of the two catagories as desired above.

The area over which checker board arrangement is possible is determined by the size of the smaller group of bundles, which normally is that of fresh bundles. It is posible to cover a larger area of the core by checker board arrangement if fewer batch refueling is followed.

The LLL philosophy requires the fresh bundles to be loaded away from the periphery of the core. This means that the checker board arrangement should be employed in the centralmost region of the core.

2.3. CYLINDRICAL CORES AND AZIMUTHAL SYMMETRY

Reactor cores are generally cylindrical. Though the problem of reload pattern appears to be a three dimensional one there is no freedom to shuffle the fuel axially within a fuel bundle. Therefore the z-axis cannot be considered in this problem. It is only the radial and not the azimuthal dimension which is relevent to the exposure distribution schemes. The radial distribution of fuel properties has to be equally good for each value of the azimuthal co-ordinate and hence has to be azimuthally symmetric.

Strict azimuthal symmetry is possible only if the reactor is a perfect cylinder and can be divided into annular cylinders of fuel with uniform properties. It also requires that it should be possible to manipulate the properties of these annular cylinders as per the requirements of the optimisation. This is not possible. At best one can hope to design reload patterns in such a way that the relevant quantities are indipendent of the azimuthal angle.

Secondly, it is only approximately that the reactors can be considered to be cylindrical. Therefore, we have to consider the importance of a particular location which may depend not only upon its distance from the center but also on its distance from the periphery. Therefore we have to sequence the fuel locations as a function of this importance rather than as a function of radius. In fact, we suggest that it can be determined by calculating the power distribution in the core which uses identical fuel bundles. Though strictly we may use this importance of a location instead of radius, for the convenience of discussion we will take freedom to use phrases like radial distribution.

2.3.1. QUADRANT SYMMETRIC CORE (QSC)

The traditional reload patterns settle only for the next best to the azimuthal symmetry ie. a quarter core symmetry. The traditionally chosen axes of reflective symmetry run parallel to the blades of the cruciform control rods. This symmetry may be called "Quadrant Symmetry" and the core loaded

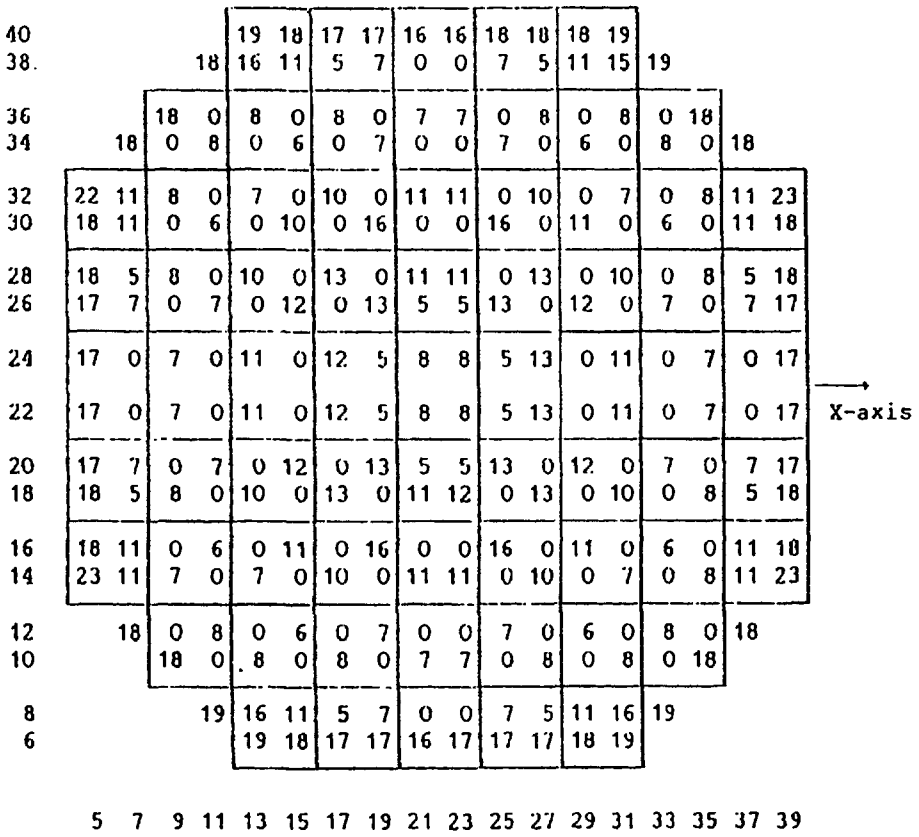

FIG. 1a. Exposure (GWd/STU) map of a QSC reload pattern (unit 1 BOC-10 of TAPS).

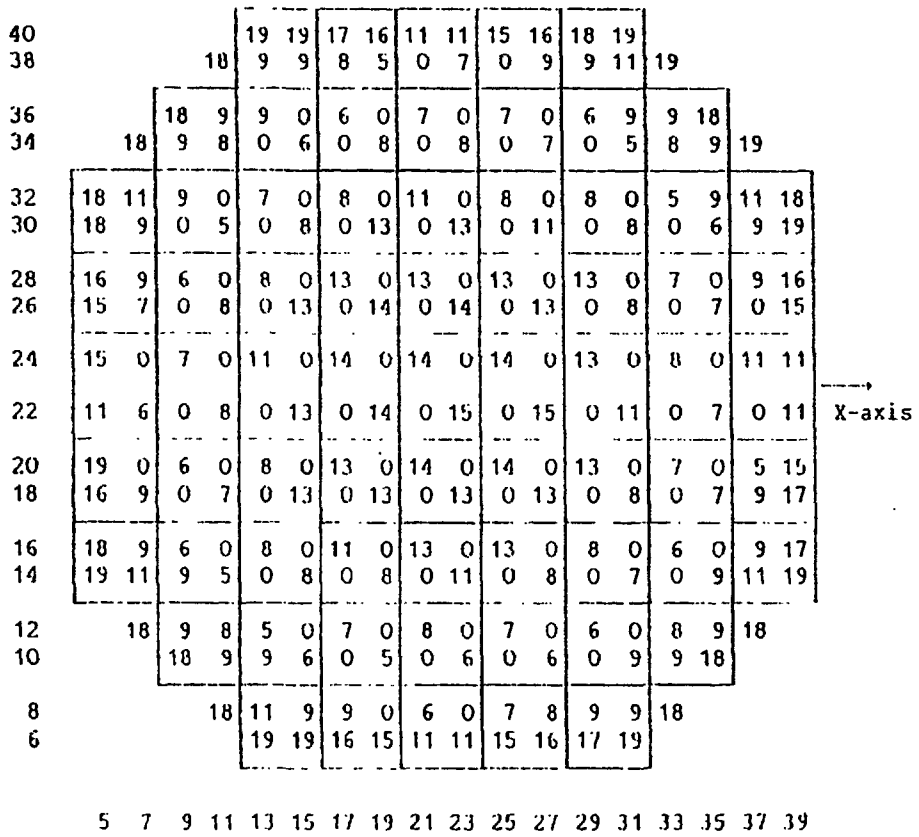

FIG. 1b. Exposure (GWd/STU) map of an SSC reload pattern (unit 2 BOC-11 of TAPS).

following this symmetry the "Quadrant Symmetric Core" (QSC). Figure-1a illustrates QSC.

Many BWRs including those at TAPS have a control rod at the center of the core. They have even number of rows and columns of fuel bundles. In this type of reactors the axes of symmetry as per the QSC pass in between the rows and columns of fuel bundles. This has two effects. If the neighbouring bundles are symmetrics of each other, their properties are similar; otherwise they can be different. Thus even for nearly same radial distance the properties of neighbouring fuel bundles can be similar or dissimilar. This brings the azimuthal co-ordinate into picture; the possibility of reduction in the number of decision variables on account of azimuthal symmetry is therefore ruled out. Secondly,it causes violation of the checker board arrangement of fresh and exposed fuel bundles in this type of cores.

We find that the symmetric locations in the region which is closer to the centre of the core are more close to each other. Therefore, loading a fresh bundle closer to the centre brings fresh bundles closer to each other in the central region. This leads to increase in the Haling power peaking. Therefore fresh bundles cannot be loaded in the central region. They cannot be loaded close to the periphery which will increase the leakage of neutrons and lead to reduction in the cycle energy. Therefore they have to be loaded only in the ring-like annular region. Thus instead of dividing the core into only the peripheral and the non-peripheral regions such a core has to be divided into the inner, outer and annular regions. This also increases number of decision variables.

2.3.2. SECTOR SYMMETRIC CORE (SSC)

Sector Symmetric Core (SSC) concept, illustrated in Figure-1b, involves a new set of axes of reflective symmetry for loading the fuel. These axes are obtained by rotating the axes of symmetry of reflection used in QSC through 45^0. Now these axes pass through bundles (diagonally) rather than between pairs of bundles as in quandrant symmetry. The bundles

through which the axes of symmetry pass become their own symmetrics. The reflective symmetry with this choice of the axes may be called sector symmetry. The core following this symmetry may be called Sector Symmetric Core (SSC) for convenience.

SSC was introduced for a different purpose (Burte and Vaidya 1986, Burte 1990) viz. for the mitigation of the problem of large reactivity worths of control rods. It was also shown that SSC is either superior or equivalent to QSC in other respects. Its characteristics also are useful for our present purpose. SSC is seen to accommodate checker board arrangement of fresh and exposed fuel bundles regorously. There is no dependence on azimuthal co-ordinate. The number of regions can be only two, the peripheral and the non-peripheral. Thus SSC satisfies azimuthal symmetry as well as uniform distribution of the pitch of the fuel bundles of each catagory in each of the two regions. This simplicity helps to reduce the number of decision variables for constitution of SSC and LLL reload patterns.

3. BIPARAMETRIC RELOAD PATTERNS

Our approach to constitution of reload patterns is based upon the traditional insistance for good reasons on checkerboard arrangement of fresh and exposed fuel bundles and optimisation based upon the traditional guidelines like LLL on the radial distribution of the fuel exposures. SSC concept is chosen for this approach since it suits the purpose well. It turns out that this approach helps in reduction in the number of decision variables, to mere two.

The full TAPS core contains 284 fuel locations and its sector contains 78 ones. We will consider the problem of assigning only the 78 fuel bundles which are properly chosen out of the given 284 fuel bundles to the 78 locations as far as the constitution is considered and assume that we have the algorithm to work out the full sector symmetric core reload pattern using the available 284 fuel bundles. The approach is not restricted only to the TAPS cores.

Application of the LLL scheme requires the delineation of the boundary between the peripheral and the non-peripheral region of the core. The extent of the checkerboard arrangement of fresh and exposed fuel bundles in the central-most region is an important consideration regarding the boundary between the peripheral and non-peripheral regions. In fact for a three batch refueling of TAPS reactors we have chosen the checker board region itself as the non-peripheral region. We may also note that the checkerboard arrangement decides the positions of the fresh bundles except the trivial decision whether the fresh ones go into white or black locations.

Having thus decided the boundary of the non-peripheral region and the positions of the fresh bundles what remains to be decided is the distribution of the exposed fuel bundles. The philosophy of LLL implies only qualitatively that the exposure distribution will be a vally along a boundary and the exposures will rise as we move away from this boundary either towards the centre or towards the periphery of the core. In this section we will first describe a procedure to constitute a reload pattern based only on these guidelines. Then we will turn to reducing the remaining number decision variables in fact only to two. The next task is to use these two parameters to find an optimum biparametric reload pattern.

3.1. SSC MAPPING PROCEDURE

We will first describe our procedure to map the fuel bundles on the fuel locations. For this purpose, the n fuel bundles are arranged in an array. On the other hand, importance numbers are assigned to the n fuel locations of the core. The importance numbers normally increase as we go to radially more distant fuel locations. The core is divided into two regions, the peripheral and the non-peripheral. The non-peripheral region comprises of a given number of the fuel locations of highest importance while the peripheral region comprises of the remaining locations of least importance. Let L represent the number of the fuel locations in the non-peripheral region and n-L, that in the peripheral region.

We intend to follow checkerboard arrangement of fuel loading in the non-peripheral region. Therefore we catagorise the fuel locations in this region into black and white catagories as in a checker board. We assume that the first L/2 of the bundles belong to one catagory and the subsequent L/2 bundles belong to another catagory. The bundles of these catagories are to be arranged according to the chacker board arrangement in the non-peripheral region. For completeness of description let us call the catagory of the fuel locations in the peripheral region as the gray catagory.

Now we introduce a procedure to assign these bundles refered by their sequence numbers in the array to the fuel locations refered by their importance numbers. We take the fuel bundles sequentially from their array and assign them to the fuel locations as follows. We assign all the first L/2 bundles to the locations belonging to, say, the white catagory starting with the locations of highest importance and proceeding to those of lower importance. We assign the next L/2 fuel bundles to the fuel locations belonging to the black catagory starting from the location with the least importance and proceeding towards those of higher importance. We assign sequentially the remaining n-L fuel bundles which appear last in the array of fuel to the gray locations ie. the locations in the peripheral region, starting from the locations of highest importance and proceeding to those of lower importance.

By checker board arrangement we normally mean a checker board arrangement of fresh and exposed bundles. In order that all the fresh bundles,say r in number, are covered by this checkerboard arrangement we must have L=2r or L>2r. Wetake 2r as the default value for L. We assume for our discusion that always this default value is used. The appearance of the fresh bundles in the begining of the array is convenient and suits our purpose and procedure. The division of the core into the peripheral and non-peripheral regions thus turns out to be based upon the number of fresh fuel bundles. Normally L<n. It is because of this that we can divide the core into peripheral and non-peripheral.

3.2. PARAMETERLESS OR BASIC LLL SSC RELOAD PATTERN

Given an array of the fuel bundles the above procedure yields a unique SSC reload pattern in which the fresh and exposed fuel bundles are arranged as in a checker board in the non-peripheral region and only the exposed fuel bundles are loaded in the peripheral region. There are no decision variables involved. The radial exposure distribution of the exposed bundles depends entirely upon how they are ordered in an array which is to be mapped on the fuel locations according the mapping procedure described above.

Let us define "basic array" as the ordered array of the n fuel bundles which starts with all the (r number of) fresh fuel bundles followed by the exposed fuel bundles arranged according to their increasing exposures. The mapping procedure operating upon the basic array of a given set of fuel bundles results in a unique reload pattern whose exposure distribution follows the LLL guidelines as described below. Therefore this reload pattern will be called the Basic SSC LLL or simply the "basic" reload pattern.

The fresh bundles in the basic SSC LLL pattern are loaded only in the central or non-peripheral region and they are arranged according to the checker board arrangement with the exposed fuel bundles. We note the peculiarities of this pattern with reference to their expected effects on the Haling power peaking factor and cycle energy.

(a)The non-peripheral region contains the least exposed of the fuel bundles. This is expected to enhance the power peaking in the central region.

(b)Fresh bundles are supposed to be loaded in the white locations in the non-peripheral region. This is expected to increase the Halng power peaking in the central region. However, following of the out-in scheme for assigning the exposed bundles to the black locations is expected to reduce the Haling power peaking in the central region.

(c)The leakage of neutrons in the peripheral region already reduces the power factors there. Therefore we need not apply the out-in scheme in this region. On the other hand we stand to gain some cycle energy by applying the in-out scheme

in the peripheral region. Therefore the exposed bundles are arranged with their exposures increasing outward.

The exposures of the consecutive fuel bundles in the basic array vary by small amounts. This leads to rather a graded radial variation except at the boundary between the peripheral and non-peripheral region.

The observations (a) and (c) show the in-out character. On the other hand the observation (b) points to the out-in character of the basic pattern. (However, if all the fresh bundles are assumed to be identicalthe observation (a) to that extent is neutral regarding the in-out or out-in character). With these balancing effects, on the whole the power peaking for the basic pattern is rather high and occurs in the center and the power factor slowly reduces outwards. The observations (b) and (c) can be considered to determine the LLL character of the unique basic pattern.

3.3. BIPARAMETRIC MODIFICATION

As mentioned above the basic SSC LLL pattern is unique with its own Haling power peaking. The probability of its being the reload pattern answering the particular requirements of the problem of the optimisation is insignificant. This obviously calls for introduction of some flexibility in the algorithm. In order to clarify our ideas about the nature of the required flexibility we note that, as mentioned above and also as per our experience, the basic SSC LLL pattern is biased towards high value for the Haling power peaking. Therefore, the flexibility in the algorithm should be in the form of the possibility of modifying the basic pattern so as to reduce the Haling power peaking to a desired extent while obtaining the optimum cycle energy for such power peaking.

We attempt to achieve such a flexibility by introducing some parameters which modifying the basic array as per a suitable procedure. These parameters, in fact, are the decision variables and their number should be kept as low as possible.

If the Haling power peaking factor exceeds the allowed value the bundle where the power peaks may not be the only one

where the limit is exceeded. There may be more bundles which cross it and a parameter has to be provided in the algorithm to specify the number such bundles. We may assume that it is the central-most region where the powers are to be reduced. Therefore the number of such bundles is related to the extent of the central region where the power factors are desired to be reduced. We also need to specify the extent to which the power factors are desired to be reduced. This specification also requires minimum one parameter. In order to reflect these two requirements we introduce only minimum required number of the decision variables. We have only two input parameters which may be refered as M and N. Hence the name biparametric.

The effect is achieved only by modifying the basic array as per the input parameters M and N as follows. Slice off a segment say S1 of the N bundles starting from the serial number L-M+1. Push upwards the segment say, S2 which comprises of the subsequent M bundles so that they are placed immediately after the position L-M in the array. Plug in the segment S1 subsequent to the segment S2 which is already pushed up as described above.

This modification affects the order in the array only of the bundles having their sequence numbers from L-M+1 to L+N both inclusive. The Figure-2 illustrates the modification.

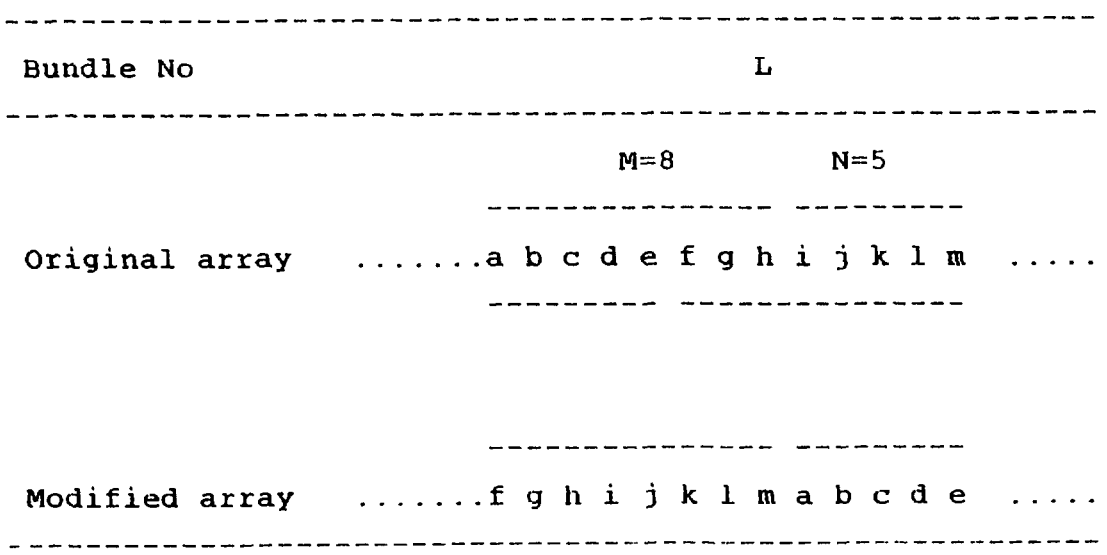

FIG. 2. Illustration of biparametric modification of the ordered array of fuel bundles (M = 8, N = 5).

Here the bundle with L as its serial number is identified as "h". The illustrative values of M and N are 8 and 5 respectively. The bundle whose serial number is L-M+1 is "a". We see that the 5 bundles a,b,c,d and e form the segment S1 which is sliced off. The next 8 bundles f, g, h, i, j, k, l and m form the segment S2. In the unmodified array S1 is followed by S2. In the modified array, S2 is followed by S1 without disturbing the internal order of these segments.

Mapping this modified array on the core according the same mapping procedure as described earlier yields a modified reload pattern. Its exposure distribution is found to be modified accordingly to the desired bias; the exposures in the central region are increased. The size of this region is governed by M and the amount of the increase in the exposures is governed by the parameter N.

We now consider the number of possible permutations of these two parameters. The two decision variables M and N in the biparametric procedure take non-negative integer values. We note here that if any one the two parameters is zero the biparametric modification leads back to the basic pattern irrespective of a non-zero value of the other parameter. Therefore the two decision variables M and N take only positive integer values for modifying the basic pattern according to the biparametric procedure. Assuming the default value 2r for L, the parameters M and N can take only r and n-2r non-zero values respectively. Thus the total number of possible biparametric reload patterns become only r(n-2r) apart from the basic pattern. Thus the biparametric algorithm reduces the number of possible reload patterns from n!/r! to only r(n-2r) apart from the basic pattern.

3.4. BIPARAMETRIC OPTIMISATION

In the problem of optimisation of reload pattern we use a given set of fuel bundles including a given number of fresh bundles and work out a reload pattern which has the maximum cycle energy under the constraint of an upper limit on the Haling radial power peaking factor.

The pattern which complies with the constraints of the problem and whose objective function is extremised may be called the solution pattern. The schemes like radial out-in, in-out etc. schemes for the exposure distribution are the dependable practical guidelines for manipulating the Radial (Haling) power Peaking Factor (RPF) and cycle energy. The biparametric procedure essencially uses them. It also allows fine-tuning the balance between these two rival schemes. Therefore the best out of the biparametric patterns is hoped to be very close to the solution pattern in satisfying the requirements of the problem.

We have seen above that the number of possible biparametric reload patterns is not enormously large. Even the brute force method appears practicable. However, it is certainly more economical to devise a search procedure to lead one to a superior pattern. As per the procedure we followed for TAPS reactors we can start with the basic SSC LLL pattern. The value of its RPF is expected to be too high to be acceptable. For reducing the same we use the maximum possible value for N. RPF is expected to decrease with M at least initially. We hope that this behavior continues till we get a reload pattern with an acceptably low RPF. We found this procedure to be satisfactory. However, exhaustive study of the behavior of the cycle energy and the RPF as functions of the two parameters M and N was desirable not only for its own sake but also for working out a better and surer optimisation method.

A two dimensional core simulation code CORECOOK written for this purpose with the biparametric algorithm introduced in it. This code uses TACHY formalism for the Haling calculations in two dimensions. The facility to calculate the cycle energy for the given value of critical Keff has been introduced in the code.

The fuel bundles of the Unit-1 at EOC-12 were used for this study. For most of the study 100 fresh fuel bundles were used though the number of fresh bundles was also varied for some part of the study. Alternate values over the entire range

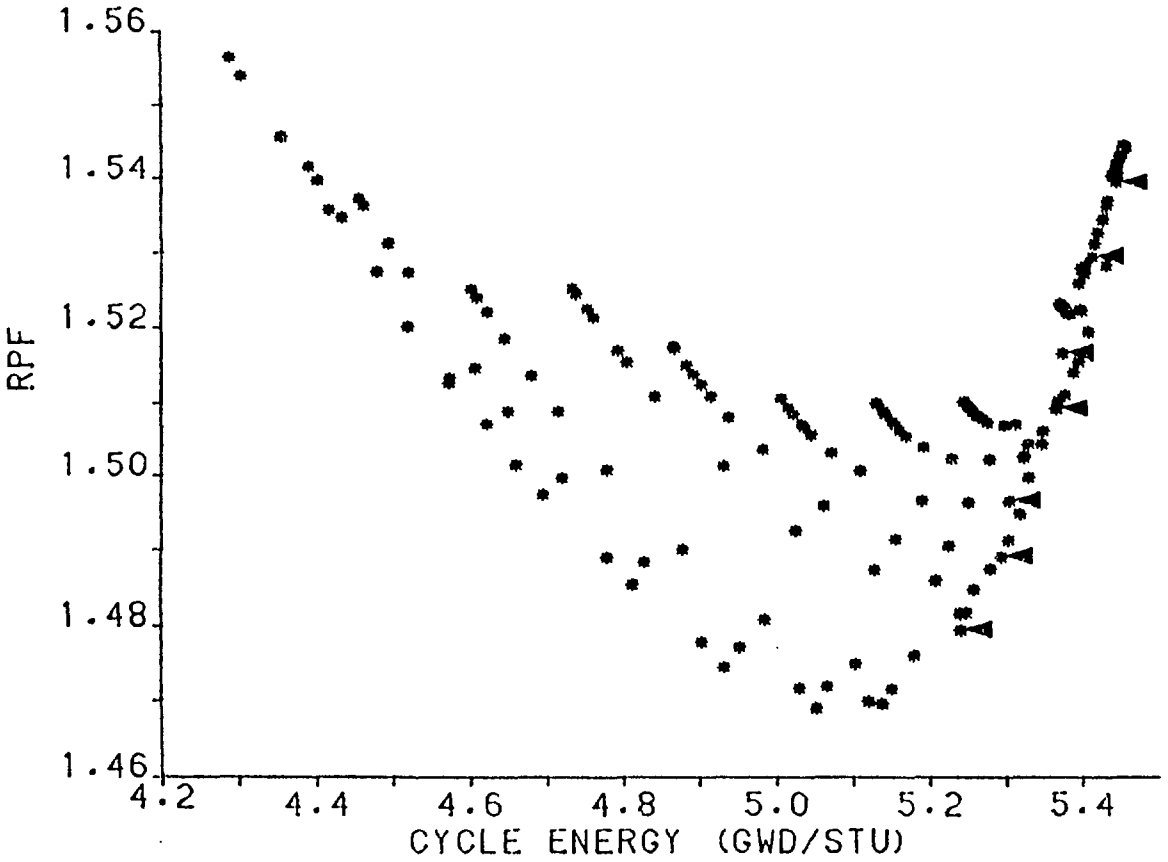

FIG. 3. Radial haling power peaking factor (RPF) versus cycle energy (each point represents a biparametric reload pattern).

of values for M (1 to r) and N (1 to n-2r) were used. Here n is the total number of bundles to be loaded in the n fuel locations of the core and r is the number of the identical fresh bundles out of them.

Figure-3 represents the relation between the RPF and the cycle energy of the biparametric reload patterns. Each point represents a biparametric reload pattern. This figure shows the shape of the entire feasible region consisting of the points representing the biparametric reload patterns using the same number of fresh bundles. It appears to be crescent-like.

The relation between the RPF and cycle energy of the biparametric reload patterns being representable by a region we have several feasible values for the cycle energy for a given value of RPF and vice versa. The point of interest is the one with largest cycle energy for a given RPF or the one

TABLE-1

BIPARAMETRIC RELOAD PATTERNS (STUDY-90)
CHARACTERISTIC RELOAD PATTERNS FOR DIFFERENT NUMBER (r) OF FRESH BUNDLES IN A CORE SECTOR CONSISTING OF 78 FRESH BUNDLES

r	RELOAD PATTERNS WITH MAXIMUM DELE,(M=N=0)		RELOAD PATTERNS WITH MINIMUM RPF				RELOAD PATTERNS WITH MINIMUM DELE M=r ,N=78-2.r	
	DELE	RPF	DELE	RPF	M	N	DELE	RPF
23	4.9992	1.7244	4.4463	1.5556	10	32	3.6703	1.6454
24	5.0938	1.6925	4.5678	1.5369	10	30	3.7711	1.6307
25	5.1767	1.6486	4.7575	1.5183	9	28	3.8833	1.6113
26	5.3327	1.5945	5.0732	1.5117	7	26	4.0091	1.5951
27	5.3883	1.5769	5.1414	1.5007	7	23	4.0601	1.5865
28	5.4549	1.5450	5.2404	1.4799	7	22	4.1780	1.5705
29	5.5625	1.5091	5.3691	1.4619	7	20	4.3200	1.5476
30	5.5961	1.4870	5.4358	1.4498	7	17	4.4486	1.5267
31	5.6560	1.4620	5.5079	1.4324	7	16	4.5811	1.5072

r : Number of fresh bundles in a sector consisting of 78 fuel bundle locations.
DELE : Cycle energy
RPF : Haling Radial Power Peaking Factor
M,N : The two decision variables of the biparametric reload patterns
ITER : Number of iterations required to converge.

with lowest RPF for a given value of cycle energy. In other words segment of interest of the envelope of this region extends from the point where RPF is minimum to the point where cycle energy is maximum. We may call the segment of the envelope as the optimum biparametric segment. It is worthwhile to compare this segment in this figure with the Figure-4 presented by Huang and Levine (1978) which shows the power peaking vs. the Keff for the various shufflings of EOC fuel. Each point in this figure represents a fuel arrangement whose serial number it bears.

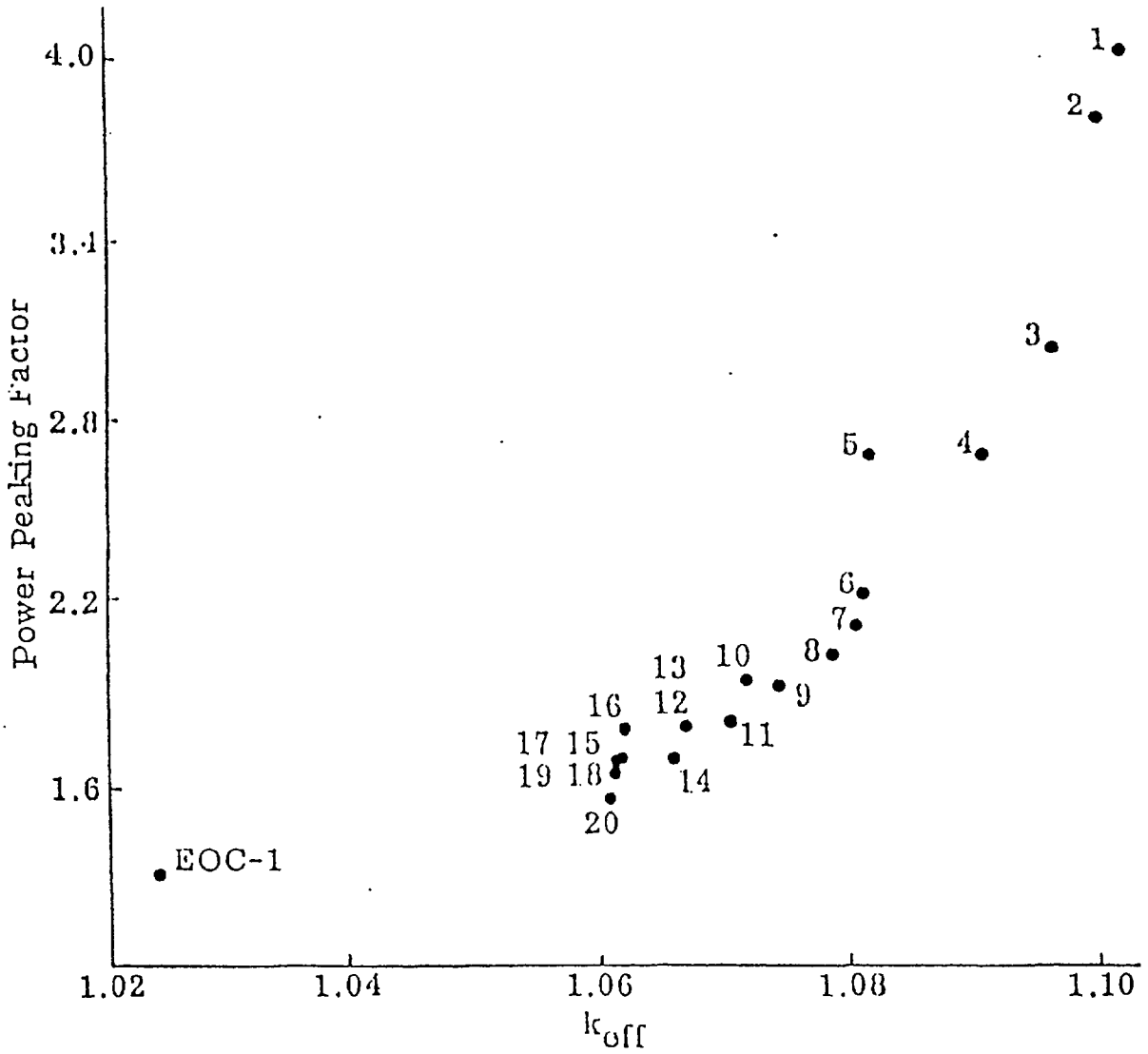

FIG. 4. Power peaking factor versus k_{eff} for different patterns of TMI-1 (Huang and Levine, 1978).

The envelope in Figure-3 continues to the left side of the point with minimum RPF. The points on this segment correspond to biparametric reload patterns with minimum possible cycles energy for a given value of RPF. This, probably, is only of academic interest. Thus for a given number fresh bundles we may consider three reload patterns as remarkable. The two of them are the ones with maximum and minimum cycle energy corresponding respectively to the basic pattern (M=N=0) and the one with maximum possible values of the two parameters viz. M=r and N=n-2r where r is the number of fresh bundles. Thus the values of the parameters for these patterns are known. The third is the one with the minimum

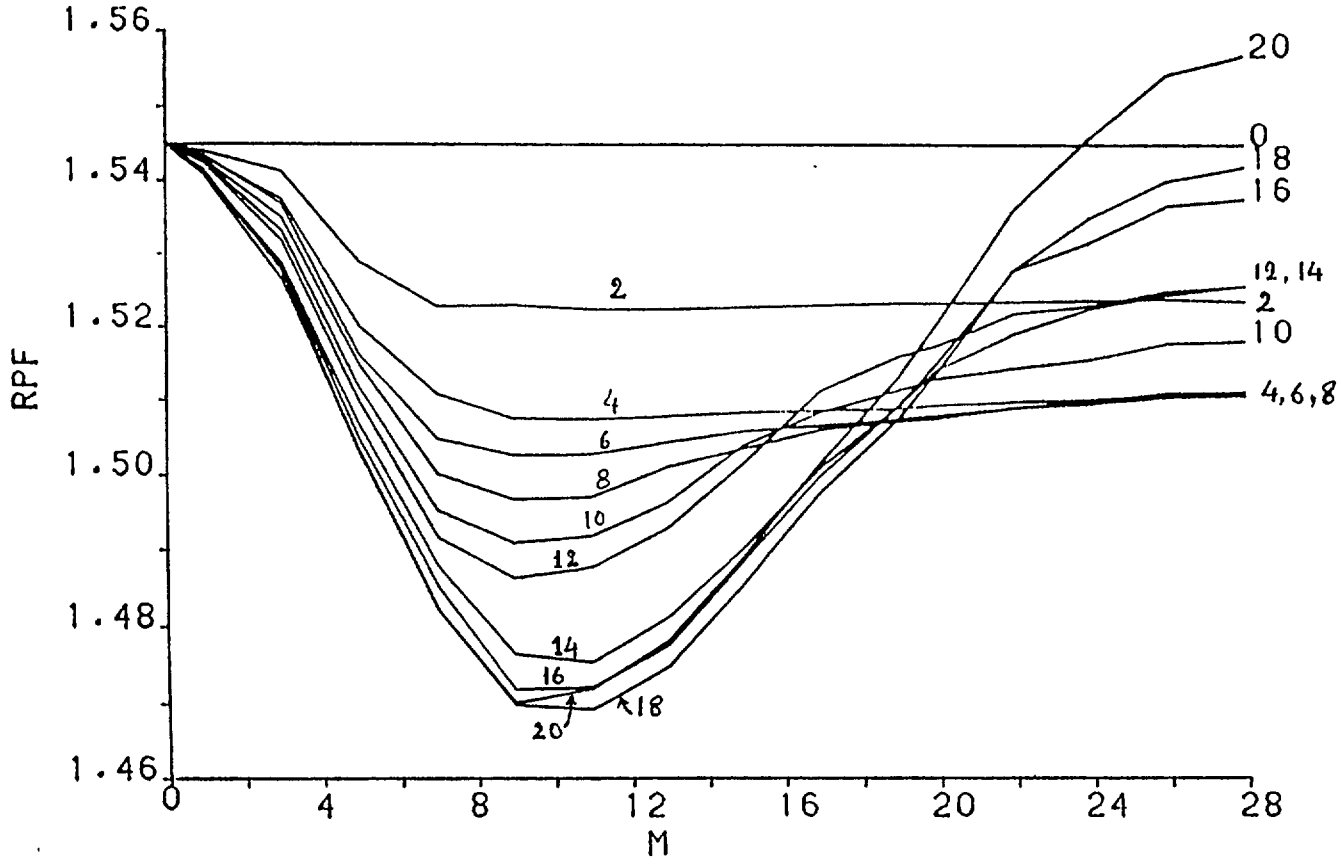

FIG. 5a. RPF as a function of M.

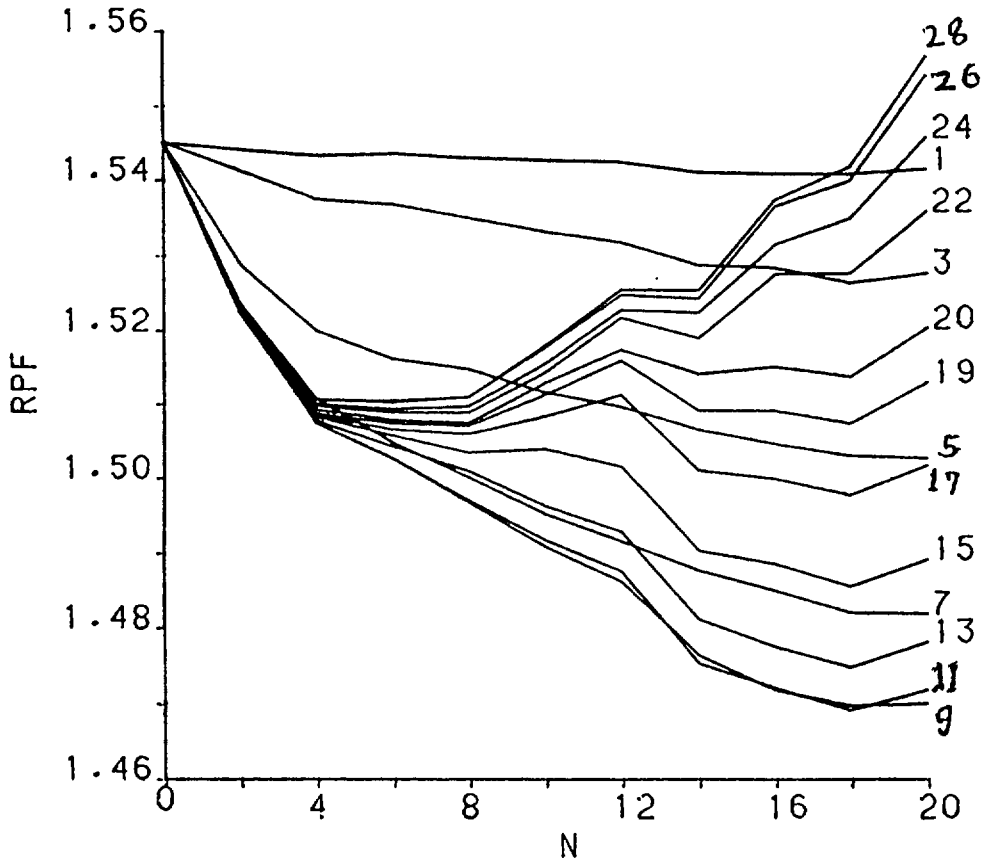

FIG. 5b. RPF as a function of N.

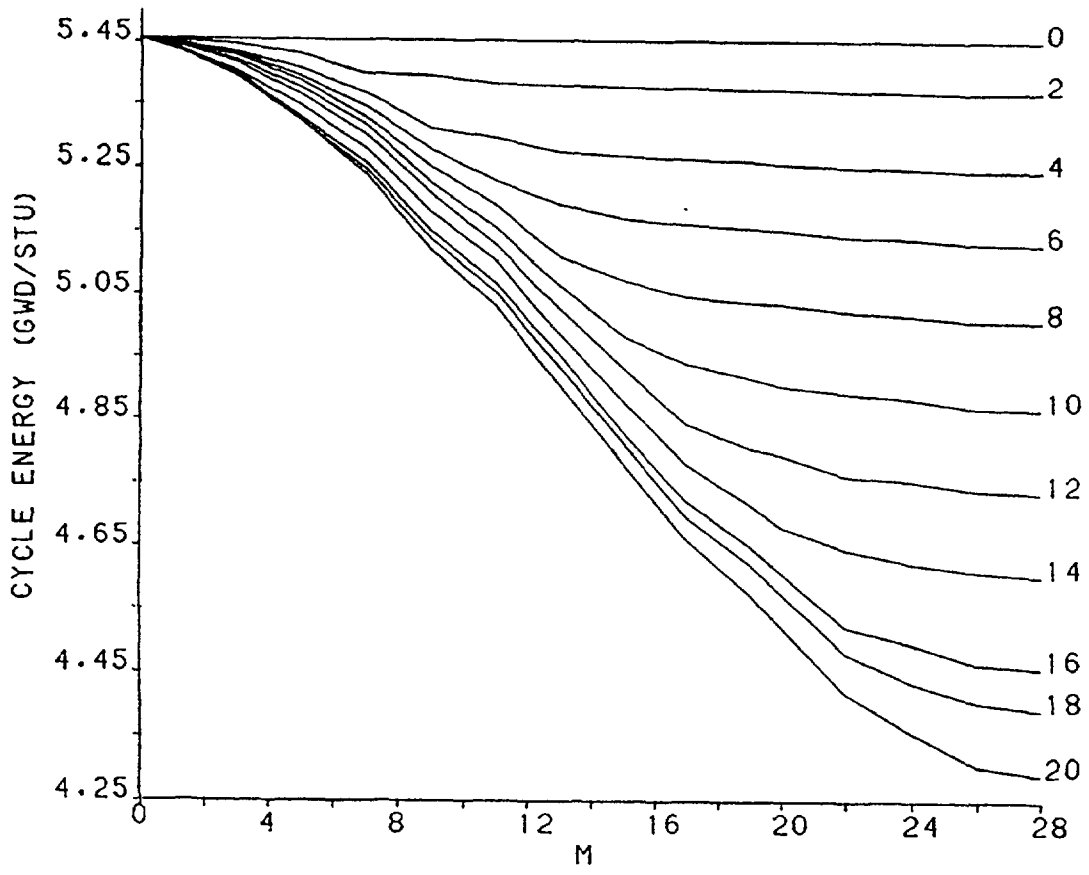

FIG. 5c. Cycle energy as a function of M.

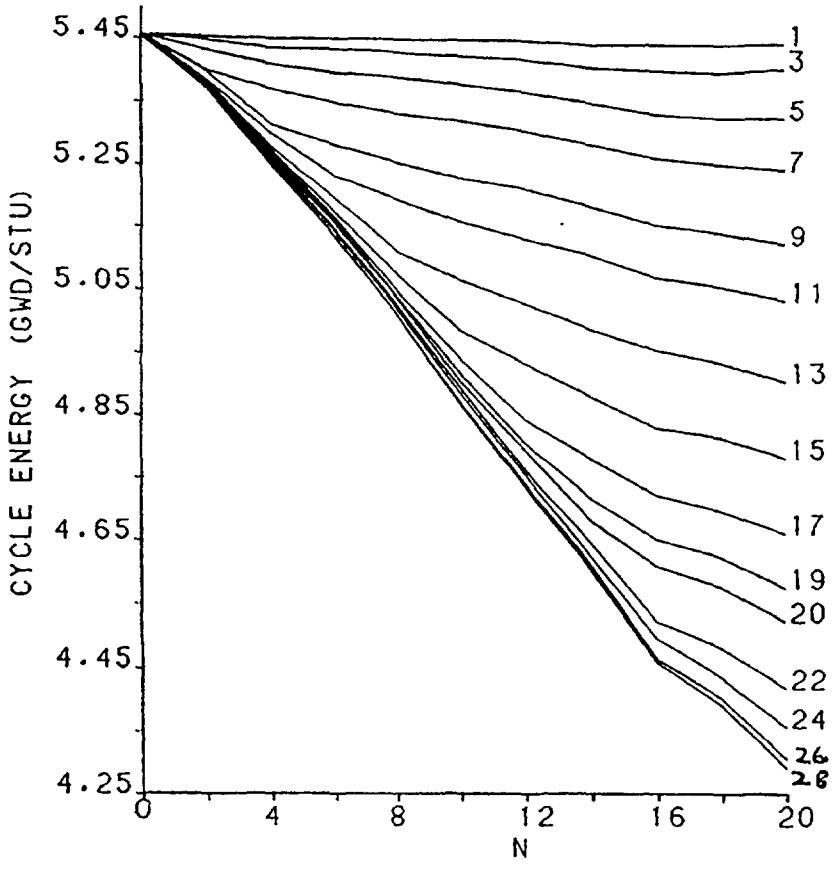

FIG. 5d. Cycle energy as a function of N.

possible value of RPF. The values of the parameters M and N corresponding to it are searched using the procedure described in a later section. All these three points lie on the envelope. This envelope is characteristic of a given number of fresh bundles in a given exercise. The above three characteristic reload patterns for different numbers of fresh bundles are also worked out. The values of their parameters M and N, their cycle energies and the RPF are given in Table-1.

From the Figure-3 we can conclude that there are optimum and non-optimum reload patterns. Secondly, the shape of the optimum segments (of the envelopes) in both, Figure-3 as well as Figure-4 indicates that higher cycle energy can be expected from fuel arrangements with higher power peaking.

The Figure-5 shows the behavior of RPF as well as cycle energy as a function of M and N as obtained in the study. The following qualitative conclusions were drawn from these figures.

(1)As M increases RPF goes through a minima. As N increases RPF may go through a minima for larger values of M.

(2)Cycle energy monotonically decreases with M as well as N.

This information can be used for working out an optimisation procedure. We have used it to improve our optimisation procedure to some extent in our code now. The procedure should be such that it keeps trial patterns from drifting away from the optimum curve and leads one to that point on this curve which satisfies the constraint on the radial power peaking. Even on this curve there may be several reload patterns with lower values of radial power peaking. Among these patterns the procedure must lead us to the one which has the maximum cycle energy.

The basic pattern is the pattern on the optimum curve with maximum cycle energy. Therefore we start with the basic

pattern as per our earlier procedure. However, we now know that RPF as a function of M as well as N can undergo a minima. If we increase M by steps, which we do for economy we are likely to miss a minima. Then there is no point in increasing the value of M. Therefore the algorithm is made to keep an eye on whether the minima is missed. If the minima is crossed before we obtain a reload pattern whose RPF is less than the given allowed value we change the aim. We now aim at searching the value of M corresponding to the minima of RPF. Thus we now have an algorithm which finds the pattern corresponding to the minimum possible power peaking. This is pressed into service if the minima in RPF is crossed while searching a reload pattern whose RPF is acceptable. This algorithm yields the pattern with minimum possible RPF if an unattainablly low value for RPF is provided to as its target. Table-1 provides RPF and the cycle energy as obtained for such patterns (with minimum RPF) as a function of the number of fresh bundles.

The minimum value of RPF is either greater than the allowed value or it is not so. If it is higher than acceptable value the solution to the problem does not exist and optimisation process stops. Otherwise, it implies the possibility of existance of reload patterns with RPF values which comply with the constraint and then we must try to search for the one among them which has maximum cycle energy. The value of M for this pattern can be expected to lie between its value at which missing the maxima of RPF was noticed and its value at which RPF has its minima. Over this range the value of M at which the cycle energy is maximum can easily be found out by continuing to narrow down the range. After this stage the program tries to decrease N till the RPF remains within the limit, so that the cycle energy increases.

If an achievable value of RPF is provided the procedure leads to a near-optimum reload pattern. The optimisation procedure is rather crude. Therefore while this reload pattern satisfies the constraint on the radial power peaking its cycle energy is only close to maximum possible value. The reload patterns obtained by this procedure are represented by the

points marked by an arrow in Figure-3. It is possible to improve it so that the point lies on the envelope (on the right side) in the figure. Given the value of the maximum allowable RPF and other data on the available fuel bundles, the algorithm constitutes about eight to ten trial reload patterns taking about three to four minutes of computor time on ND-500 before obtaining such a pattern. The Table-2 gives the power peaking vs. the cycle energy for the biparametrically optimised reload patterns in which the same set of fuel bundles including the number of fresh bundles was used. The required number of the exposed bundles used in this were from EOC-12.

4. MULTIPARAMETRIC MODIFICATION

The biparametric reload patterns follow the guidelines for optimisation. On that count the optimum one out of them is expected to be close to the overall optimum pattern. A multi-parametric algorithm involving an arbitrary number of radial zones is presented. It allows shuffling the fuel bundles from one zone to another. In our case we have chosen seven such zones. One can use of this algorithm interactively to improve the optimisation till satisfaction. This algorithm serves two purposes. Firstly, it can evaluate efficacy of a given biparametric reload pattern. The scope for improvement by radial shuffling will depend upon how far the given pattern is from the optimum pattern. The optimum pattern will provide no scope for improvement. Any radial shuffling would result in a worse reload pattern. By this method it is found that the optimum of the biparametric reload patterns is very close to the overall optimum pattern. Secondly, it enables one to improve the biparametric pattern to the extent possible. The input corresponding to the fruitful shuffling to improve the optimum biparametric reload pattern is for the TAPS reactors is now standardised. The Table-2 includes also the data (the RPF and the cycle energy) on the improvement over the biparametrically optimised patterns as described above.

TABLE-2

BIPARAMETRICALLY OPTIMISED RELOAD PATTERNS (STUDY-90)
RPF AND CYCLE ENERGY FOR GIVEN UPPER BOUND ON RPF
WITH TOTAL NUMBER OF FRESH BUNDLES 100
(OR 28 IN A SECTOR CONSISTING OF 78 FUEL LOCATIONS)

UPPER BOUND ON RPF	OPTIMUM BIPARAMETRIC PATTERN				IMPROVED PATTERN	
	RPF	DELE	M	N	RPF	DELE
1.54	1.5400	5.4452	2	3	1.5373	5.4866
1.53	1.5300	5.4140	2	22	1.5273	5.4560
1.52	1.5172	5.3748	4	15	1.5146	5.4172
1.51	1.5097	5.3667	5	12	1.5085	5.4086
1.50	1.4969	5.3052	6	17	1.4845	5.3022
1.49	1.4894	5.2949	7	13	1.4885	5.3369
1.48	1.4798	5.2417	7	22	1.4801	5.2846

5. APPLICATION

The biparametric algorithm with the search procedure as well as the algorithm for radial exposure distribution improvement are incorporated in the computor code CORECOOK and form its main feature. The code has been successfully used to generate four out five SSC reload patterns loaded into TAPS cores so far and it is now in regular use. This also shows that the range of the power peaking factors of the optimum biparametric reload patterns is adequate for the practicle requirements of the current cycles of TAPS reactors.

6. CONCLUSIONS

(1)Azimuthal symmetry, Low Leakage Loading and checker board arrangement of fresh and exposed fuel bundles are built into the method presented here for constituting reload

patterns for BWRs at TAPS. An enormously large number of undesirable reload patterns, therefore, are left out of consideration.

(2)The balance of optimisation can be shifted towards lower Haling power peaking or higher cycle energy with the help of two input parameters.

(3)The number of these biparametric reload patterns is small enough to permit exhaustive study. The set of points representing the biparametric reload patterns exhaustively, on a plot of the radial (Haling) power peaking factor vs. the cycle energy, are seen to be enveloped into region with a crescent-like shape.

(4)The segment of the envelope consisting of the points with maximum cycle energy for a given radial power peaking factor shows that higher cycle energy can be obtained for higher Haling power peaking. Secondly, we see that the range of the power peaking factors covered by this segment is adequate for the practicle requirements of the BWRs at TAPS.

(5)The optimum biparametric reload patterns are found to be very close to optimum reload patterns. However, the method has been augmented with a multiparametric algorithm which can be used to exploit the little scope for further improvement.

(6)The task of generating optimum reload patterns under the relevant constraints is computorised. A code CORECOOK has been written primarily for this purpose. The code has been successfully used to generate five recent-most reload patterns loaded into TAPS reactors so far and it is now in regular use.

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HEXANES CODE SYSTEM FOR VVER IN-CORE FUEL MANAGEMENT CALCULATIONS AND LOADING PATTERN ANALYSES

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Abstract

In the new evaluation of safety aspects of VVER operating reactors the idea of expansion of the reloading design limitations is suggested. The changes in the in-core fuel management strategies have brought about some changes in the neutron - physics characteristics of the core. The development and validation of advanced core analysis code system, which will not only provide a set of parameters satisfying the new design limitations but also can serve as an input for the safety analysis is an important and up to date task. In this case the accuracy of the calculated parameters and the efficiency of applied models should be taken into account.

To meet the needs of advanced loading patterns and in-core fuel management improvements in VVER the HEXANES code system is being developed in the Institute for Nuclear Research and Nuclear Energy at the Bulgarian Academy of Sciences. This requires routine use of fast accurate 3D core model with updated and extended cross section libraries. The system HEXANES consists of four interconnected codes: the NESSEL-IV-EC code for cell and fuel assembly lattice calculations and few group cross section generation; the HEXAB-2D code for detailed pinwise diffusion core calculations; the HEXAB-3D code for few-group corrected finite - diffrence core simulation in hexagonal geometry; the TCALC code - for thermal-hydrolic analysis. The current status and the recent progress made in this code system are presented in the paper. The validation strategy and the HEXANES capability for power and burn-up calculations of VVER cores have been studied by detailed comparison of results with various VVER benchmarks. Some recent HEXANES test investigations on integrated burnable absorber use in VVER are also included in the paper.

I. INTRODUCTION

To meet the needs of VVERs advanced loading patterns and in-core fuel management improvements the HEXANES code system is being developed and qualified. The implementation of advanced fuel and burnable absorber designs, the usage of low leakage loadings, the increase in the reloading enrichments and the reduction of steel content in the active core have made the reactor core analysis more demanding and the definition of loading patterns - more difficult.

This complexity requires routine use of fast accurate 3D core model with updated and extended cross section libraries. Details of the methodologies and verification of the HEXANES code system have been previously reported [1,2,3]. This report summarizes the development of methods and modifications in the presently available codes, the investigations, verifying the existing and up-dated data libraries and the establishment of VVER benchmarks , based on realistic design and experimental reactor data.

II. CODE SYSTEM DESCRIPTION

The system HEXANES consists of four interconnected codes: NESSEL-IV-EC code for cell and fuel assembly spectral calculations; HEXAB-2D code for detailed pinwise, few-group, diffusion core calculations; HEXAB-3D code for few-group corrected finite-difference core simulation in hexagonal geometry; and TCALC code for thermal-hydraulic analysis.

The NESSEL-IV-EC code [4] is intended to calculate the local neutron physics characteristics of light water moderated reactor cores. It calculates effective few-group diffusion parameters and depletion not only for a given subzone, but also for the entire assembly , taking into consideration the strong inhomogeneities inherent to this type of reactor cores. It uses 34-group data library LIB4P containing microscopic data for about 200 isotopes.

The two-dimensional few-group code HEXAB-2D [5] performs rodwise power distribution calculations for VVER cores. Mesh points coincide with the fuel pin centers.

A three-dimensional few-group calculational model in hexagonal geometry, based on the diffusion theory has been developed in HEXAB-3D code [6]. The standard inner-outer iterative strategy is used. The high effective two-sweep iterative method

AGA DSOR has been used for the inner iterations in each energy group [7]. The power method, combined with Chebishev polynomial acceleration for outer iterations is applied in the code. Numerical investigations have been carried out for determination of both the optimal combination between Chebishev acceleration and the factorization methods and the minimization of total CPU time [2].

An effective thermal-hydraulic model is used in the THCALC module to determine the fuel and moderator temperatures and coolant mass flow rate. The general assumptions used for thermal-hydraulic analysis are discussed in [8].

III. METHODICAL ADVANCES

III.1. Model for Improving the Diffusion Problem Solution Accuracy in Hexagonal Geometry

To improve the accuracy of the calculated integral and local reactor parameters without significant increase in the computer time and storage it has been developed an effective coarse-mesh method [1,2,3]. This method decreases errors due to the coarse-mesh implementation by means of correcting the coefficients of finite-difference scheme. A consistent two-dimensional procedure is used in HEXAB-3D code to obtain correction parameters. As a result of this two problems have been solved - the first one is the improved radial neutron leakage expression and the second one - the introduction of average flux value in the reaction rates calculation in the diffusion balance equation.

The computation time for the correction procedure, applied in HEXAB-3D code is only 30% of the total CPU time. To attain the correction parameters accuracy, required in practice, 3-5 recalculations are enough.

III.2.Modified Two-Sweep Iterative Method MAGA DSOR

Inner iterations are solved using two different incomplete factorization techniques: AGA two-sweep iterative method [7] and modified AGA two-sweep iterative method [9] both accelerated by the double successive overrelaxation procedure. The introduced modification of AGA-iterative scheme allows to improve the convergence rate in comparison with two-sweep method AGA and, thus to increase the efficiency of the code.

IV. VVER BENCHMARKS AND VALIDATION STRATEGY

The validation of the HEXANES capability for power and burn-up calculations in VVER cores includes detailed comparison of results with various VVER benchmarks.

A lot of numerical calculations with HEXAB-3D code have been carried out on test models for VVER-440 [10] and VVER-1000 [11] reactors and the results are presented in [1,2,3]. The typical discrepancies between the reference and HEXAB results are as follows:

- K_{eff} - within $\pm$ 0.15%;
- power distribution within $\pm$ 2%.

The HEXANES system can be applied to the in-core fuel management benchmark calculations and project and non-project loading patterns analyses of VVER-1000 reactors (Units 5 and 6 at the Kozloduy NPP), when this reactors will be operated in three years fuel cycle with burnable absorbers. Because all of the Russian VVERs in Kozloduy NPP are of the standard V230, (440 MWe,Units-1,2,3,4) or V312, (1000 MWe,Units-5,6) types both can serve as models for providing data on the benchmark specifications, developed according to the IAEA Co-ordinated Research Programme. These include best estimate design information, set of realistic reactor data and results of measurements [12]. Specific parameters have been defined for performing the benchmark calculations. Some of the measurements and operating data have been used for verification and comparison of the calculated by the code system parameters . The results obtained for Unit 3 of NPP Kozloduj show,

Table 1

Benchmark calculation results in two-dimensional hexagonal geometry for Unit 3 of Kozloduy NPP

ε =1.10^{-5} and ε =ε =1.10^{-3}, mesh point number - 6364, energy group number - 4, mesh step - 1.22cm

Method	K_{eff}	Number of iterations outer	inner	CPU-time (sec)
AGA DSOR	1.10256	185	185*4=740	1509
MAGA-DSOR	1.10248	141	141*4=564	1221

Table 2

Gadolinium burnable absorber investigations
Calculation results for the first case

$K_{inf}(B^2=0) = 1.11105$

Few-group flux distribution

group 1

8.462864-1	8.427024-1	8.468490-1
8.427024-1	8.312431-1	8.433111-1
8.468490-1	8.433111-1	8.474752-1

group 2

9.985084-1	9.962627-1	9.992411-1
9.962627-1	9.890136-1	9.970212-1
9.992411-1	9.970212-1	1.000000+0

group 3

8.434274-1	8.420504-1	8.441035-1
8.420504-1	8.360178-1	8.427204-1
8.441035-1	8.427204-1	8.447567-1

group 4

5.585392-1	5.270900-1	5.588694-1
5.270900-1	3.782567-1	5.274295-1
5.588694-1	5.274295-1	5.592701-1

One group fluxes

3.247	3.208	3.249
3.208	3.035	3.210
3.249	3.211	3.252

Relative Power Distribution

1.111	1.062	1.111
1.062	0.302	1.062
1.111	1.062	1.111

Absorption rates distribution

0.0829	0.0796	0.0829
0.0796	0.1950	0.0796
0.0829	0.0796	0.0829

Fission rates distribution

0.0468	0.0446	0.0469
0.0446	0.0124	0.0446
0.0469	0.0446	0.0469

Table 3

Gadolinium burnable absorber investigations

Calculation results for the second case

$K_{inf}(B^2=0) = 1.382856$

Few-group flux distribution

group 1

8.449103-1	8.453199-1	8.456459-1
8.453199-1	8.462077-1	8.460299-1
8.456459-1	8.460299-1	8.463458-1

group 2

9.983035-1	9.986860-1	9.991850-1
9.986860-1	9.992585-1	9.995278-1
9.991850-1	9.995278-1	1.000000+0

group 3

8.486729-1	8.486974-1	8.494189-1
8.486974-1	8.478757-1	8.494009-1
8.494189-1	8.494009-1	8.500873-1

group 4

6.383233-1	6.382390-1	6.388232-1
6.382390-1	6.372304-1	6.386923-1
6.388232-1	6.386923-1	6.392529-1

One group fluxes

3.330	3.331	3.333
3.331	3.331	3.331
3.333	3.331	3.332

Relative Power Distribution

0.995	0.995	0.996
0.995	1.003	0.996
0.996	0.996	0.996

Absorption rates distribution

0.0925	0.0925	0.0926
0.0925	0.0953	0.0926
0.0926	0.0926	0.0927

Fission rates distribution

0.0534	0.0534	0.0534
0.0534	0.0555	0.0534
0.0534	0.0534	0.0535

that the use of modified two-sweep iterative method MAGA DSOR improves the code efficiency, respectively decreases the CPU-time up to 20% compared to the AGA DSOR method (see Table 1).

As a part of the IAEA Research Coordinated Program on burnable absorbers in VVERs some test calculations have been carried out by the HEXANES code system investigating the influence of Gd presence in the fuel on the main reactor physics parameters. It has been considered two cases in the NESSEL-IV-EC part of the calculation - a simple reactor lattice, with a fuel pin in its center containing Gd in the first case, and not containing Gd in the second case. HEXANES calculation results for the K_{inf} values, the relative power distribution, the few-group and one-group flux distributions, and the most important reaction rates for both cases are shown in the Tables 2 and 3 respectively. The central fuel rod has been divided into 10 fine mathematical zones in the thermal region calculation. The geometry and material compositions input data are as follows:

Fuel pellet radius: 0.41135 cm
Inner clad radius: 0.4215 cm
Outer clad radius: 0.4815 cm
Lattice pitch: 1.35 cm

Isotopes Number densities $[10^{24}/cm^3]$

	Reg.fuel	Clad	Water	Gd fuel	Zero Gd
^{1}H	-	-	6.6763-2	-	-
^{16}O	4.5179-2	-	3.3382-2	4.5249-2	4.5249
Zr	-	4.3241-2	-	-	-
^{155}Gd	-	-	-	1.4788-4	0.0
^{157}Gd	-	-	-	1.5637-4	0.0
^{235}U	7.2953-4	-	-	7.7052-4	7.7052-4
^{238}U	2.1860-2	-	-	2.1101-2	2.1101-2

The code system will be further improved, new options will be added and benchmark calculations will be performed. At the time being we incorporate burnup and thermal hydraulic feedback modelling and validate the improved code system on NPP design and operational data. In this connection the IAEA CRP activities

on in-core fuel management benchmarks and burnable absorber investigations are very useful for our country.

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RELOADING TECHNIQUES THROUGH EXPERT SYSTEMS: THE CUBAN EXPERIENCE

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Abstract

In the ICFM activities the selection of an optimal reload pattern plays a special role. Since 1989 began the development of an expert system, ROSE, designed as a computational reloading pattern generator. The nowadays system version included the passive utility to manage square or hexagonal lattices in standard and low leakage approach. The preference rules made use of heuristical evaluations. The system is currently confeed by means of an interface to the coarse mesh simulator SPPS-1 in order to evaluate the performance of resulting patterns for hexagonal WWER type reactors and data between ROSE and SPPS-1 is easily transferred in both directions. In system adjustment and validation for hexagonal lattices LOVIISA NPP (Finland) and Juragua NPP (Cuba) data has been used.

The possibilities of the system for actual reload patterns generation and for developing shield in reloading are also analyzed in this paper.

1. Introduction.

In the In-Core Fuel Management activities a primary attention is given to the optimal reload pattern selection. It consists in the search of a reloading design able to satisfy predetermined requirements taking into account the specific needs of some NPP, and of course of some reactor type, and the electroenergetic system to which it will be coupled.

In order to realize this task several approaches have been used. The more relevant ones are the algorithmical and the artificial intelligence approach.

The optimization algorithmical methods development requires the implementation of relatively complicated codes and the manipulation of a big amount of data in the form of group constants libraries. In this field we have already developed a system for the axial Gd distribution optimization in a reactor with physical and neutronical characteristics typical for a WWER [1].

The workgroup previous experience was concentrated in the individual skills of the experts. They were empirically able to determine the better distributions for the fresh fuel and for the medium, lower or higher enrichment fuel in the case of the WWER cores several times already investigated by means of the three dimensional simulators like BIPR [2] or SPPS-1 [3].

The increasing complexity for the reloading strategies and the need to perform a reload design in a short time constituted a strong motivation for the development of a reloading patterns computarized generator.

This feature has made that in several countries (France, USA), the use of expert systems to this aim has been extended.

The design of a reloading pattern for the NPP involves a several and multiple stage proccess, with a series of design decisions taken at different temporal points . The design of the reload pattern with a fixed number of fuel assemblies of each type for the fresh fuel is prepared months before the reloading operation, assuming expected parameters for the partially burned fuel. The final situation may be, however, very different than the expected by extrapolation.

The decisions to assume at that moment may include the variation of the fuel enrichment for the assemblies to be reloaded or the variation of the number of fresh assemblies for a given enrichment to be inserted in the core.

All these reasons have made that as a first step, and trying to avoid the utilization of conservative schemes that imply a non optimal use of nuclear fuel , we have decided to develope the expert system ROSE [4] condensating the rules that a nice expert would apply in his pattern selection and using the artificial intelligence methods.

The ROSE system is a computerized generator of reloading patterns for the power reactor cores, with square or hexagonal lattices. Its main objectives are the quick generation of reloading paterns or the modification of the already existing ones. It is also a tool for the formulation, adjusdment and proof of the knowledge rules used by a skilled fuel manager and may be used as a learning consultant.

It is written in TURBOPASCAL for personal computers IBM compatibles with requirements of 640 Kb RAM and a high density floppy disk. The resulting execution times are similar or lower than reported by literature for the same task problems, by means of systems using logical languages as LISP.

2. Knowledge modelling through elimination and preference rules in the expert system ROSE.

Up to now, the selection of the optimal configuration was namely done from the analysis and inspection of the fuel batch on the basis of the already accumulated experience. This experience let us to sintetize as elimination rules the consecuences arising from a given assembly allocation in a determined position into the reactor core. In the case of hexagonal lattices with standard and low leakage patterns these consecuences were known but that was not the case for square lattices. In spite of this, the available literature reported the development of reload pattern generators only for square lattices and this was our starting point [5,6,7].

The elimination and preference rules depend on the investigated lattice type and on the strategy to be applied for a given fuel batch. Those rules condensate the human experience and its expresion is then always able to be improved.

In the case of low leakage reload patterns, a set of rules were developed to the aim of getting the desired objective . The system adjusdment and comprobation was made on the basis of typical problems for PWR, it means for square lattices. Later on, the knowledge basis was enhanced by means of the insertion of appropiate elimination rules for getting low leakage patterns in hexagonal lattices, for both reduced and complete zones corresponding to Loviisa (Finland) and Juragua (Cuba) NPPs, respectively. The complete zones resulted the more complex ones, of course requiring the higher number of preference and elimination rules.

From square (PWR) to hexagonal (WWER) lattices the hardest work arose from the regulation assemblies consideration (ARK), that physically provocate the most crude changes in connection with the core properties because they are strong heterogenities placed into the zone.

For the expert system work the strategy breadth-first was developed [8]. It consists in the level expansion that goes not to the allocation of a new assembly in the core until the former one has not been shuffled to all the allowed positions generating a set of partial reload patterns.

The originated tree is shaped by levels using the preference rules in order to limitate the set of possible solutions. Afterwards, the heuristical evaluators were included inside the system as a numerical way to evaluate the preferences allowing to considerate not only the best values for the power peaking factors but also the desired cycle extension [7].

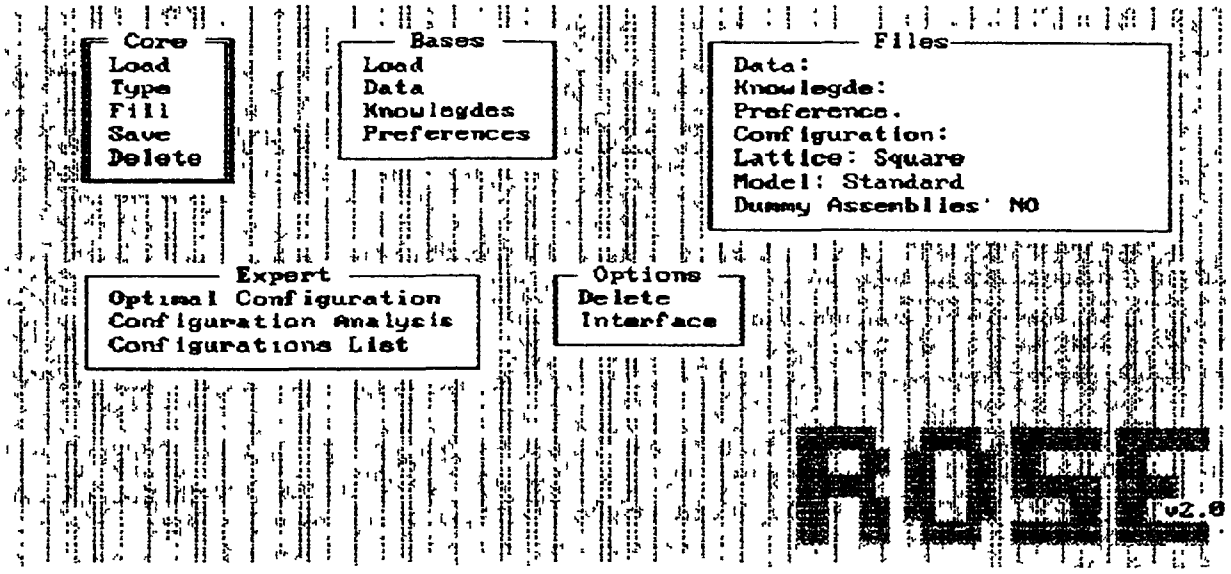
The elimination rules are applied to each assembly allocation in a predetermined position whilst the preference rules are applied to each obtained pattern (partial or total) in order to choose the best zones from that set.

The clustering techniques insertion is foreseen as an unavoidable step looking for simplification. This will allow to apply the analysis routine over a pattern representative of the class obtaining conclusions true for all the class. The development and application of those techniques will include the creation of a pool containing several used patterns in PWR and WWER NPPs with previously known performance. Some elements of classification techniques are currently used in the system for the levels expansion [9,10].

3. System description and Structure. Environment where it has been developed.

The main window for ROSE system is composed of four menus. They bring the possibility to perform the four main actions of the system: to manipulate zones, to manipulate data, knowledge and preference basis,to consultate the expert system and to

perform calculations with a three dimensional simulator for evaluation of a selected hexagonal zone.



Besides this in that screen appears an information window concerning the files in use. Those files may include the knowledge basis files (preference and elimination rules), files containing description data for fuel batch, and files with description of the analysed zone.

The Zone menu allows to define the zone "type " to be used. Its lattice may be hexagonal or square , may use a standard or a low leakage model and may have or not some dummy assemblies in the core. It is possible also to fill the zone in a desired way and afterwards to consultate the expert about the feasibility or not of that conformed pattern and also to calculate the power peaking factor by using the three dimensional simulator. This zone or any other generated by the system may be "saved" and "loaded" in any other ocassion. If some pattern is not needed anymore it may be deleted by the own system.

The Basis menu allows to define the batch of data that will be used in the analysis , defining the assemblies with their properties and the characteristics for each zone position. The "Knowledge" basis including the "Preferences" are defined through the production rules according to the sintaxis used in the code In both the cases , it becomes neccessary to define first the data basis from the fuel batch. All the defined parameters may be saved inside each option in an independent way.

Besides this all may be loaded simultaneously defining the knowledge basis name and in this way the files of data and preferences with the same name will be also loaded.

The files names are created according to the zone main features, the first two letters are CD for square lattices and HX for hexagonal lattices, the following letters are ST for standard model and BF for low leakage model.

The work menu with the expert system allows, once one has defined the zone and the knowledge and preferences rules to be

used, to determine "Optimal Configurations" up to a prefixed quantity. The fuel assemblies are allocated in the zone following the principles set by the rules in order to obtain the desired effect, for example the minimum for the power peaking factor.

Once the expert system proposes a set of zones you may "Analyse the configurations" asking to the system the reasons of its selection in both global and particular assemblies allocation. (Why that zone?, Why that position for a given assembly?).

The "Configurations Listing" with the assigned values according to its preference may be obtained by the third option. The listing is ordered from the lower preferences to the higher ones.

The last Options menu gives the possibility to "Delete" everything already defined: data, rules and zones in order to start a new analysis from the begining. The "Interface" helps in transfering files from ROSE to the SPPS-1 and backward. With this aim it is neccessary to define the path to find the SPPS-1 code when ROSE is being executed.

The information exchange is done by taking from the input file for SPPS-1, the batch of data that it uses and creating a data basis fo ROSE. In the same way, once the new assemblies distribution in the zone is done, the data for ROSE may be put in the input file for SPPS-1.

Input Data

For the entering of the assemblies and their characteristics the following screen format will appear.

No.	S	C	1	2
1	ARK	7	0	-1
2	3A9	1	2.63E+01	2
3	3A8	1	2.44E+01	2
4	3A7	1	2.39E+01	2
5	3A6	1	2.35E+01	2
6	2A6	2	1.51E+01	1
7	2A1	1	9.97	1
8	1A	4	0	0
9	1A	4	0	0
10	1A	4	0	0
11	1A	4	0	0
12	1B	2	0	1
13	2A4	1	1.30E+01	1
14	2A1	1	9.97	1
15	2A3	2	1.23E+01	1
16	2A5	2	1.46E+01	1
17	2A2	2	1.15E+01	1
18	2A6	2	1.51E+01	1
19	2A7	2	1.58E+01	1
20	3A3	1	2.24E+01	2

-Characteristics-
Lattice: Hexagonal
Model:Low Leakage
Dummy Assemblies

Quantity: 26

In the table the ordinal numeration, symbology, and number of elements for a given type appear in different columns, the following two contains the burnup degree and an assembly identificator (type) with -1 value for the ARK or regulator assemblies, 0 for fresh ones, 1 for the once burned assemblies and 2 for the twice burned assemblies. These characteristics are used by the rules editor for their implementation.

Elimination Rules Input

The elimination rules are used in order to forbid the assembly allocation in the zone under given conditions that may include the assembly characteristics , and the first and second order neighbors characteristics . The following is a rule example:

Elimination Rule # 5

-Characteristics-
Lattice: Hexagonal
Model: Standard

If for the assembly it is true that:

The Type (-1,ARK; 0,A; 1,2A; 2,3A)is equal to 2

and the Coordinate X is greater than 7

and for at least 1 neighbor asembly is true that:

The Type (-1,ARK; 0,A;1,2A;2,3A) is equal to 2

and the Coordinate X is greater than 7

Remarks: Burned with neighbors burned to the periphery.

The contents of the rule is explained in the remarks. "The Type" is a characteristic for the assembly to be inserted in the zone and "The X Coordinate" is a zone property (hexagonal). It expresses the distance in assembly position units form the core center (there are 10 positions in total).

The rule tells that an assembly that has already been two years in the core and whose coordinate is greater than 7 cannot be allocated besides other two years old assembly that is also allocated in coordinate greater than 7 . In the place of the coordinate X the variable Radius migth be used, defining the radial distance to the center of the core.

The rule is defined using the properties of the core position (Radius, Coordinate X, etc.) and the characteristics of the assembly to be allocated in the zone . Afterwards it will be defined a relation (greater than, equal to, etc.) and a value to make the comparisons.

This rule is different from the classical production rules. In this case the rules have always the same succedent: The assembly cannot be allocated in that position. In addition the uncertainties are not used and because of this the achieved conclusion is always totally true.

Preference Rules

These rules are used once the zone has been arranged in order to select, in between all the possible zones given by the elimination rules application, those which exhibit the better properties. An example of this for hexagonal lattices, with low leakage scheme and dummy assemblies :

Preference Rule #3:
For the occupied positions where:

The Type (-1,ARK; 0,A; 1,2A; 2,3A) is greater or equal to 1
and the Burnup is greater than 15
and the Radius is smaller than 1.1E+02
The following magnitudes are added:
A

For each position where the rule is true:
A keeps the Radius value
B keeps the Burnup value
A keeps its value B times
After the comprobation of the rule for all the positions
A keeps its value divided by 250
A keeps it value divided by 30

Variable used for the rule valoration:A

Importance coefficient for the Rule:0.001

Remarks: Burned assemblies allocated in the innermost region (Radius < 125) and (Burnup > 15).

The sense of this kind of rule is to perform the search overall the zone of the positions that satisfy a given condition and to perform some numerical operations with one of its representative variables. The resulting for those operations value is the zone identifier, and the zones with the greater value for that identifier will be chosen.

In this case the main purpose is to take those zones that have the higher concentration of assemblies with fuel burnup greater than 15 allocated in positions with radius smaller than 110 but the nearest to this limiting value. To this aim for each zone we look for the assemblies satisfying both conditions (fuel burnup greater than 15 and radius smaller than 110). The partial products are added, and the final value is divided by the core radius and by the limiting burnup for normalization purposes.

The resulting value will be the final contribution to the total preference value for this zone that contains the partial contributions coming from all the preference rules. Before it is added to the total preference value it must be weighted through an importance coefficient that expresses the the importance of this rule for the total value.

Neither the rule importance nor the contributing values satisfy the uncertainty theory because they are not found in the (-1,1) interval and the relationships that operate over them are not entirely defined by the expert system [11,12].

In spite of this the final value for each preference rule may be interpreted as its weight in the total preference . The total preference value may be understood as the veracity of that this zone is the best. The higher value in between all the zones will be the total certainty.

Optimal configuration

For the optimal configuration search, it may be asked to the system to show the proofs it performs for the assembly allocation but this makes much more slow the proccess. The zone may be analysed partially filled asking to the system to do a pause once the allocation of an "interesting" assembly has been performed.

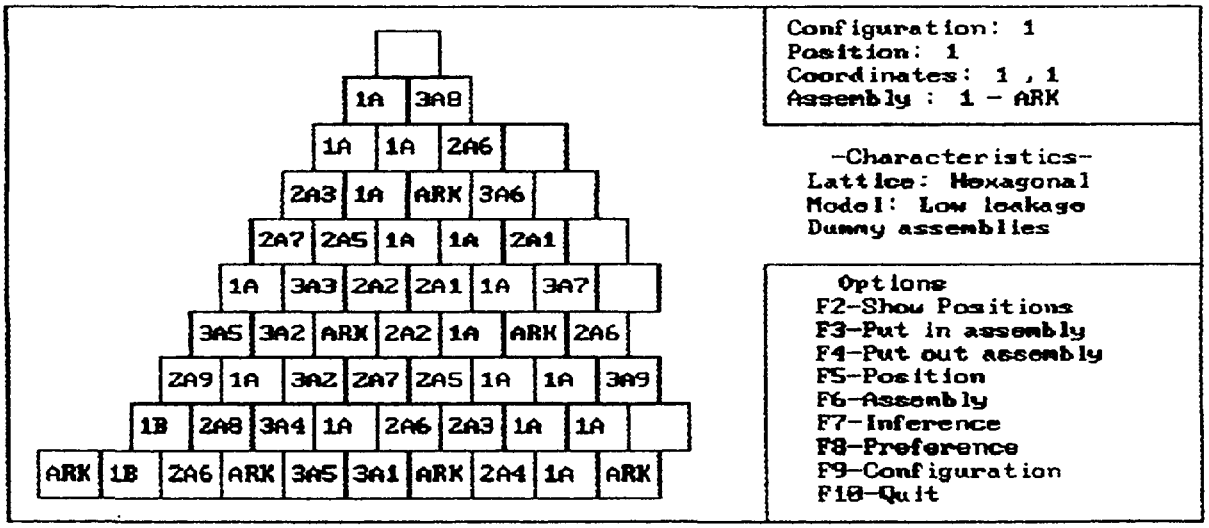
The number of zones expected to be find by the expert system must be also defined. A very small number of zones to be find may originate a no solution proccess because it chooses the configurations on the step and with partially filled ones.

4. Using the system for learning purposes.

The system may be also used as a way for learning:

Configurations Analysis

Any configuration contained in the program configuration listing, generated by the expert system or by the user may be analysed in the following screen format.



There appears the zone (hexagonal in this case) with positions occupied by the assemblies. Using the cursor one may consult the system about the reasons for an assembly to be in a given position, or about the total preference (for the overall zone) or local (the contribution of some assembly to the total). In the same way may be suppressed assemblies from their positions and shuffled to another positions asking to the system its opinion about each assembly shuffling performed in that zone.

In this way may be detected some fail or fissures in the knowledge basis included in the expert system (elimination and preference rules) and to correct them. It is also possible to increase the experience of an unskilled user in connection with the zones reloading.

Interface

In order to be read by ROSE expert system, any input data coming from the SPPS-1 code, the first zone map that appears must be extracted with some text proccessor from the .OUT file:

```
                                    1A/52 3A/53
 3                                  0.752 0.319
                                     0.00 25.77

                                 1A/49 1A/50 3A/51
 4                               1.178 0.954 0.422
                                  0.00  0.00 22.55

                              2A/45 1A/46 1B/47 2A/48
 5                            1.161 1.290 0.895 0.550
                              10.80  0.00  0.00  9.03

                           2A/40 2A/41 1A/42 1A/43 2A/44
 6                         1.143 1.160 1.299 1.060 0.555
                           14.53 13.40  0.00  0.00 11.75

                        1A/34 2A/35 2A/36 3A/37 1A/38 2A/39
 7                      1.382 1.149 1.186 0.973 1.059 0.549
                         0.00 13.86  9.32 24.61  0.00  9.03

                     2A/27 3A/28 2B/29 2A/30 1A/31 1B/32 3A/33
 8                   1.156 1.105 0.977 1.186 1.297 0.893 0.417
                     14.66 18.90 12.52  9.32  0.00  0.00 23.28

                  2A/19 1A/20 3A/21 2A/22 2A/23 1A/24 1A/25 3A/26
 9                1.127 1.374 1.105 1.148 1.159 1.288 0.951 0.311
                  13.75  0.00 18.90 13.85 13.39  0.00  0.00 26.59

               1B/11 2A/12 2A/13 1A/14 2A/15 2A/16 1A/17 1A/18
10             1.082 1.127 1.156 1.381 1.139 1.159 1.177 0.750
                0.00 13.74 14.66  0.00 14.76 10.79  0.00  0.00

    2B/01 3A/02 2A/03 2B/04 2A/05 3A/06 1B/07 3A/08 2A/09 3B/10
11  0.757 0.920 1.099 0.979 1.191 1.044 1.079 1.005 0.895 0.326
    11.74 20.87 11.74 11.64 12.13 24.62  0.00 19.21  8.53 18.88
```

With this information ROSE creates an data basis that may be used for generate patterns with a better performance than this one. These generated zones may be automatically inserted in to an input file to the SPPS-1 being executed the code and obtaining a numerical qualification of the proposed by the expert zones (or by the user).

Most used Models

For several models ROSE already has knowledge basis for the reloading patterns generation.

For square lattices the Low Leakage and Standard models exist. For hexagonal lattices (WWER) the same models are used and besides them, for the low leakage patterns dummy assemblies may be used arising a modified low leakage pattern.

In all cases one asks for, the chosen zones to have the lesser the power peaking factor, although this criteria may be changed by the largest fuel cycle length criteria.

Several examples of obtained reload patterns for different lattices and strategies are shown in the Annex 1.

5. Conclusions .

The second version for the expert system ROSE has been developed for the design of reloading patterns for WWER and PWR with standard and low leakage (complete or with dummy assemblies) strategies using an appropiate knowledge basis and improving the previous preference rules introducing the heuristical evaluators.

The SPPS-1 has been coupled to the system as auxiliary tool for the qualifying of the obtained patterns in the case of hexagonal lattices. The general system environment has been improved through the introduction of several options in menu format.

The main tasks to be solved by the system are :

For a given fuel batch in square or hexagonal lattice it finds the optimal disposition according to the selected model (standard, low leakage or dummy assemblies) .

The found solutions in hexagonal lattices are evaluated determining the assemblies with maximum power peaking factors by using the SPPS-1 code.

The system improves the skill of an unexpertised user answering the questions about the allocation of an assembly in a given position.

The system allows to compare two different reloading patterns assigning to each one a given preference number.

In spite of this, we hope that a more intensive and wide use as verification will be a very good contribution to further development and adjusdment of the system to actual situations of nuclear power plants [13].

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Lattice: Square
Model: Standard

A	A			
B11	B15	A	B14	
C13	C17	A	A	A
C12	C16	C19	C18	A
C15	B17	B16	A	
B13	B18	C14		
C11	B12			

Lattice: Square
Model : Low leakage

A	B12	A			
A	C18	B14	B18	A	
C12	A	C17	B17	A	A
B13	C22	A	C21	B16	
C20	C13	C11	C16		
C15	B11	C19			
B15	C14				

Lattice: Hexagonal
Model: Standard

				1A							
				1A	1A						
			2A3	1A	1A	1A					
			2A5	2A4	ARK	1A	1A				
			2A1	3A7	2A9	3A9	1B	1A			
			2A6	3A1	2A2	2A9	3A9	1A	1A		
			2A7	3A2	ARK	2A5	2A8	ARK	1A		
			3A4	2A7	3A2	3A1	3A7	2A4	1A	1A	
			3A8	3A4	3A3	2A6	2A2	3A6	2A3	1A	1A
		ARK	3A8	3A8	ARK	3A3	2A1	ARK	3A5	1A	ARK

Lattice: Hexagonal
Model: Low leakage

				3A7							
				1A	2A1						
			1A	1A	2A6	3A8					
			2A2	1A	ARK	1A	3A6				
			3A1	2A3	1A	1A	1A	3A6			
			1A	3A2	2A5	2A4	1A	1A	3A8		
			2A7	2A6	ARK	2A4	1A	ARK	2A6		
			3A5	3A4	2A6	3A2	2A3	1A	1A	3A9	
			3A3	3A5	3A4	2A7	3A1	2A2	1A	1A	3A7
		ARK	1B	3A3	ARK	2A7	2A5	ARK	2A1	1A	ARK

Lattice: Hexagonal
Model: Low leakage
Dummy assemblies

				1A	3A8					
			1A	1A	2A6					
			2A3	1A	ARK	3A6				
			2A7	2A5	1A	1A	2A1			
			1A	3A3	2A2	2A1	1A	3A7		
			3A5	3A2	ARK	2A2	1A	ARK	2A6	
			2A9	1A	3A2	2A7	2A5	1A	1A	3A9
			1B	2A8	3A4	1A	2A6	2A3	1A	1A
		ARK	1B	2A6	ARK	3A5	3A1	ARK	2A4	1A

APOLLO-2: AN ADVANCED TRANSPORT CODE FOR LWRs

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Abstract

APOLLO-2 is the successor of the spectrum transport APOLLO (also called APOLLO-1) which has been used for almost two decades as the main computational tool for neutronic analysis and transport assembly calculations in the french PWR program. As APOLLO, APOLLO-2 has been written at the French Commissariat à l'Energie Atomique.

At present APOLLO-1 is a part of routine calculational scheme of EDF (Electricité De France) the French utility and APOLLO-2 will be a part of the next industrial route SCIENCE which is under progress at FRAMATOME the French vendor.

Taking advantage of new software capabilities and taking into account the large accumulated experience gained from the use of APOLLO-1 and other codes developped in the same place some conclusions were drawn :

If APOLLO-1 was an independant code, APOLLO-2 is a part of a larger system called SAPHYR (In french Advanced System for Reactors PHYsics) involving also CRONOS-2 dealing with 3D core diffusion calculations and FLICA-4 dealing with Thermohydraulics problems. So SAPHYR is able to handle all this kind of matters and their couplings.

APOLLO-2 is a fully modular code in which each module corresponds to a specific task : access to the cross-sections libraries, creation of isotopes medium or mixtures, geometry definition, self-shielding calculations, computation of multigroup collision probabilities, flux solver, depletion calculations, transport-transport or transport-diffusion equivalence process, SN calculations, etc.... Modules communicate exclusively by "objects" containing structured data, these objects are identified and handled by user's given names .

Among the major improvements offered by APOLLO-2 the modelization of the self-shielding : it is possible now to deal with a great precision, checked versus Monte-carlo calculations, a fuel rod divided into several concentric rings. So the total production of Plutonium is quite better estimated than before and its radial distribution may be predicted also with a good accuracy. Thanks to the versatility of the code some reference calculations and routine ones may be compared easily because only one parameter, is changed ; for example the self-shielding approximations are modified, the libraries or the flux solver being exactly the same.

Other interesting features have been introduced in APOLLO-2 : the new isotopes JEF.2 are available in 99 and 172 energy groups libraries, the surface leakage model improves the calculation of the control rod efficiency, the flux-current method allows faster calculations, the possibility of an automatic convergence checking during the depletion calculations coupled with fully automatic corrections, heterogeneous diffusion coefficients used for voiding analysis ...

INTRODUCTION : the SAPHYR System

Before presenting APOLLO-2 itself, it is important to notice it is only a part of a code system called SAPHYR (which in French stands for Advanced System for Reactors PHYsics) involving other new generation codes like CRONOS-2[1] and FLICA-4[2]. CRONOS-2 is a core calculation code which solves the multigroup diffusion equation by the finite element or finite difference methods for steady states and kinetics ; it may perform three dimensional pin by pin calculations and recently, a new possibility of three dimensional transport calculation[3] based on the even parity transport equation was

implemented. FLICA-4 solves three dimensional two-phase flows, steady states or transients in PWR reactor cores or in tube bundles with rectangular, triangular or hexagonal geometries.

Originally, the previous versions of these three codes were fully independent : they communicate only through external files with only one exception : for dealing with thermohydraulics feedbacks a specific version of FLICA-3 was introduced inside CRONOS-1 Code. But, as the user's demand for exchanging informations was always increasing and in order to make easier the information transmission between the codes, it was decided to integrate the new version into a global system.

Consequently SAPHYR is based on three basic ideas : the modularity, one unique command language and one unique storage system. Each of the constitutive codes of SAPHYR is fully modular, that means a module is performing a specific task (access to library, self-shielding, depletion calculation ...) by handling structured objects. As an input, the module finds object(s), previously created by other modules, and data given by the user : for example when the user wants to execute a depletion calculation he has to indicate to the module performing this task a set of objects like the geometry, the flux and the depletion chain (by giving their names) and specific data like the time step or the criterion precision on nuclei concentrations.

Modularity is fully exploited through a macrolanguage, called GIBIANE, which allows the user to define its own chain of calculations as a sequential calling of modules. This high-level macrolanguage supports "if" and "loops" ; procedures integrating a chain of macrolanguage commands may be stored and used later in the same way as an elementary module.

The third basic idea, is to have the same structure for all the objects handled in the various modules of SAPHYR, so the Storage/Retrieval System is unique in SAPHYR, there is no more need to have, as it was in the past, three distinctive storage process for the three codes. As SAPHYR is portable, the storage function has an other aim : it allows to transfer objects from one computer to another. For example, it is possible to execute calculations on a CRAY computer, to store resulting objects, and later to postprocess them on a workstation.

In fact there is no more frontier between APOLLO, CRONOS and FLICA modules and to speak about APOLLO-2 is a little bit abusive ; it is in fact an inheritance of the past and an easy way to distinguish different development teams and different subsets of modules. The resulting versatility of the system and its large scope in the reactor physics make SAPHYR a very powerful tool.

APOLLO-2 : GENERALITIES

APOLLO-2 is the successor of the spectrum transport code APOLLO[4] (also called APOLLO-1) which has been used for almost two decades as the main computational tool for neutronic analysis and transport assembly calculations in the French PWR program. As APOLLO, APOLLO-2[5] has been written at the French Commissariat à l'Energie Atomique (CEA).

For the moment APOLLO-1 is a part of the routine calculational scheme of EDF (Electricité de France), the French utility and APOLLO-2 will be a part of the next industrial route SCIENCE which is under progress at FRAMATOME, the French vendor.[15]

APOLLO-2[12] like APOLLO-1[13] will be used for criticality calculations. In this field the fact that APOLLO-2 has a module performing discrete ordinate (SN) calculations and an other allowing to calculate equivalence factors between PIJ and SN calculations are important advantages by comparison with APOLLO-1. The PIJ calculations (calculations using the first-flight collision probability technique) are necessary because it is the only way to take into account the self-shielding and the heterogeneity of a cell. After this step it

is possible to homogenize a cell, and to use an SN method. But it was proved[14] an equivalence process is necessary whatever the number of group used in the SN may be.

Taking advantage of new software capabilities and taking into account the large accumulated experience gained from the use of APOLLO-1, APOLLO-2 is a fully modular code in which each module corresponds to a specific task : access to the cross-sections libraries, creation of isotopes, media or mixtures, geometry definition, self-shielding calculations, computation of multigroup collision probabilities, flux solver, depletion calculations, transport-transport or transport-diffusion equivalence process, SN calculations, etc....

The modularity is a very strong improvement because it makes easier the user's job : when he decides to change something in its calculational scheme the modifications are very localized : to change the number of goups of the cross-section library is quite immediate in APOLLO-2. Only data in the module preparing calculation and output meshes must be changed. But beside this, new features were introduced like the possibility to perform SN calculations and equivalence between SN and PIJ calculations , and also many improvements have been made in the standard process. The major improvements will be discussed thereafter.

APOLLO-2 : SELF-SHIELDING IMPROVEMENTS

The self-shielding model is a very important part of a code like APOLLO-2 ; even if the PIJ flux calculations techniques were perfect it is impossible to get a good result if the self-shielded cross-sections are not very precise. An other thing to emphasize is the fact that other techniques like SN calculations have to rely on the self-shielding calculated by PIJ codes.

For APOLLO-2, one of the most important challenge was to improve the knowledge of the Plutonium balance. To-day, the amount of Plutonium is overestimated by APOLLO-1 from 6 to 7% by comparison with the actual amount measured at La Hague reprocessing plant.

Among the various candidates analyzed for explaining this important discrepancy the most important cause seemed to be a lack of precision in the self-shielding modeling. In order to avoid using 10000 or more groups in the energy-mesh, which is necessary to take into account correctly the shape of the resonances, it is necessary to use a model ; it is the only way to keep an energy mesh with a number of groups around 100. That is the role devoted to the self-shielding to allow to deal the energetical aspect with such a reduced mesh. In APOLLO (1 and 2) the self-shielding is based on a double-equivalence :

- Multigroup equivalence : This is the last step of the self-shielding process. Self-shielded multigroup cross-sections are calculated by preserving actual reaction rates in each group. This is done by iteratively solving a non-linear problem. At the end of this part, the multigroup cross sections (in the APOLLO mesh) are known for all the resonant isotopes and the following steps of the APOLLO calculations will be performed by using the cross-sections defined here.

- Equivalence with an homogeneous medium : The aim of this party is to determine the actual reaction rates which will be used in the multigroup equivalence process. The difficulty is that these reaction rates are depending on the isotopes and on their location. So the spatial domain is divided into several self-shielding regions. For each of them and for each isotope, an equivalent homogeneous medium has to be determined. This determination is based on the fact that the resonance integral calculated with an approximate slowing-down model for the resonant-scattering is the same for both the homogeneous medium and the self-shielded region The use of the exact slowing-down is impossible for the actual geometry because we do not know how to solve the exact equation in order to perform the actual reaction rates. Then, when the correspondance with the homogeneous medium is well established for the approximate slowing down model, it is assumed that this equivalence is still valid for the exact slowing-down. This is the basic assumption of the

modeling. In this case the calculation of the exact reaction rates using an exact slowing-down and a much more refined mesh than APOLLO is performed off-line for a homogeneous medium. Its results are tabulated as a function of the background cross-section (which is the ratio of the total cross-section of non-resonant isotopes divided by the concentration of the resonant one) which, with the temperature fully characterizes the homogeneous medium.

Due to the will of improving this very sensitive and strategic part of the code many works in APOLLO-2 were devoted to the self-shielding. The more important improvements made, in this area, in APOLLO-2, are listed below :

Possibility of using various slowing-down models :

As it was said earlier, the use of the exact slowing-down is impossible and it is necessary to resort to a slowing-down model. In APOLLO-1 only the NR model (Narrow resonance model which assumes that the average lethargy gained by a neutron after a collision with the resonant isotope is much more important than the width of the resonance) is available. But it is important for determining the homogeneous equivalent medium to use a model which is as close to the reality as possible. But, if physically at high energy the NR model is well suited, it is not the case for energy lower than 50 eV. So it was decided to implement other models like the Wide resonance one (WR) which is better when the resonance width is larger than the lethargy gained by the neutron and the Intermediate resonance one (IR) which is a combination of the two previous ones. Later an other model called statiscal one (ST) was proved to be more satisfactory than the NR one ; the difference is that the NR model assumes that the resonance is isolated which is not physically true, whereas the ST model avoids to have to make this assumption.

Possibility of using a group per group equivalence process :

In APOLLO-1 the equivalence process necessary to determine the homogeneous equivalent medium was made globally on the whole energy range and after, some other hypotheses were made in order to define group per group the background cross sections. In APOLLO-2 this possibility still remains because it is time saving but beside a more accurate process is offered to the user : he may directly determine an equivalent medium group per group. This is specially interesting in APOLLO-2 where it is also possible to change the slowing-down model from one group to another. And practically it is advised to use a group per group equivalence with the ST model for groups over 50eV and WR model for the other groups. Furthermore this allows to keep exactly the same modelization even for very thermal resonance like the 1 eV Pu40 one. In APOLLO-1 the self-shielding process described above was used only in the epithermal domain (above 2.76 eV) and very simplest ones were used for dealing with thermal resonances of Pu240 and Pu242.

Possibility of using the background matrix formalism :

One of the difficulties which occurs in the self-shielding process is the following : when one has to deal with more than one self-shielded area, equations show the need to know the slowing-down reaction rates in the other areas ; for avoiding this complication some complementary simplification has to be made : all the areas are calculated one by one and for each of them alternatively, the slowing down in the other areas is considered to be the same as in the area for which the calculation is under progress. This a very strong approximation which may be wrong specially when a WR model is used. The Background Matrix formalism[6] of APOLLO-2 avoids this difficulty by dealing with all the self-shielding areas altogether : no more simplification is requested.

Possibility of using an iterative process :

In the fuel rod there is a mixture of resonant isotopes ; today it is not possible to directly calculate the self-shielding of the mixture. Each of the resonant isotope of the mixture is calculated separetely the other ones being considered as moderators. In APOLLO-2 it is possible to start again the self-shieling process in using the muligroup cross sections calculated in the previous self-shielding process ; it is a way to partially take into account the interference between resonances of two different isotopes. This possibility was not existing in APOLLO-1

Possibility of using modularity :

Previous improvements discussed here make the code much more precise but also more expensive ; as an example of the enhanced capabilities of the code due to the modularity the diminution of the calculation costs is very easy : instead of using an assembly geometry for self-shielding calculations a fuel cell geometry can be used. Thus, time calculation requested for self-shielding is decreased from 400 s to 1 s (on a CRAY XMP) and after the self-shielded multigroup cross-sections may be used in an assembly geometry description for a muticell or 2D "exact" flux calculation .

Qualification of the self-shielding module of APOLLO-2 :

In APOLLO-1 the qualification was restricted to the global absorption in the total range of energy ; it was quite normal because there was only one calculation point inside the fuel and because the self-shielding energy range was treated as a whole. For APOLLO-2 and its new capabilities a more detailed analysis was worth doing.

In order to check the validity of the new APOLLO-2 self-shielding module a reference is needed. It is impossible to rely on experiments because in an experiment the whole range of energy has to be taken into account and not only the range where self-shielding exists. So we have to build a numerical experiment. That means the reference will be an other calculation using no model for dealing with the self-shielding. In fact we use Monte-Carlo code TRIPOLI[7] and slowing-down reference code SECOL[8]. Extensive comparisons showed a good agreement between APOLLO-2 models and reference calculations. The analysis was performed for the most important nuclei and for a fuel rod divided into ten equivolumetric rings ; for the whole energy range, the absorption rates inside the rod and inside each ring are quite satisfactory. Only some small discrepancies appear when the radial absorption shape is analyzed in some specific groups[9].

Consequences of the self-shielding improvements :

Before, the search of tendancy[10] was made by using APOLLO with a 99 groups library and its self-shielding capability. In this work, there was only one calculation point in a fuel rod, because the self-shielding model of APOLLO-1 was previously qualified in this condition and for the total absorption. Recently the same analysis was made again in order to validate JEF2 data. In this case APOLLO-2 was used with a 172 groups library, its best self-shielding model (group per group with ST model above 50 eV and WR one below) and six equivolumetric rings for describing the fuel element. The results got in this way show two very important results[11] : the JEF2 evaluated data files need no corrective factor and in particular the former trend to diminish the effective resonance integral has vanished. Rather than a consequence of the move in the group number this is a consequence of having a better modelization of the self-shielding ; so it is very satisfactory to dispose at the same time of a good evaluation and a good self-shielding model.

Consequently it was interesting to evaluate the impact of the self-shielding model on the calculation of the Plutonium balance problem which originated the developments discussed above. The preliminary results have been obtained by comparing two cell depletion calculations. The cell is a standard PWR one with 3.25% enriched Uranium. The only difference between the two calculations is the discretization of the fuel rod : in the first there is only one point and in the second the rod is divided into 10 concentric equivolumetric rings.

Table 1 : Comparison on the concentrations (10^{24} n/cm3) of the various plutonium isotopes at 36000 MWd/t

Isotopes	Calculation with 1 ring	Calculation with 10 rings	discrepancies (%)
Pu238	3.2438.10-6	3.2730 10-6	0.90
Pu239	1.2732 10-4	1.2399 10-4	-2.61
Pu240	5.3036 10-5	5.1859 10-5	-2.21
Pu241	3.2942 10-5	3.2559 10-5	-1.16
Pu242	1.3688 10-5	1.3835 10-5	1.07
Pu total	2.3023 10-4	2.2552 10-5	-2.05

This table shows an improvement as the total amount of plutonium is reduced of about 2%. when the number of rings is increased from one to ten The reduction is slightly higher (2.6%) at 60000 MWd/t. To this effect we have to add the difference between APOLLO and APOLLO-2 wich is about one percent. So the use of APOLLO-2 with ten rings instead APOLLO-1 with only one ring can explain about the half of the sought effect.Aniway, it is only a preliminary examination and other aspects have to be examined carefully : here the self-shielding calculation was performed at the step zero only as it is done in standard route using APOLLO-1 ; calculate again the self-shielding at each step, is quite easy with the modularity of the APOLLO-2 and could improve the results. An other important aspect which is not taken into account here, is the temperature distribution inside the rod. In order not to mix many effects the two calculations were performed with the same temperature 650.°C. In fact when there is only one ring, only one temperature can be used and in fact, the shape of the temperature distribution is taken into acccount through an effective temperature. When several rings can be used it becomes interesting to give the actual temperature shape to the self-shielding module. As a consequence it will be useful to include a thermomecanichs module inside SAPHYR. This development is planned and will allow to fully benefit from the improvements made in the self-shielding modelization.

APOLLO-2 ; DEPLETION CALCULATION IMPROVEMENTS

Multigroup library :

In APOLLO-2 the basic idea was to collect all the informations needed by the code and to regroup them in the same place, the APOLIB-2, which is the multigroup library of APOLLO-2. As in APOLLO there are multigroup cross-sections, but the difference is, that matters related to depletion calculations,like yields, n-2n cross-section,fission energies,... are also included. It is much more convenient and coherent : for example, it avoids using a different n-2n cross-section in the reactivity and in the depletion calculations. Furthermore new potentialities have been added : the yields and the fission energies can have different values according the neutron energy ; today this improvement is not used in APOLLO-2 because, in order to qualify the depletion module and to ensure the continuity with APOLLO the informations are kept exactly the same (even if their location is not) ; and in APOLLO the depletion equation can be dealt with only one energy group. Later it will be useful because with this improvement it will be no more necessary to have different values of one group yields depending on wether an Uranium or a MOX assembly is processed. An energy dependent spatial approximation may also be used in depletion calculation[17].

An other difference is that the APOLLO-2 library may be used by its SN calculation module, so any anisotropy development is supported, while in APOLLO only P0 and P1 developments can be stored.

Today two standard 99 and 172 groups library exist for APOLLO-2. The content of the 99 groups one is quite identical to the APOLLO-1 validated library.. This makes easier intercomparisons between the two codes. Both of these libraries are generated by the THEMIS-NJOY system.

Analysis of the depletion process.Checking of the precision :

Due to the modularity of the code, the depletion calculation function was splitted in several sub-functions carried out by different modules. One of this module is in charge to extrapolate the depletion matrix and to perform the calculations of the nuclei concentrations at the next step. This is very close to what APOLLO was performing. But the difference is that the process is not yet finished in APOLLO-2 : an other module checks the results validity. By using the new concentrations it interpolates the matrix depletion and determine other concentrations. It creates a logical GIBIANE value, which is true if the concentrations calculated by the two modules are close enough, and false in the other case.

Corrective actions :

Of course, if the GIBIANE value is true, the depletion process may go on. But if the value is false, the user may decide to undertake a corrective action. He can write in advance a GIBIANE procedure which will be activated as soon as such a problem will be detected. The user is free to define its own procedure but, in this case the most commonly used procedures consist either to come back to the initial step and to divide by two the length of the step or, to iterate on the final concentrations in keeping the same length. So, even if a too large step was specified by the user, the code will perform an automatic correction allowing the depletion process to go on satisfactory.

APOLLO-2 : OTHER MAJOR IMPROVEMENTS

Collision Probabilities :

Three important features, which did not exist in APOLLO, have been developped in APOLLO-2. In the case where a 2D direct method (direct integration on the entire geometry) is performed, an approximate treatment of the reflexion at the external side of the geometry can be used : all the neutrons reaching the external surface are reinjected with the assumption tthat heir distribution is uniform on the surface and their direction is isotropic. This process is very time sparing and is only changing scarcely the precision of the results.

The second one is an extension of the 2D exact calculation. Formerly the geometry had to be a XY one (of course cylindric cells could be located inside the rectangular cases). Now, any kind of 2D geometry can be processed provided that the separation between the flat-flux region be either an arc of a circle (or an entire circle) or a straight line segment.

The third one can be used in the framework of the multicell method. It is the possibility of using a flux-current method instead of the standard one where currents are eliminated in order to let a system of equation whose the only unknowns are the fluxes. That method was proved to be very performing and specially when the flat-flux region number is important.

Discrete Ordinates Method :

Using exactly the same libraries isotopes,media or geometry objects created by the same modules as in the case where the PIJ formalism is used, a module performs the heterogeneous flux calculation by using discrete ordinate solution of the integro-differential transport equation. The perfect integration of this method inside APOLLO-2 allows to use also the module carrying out the equivalence process. The equivalence factors necessary for calculating a reactor core with a SN method can be easily got in APOLLO-2.

Leakage :

Besides the well-known volumetric leakage, already used in APOLLO, and consisting in distributing uniformly on the whole volume of the geometry the leakage as an extra absorption , an other model, the surface leakage one was implemented in APOLLO-2. The idea is to modelize leakage by a term of incoming current on the outer surface. This may seem more accurate from a physical point of view in the sense that the leakage takes place at the surface of the geometry and not inside. This was proved to have an effect of about 3 % on the contol rod efficiency calculated by APOLLO-2 [14]

Cell leakage coefficients :

In order to well calculate the voiding effects and to take into account the effect of the heterogeneity of an assembly on neutron leakages, a new method[16] using the directional first flight collision probabilities was introduced and is under validation by comparison with experiments.

Portability :

One of the basic idea of SAPHYR system was to be able to be run on a large number of different computers. Today APOLLO-2 is used with CRAY-1 (XMP and YMP), CRAY-2 and many workstations as SUN 4, IBM RISC 6000 and HP.

CONCLUSIONS

Many improvements make APOLLO-2 still more powerful than APOLLO-1. Among the very large range of its possibilities, it is always possible to find a solution which fits the user's wishes, and optmizes the precision/cost ratio. Its integration inside the SAPHYR system and its fully modularity enhance its capabilities and let open the place for many future improvements.

APOLLO-2 may be considered as a tool box, and with GIBIANE's help, the user may build up a scheme, perfectly suited to his need. As in APOLLO-2 very different tools are present for doing the same work, a calculation time may expand from a few seconds to several hours, to deal with a PWR assembly, according the requested degree of precision. Anyway, the presence of best-estimate options are very useful for defining an industrial scheme.

So through its many capabilities APOLLO-2 is suited well for in-core fuel management requirements. In this field it may be used for PWR, BWR or experimental facilities. Furthermore if the consequences of a simplification of the routine scheme have to be examined it is quite easy to upgrade the scheme inside APOLLO-2 itself and to analyse it from the double points of view of cost and precision comparing to a more sophisticated one (best-estimate if necessary).

Apart from this, the code may also be used in the field of criticality analysis : having a build-in SN module (with finite difference and nodal method) and the capability of determining the equivalence coefficients between heterogeneous PIJ transport calculation and homogeneous SN transport calculation, it may be very useful in this area.

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METHODS FOR RELOADING

(Session 3)

DEVELOPMENT OF THE FAST STOCHASTIC LOADING PATTERN OPTIMIZATION METHOD

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Abstract

Stochastic optimization methods based on the simulated annealing algorithm have become very important in the recent years for solving difficult optimization problems. Application of these methods for in-core fuel management optimization problems is still at the beginning.

In this work we propose an algorithm which is expected to be more efficient than the classical simulated annealing algorithm in obtaining near-optimal solutions for the loading pattern optimization problem. The algorithm combines an enhanced simulated annealing cooling schedule with the heuristic solution generator.

Preliminary results obtained by solving difficult integer programming problems, confirmed the improved efficiency over the pure simulated annealing method. The basic version of the loading pattern optimization code has been developed by coupling the optimization algorithm with the 1.5-dimensional core depletion simulator. The new algorithm and methodology, the structure of the related computer code(s), as well as the initial results are discussed here.

1. Introduction

Theoretical studies of stochastic optimization date back to fifties and early sixties, but its practical use really began in the last decade, at the time when modern computers were able to match high requirements in number of repetitive calculations typically involved in random simulations. One of the most used stochastic techniques last ten years is simulated annealing (SA), first proposed by Metropolis et al. [1], and then rediscovered by Kirkpatrick et al. [2]. SA exploits an analogy between optimization systems and physical systems. Slow annealing of a real physical should bring it into its state of equilibrium with the ambient temperature T and thus for $T->0$ the system moves into its ground state(s). Similarly, the proper simulation of this procedure treating an optimization problem as a physical system should result in the simulation finding the optimal solution.

To escape from the local minima in the SA algorithm a stochastic acceptance criterion, so called Metropolis criterion is employed. The novelty of Metropolis criterion is that it may accept a new configuration of higher cost than the previous one. The probability of acceptance of the new configuration S' is:

$$P(S') = \exp(-\frac{\Delta F}{T}) \quad , \quad if\ \Delta F>0$$
$$= \quad 1 \quad , \quad otherwise \qquad (1)$$

where T is a parameter which has the same role as the temperature in a physical thermodynamic system. ΔF is F(S')-F(S), i.e. the difference between values of the objective funciton for the new configuration S' and the old configuration S. For high T all configurations are almost equally probable, but by decreasing T one can reduce the number of accessible configurations until the algorithm freezes in a low-cost configuration.

This behaviour was extensively exploited in solving difficult, especially combinatorial problems, like "traveling salesmen problem" and integrated circuit design. Recently, SA was applied for loading pattern optimization [3]. However, classical simulated annealing as used in [3], combined with accurate reactor core depletion modelling, still requires enormous amounts of computer time if near-global-optimum solution is searched for.

In our attempt to make an efficient loading pattern optimization tool, we have combined the SA cooling schedule developed by Aarts and van Laarhoven [4], with a heuristic solution generator, which is based on works of Schumer and Steiglitz [5] and Parks [6]. We adapted this heuristic tool for integer programming problems. To test and quantify behaviour of a newly designed algorithm we analyzed three constrained integer programming problems which were used as tests in literature [7]. Satisfactory results confirmed our starting assumption that this algorithm could be used as an efficient loading pattern optimizer.

In this paper we describe the structure of this algorithm and results obtained by solving test problems, as well as its adapted version for loading pattern optimization and preliminary results obtained in loading pattern optimization.

2. Methodology

Main part of the new algorithm called ANNEAL, is the simulated annealing cooling schedule given in [4]. Two other important features are adaptive step size search algorithm (or so called heuristic solution generator), and penalty function method. Heuristic solution

generator represents improvement of the classical simulated annealing in the sense of getting the same quality of the solution in much shorter time, while penalty function method practically enables algorithm to treat any kind of optimization problem. A short description of each of these features follows.

2.1. Cooling Schedule Parameters

Every cooling schedule has essentially the same goal; definition of the speed of cooling so that the request for equilibrium distribution of states in the physical system (or solutions, in optimization problems) at each temperature is obtained. Practically this means that one has to define a finite sequence of values of the control parameter (equivalent of the temperature in physical systems), and a number of transitions at each value of T. Even with very simple cooling schedules most of the advantages of the SA can be revealed, but the one that is used in the ANNEAL algorithm has the advantage of getting automatically some of the parameters necessary for performing SA.

First, it concerns the determination of the initial control parameter value T_0, which depends on the actual objective function, which is different from problem to problem. In the ANNEAL algorithm user specifies only the solution acceptance ratio which is wanted at the beginning, χ_0, and the code gets the temperature T_0 using so called "melting" procedure.

The rate of the temperature decrease is regulated through the user specified parameter δ, but the code also changes the pace according to the distribution of the solutions at each temperature. If the spread of values of the objective function at the certain value of control parameter is bigger than the one before, temperature drop slows down, allowing the algorithm to explore more solutions close to the particular value of T.

Final value of the control parameter T is defined by two stopping criteria; either the averages of the objective function at the last consecutive values of T are close enough, or there is no transitions to new solutions at all, at the last consecutive values of T. Also, there is a time limit criteria in order to prevent too long runs.

A maximum number of "successful" transitions L_0, at each value of T, and a maximum number of the points allowed to be searched N_{LIMIT}, at each value of T, are given by the user. The role of the parameters described above is more clearly depicted in Figure 2, which shows the flow diagram of the ANNEAL algorithm.

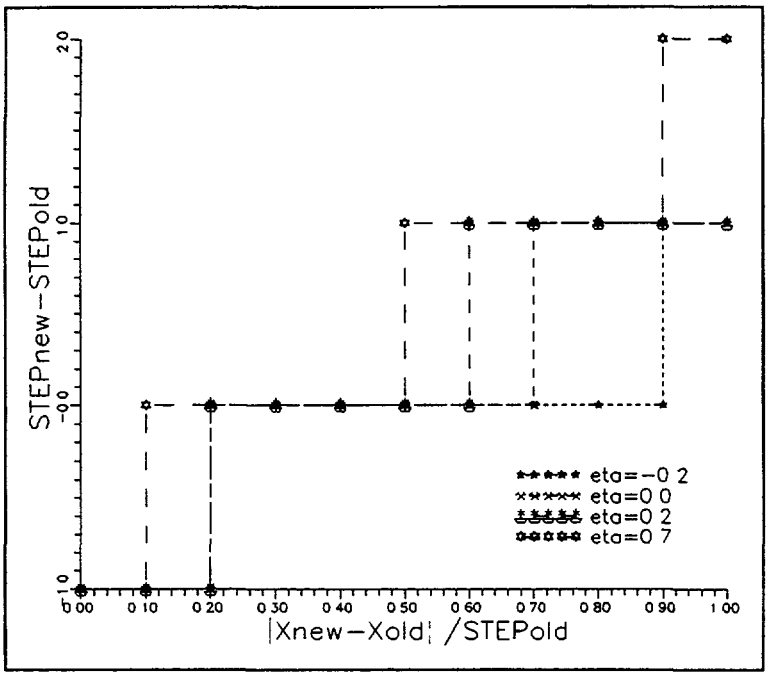

Figure 1. Illustration of the heuristic transition generator behaviour. Step size change depends on the point to which the next successful transition is made and the value of user given parameter η.

2.2. Adaptive Step Size Search - Heuristic Solution Generator

This part of the algorithm has an important feature of localizing the transitions from point to point, to areas where the probability for improving the value of objective function is higher. Given an old value of variable X_j^{old}, a new value is calculated by:

$$X_j^{new} = X_j^{old} + random[-1,1] \cdot ST_j^{old} \qquad (2)$$

where random[-1,1] is a random number between -1 and +1, and ST_i^{old} is a maximum transition step size associated with variable X_j. After each successful transition to a new point (value) X_j^{new}, step size is updated according to:

$$ST_j^{new} = a_1 \cdot ST_j^{old} + a_2 \cdot |X_j^{new} - X_j^{old}| \qquad (3)$$

where

$$a_1 = 1 - 1/ST_j^{old}$$

$$a_2 = (2+\eta)/ST_j^{old}$$

Algorithm ANNEAL

1. Determination of T_0 ("melting the system"), according to input parameter χ_0
2. If $N_{SUCC} \leq L_0$ and $i \leq N_{LIMIT}$ then

   make transition from $\mathbf{X}^*$ to $\mathbf{X}_i$,

   ($X_{i,j} = X_j^* + \text{random}[-1,1] \cdot ST_{i,j}$) ;

   Otherwise; Go to step 4.
3. If $\text{random}[0,1] \leq \exp \{ - [F(\mathbf{X}_i) - F(\mathbf{X}^*)] / T_i \}$ then

   $\mathbf{X}^* = \mathbf{X}_i$;

   If $F(\mathbf{X}_i) \leq F(\mathbf{X}^{BSF})$ then $\mathbf{X}^{BSF} = \mathbf{X}_i$;

   Change step size,

   $ST_{i+1,j} = a_1 \cdot ST_{i,j} + a_2 \cdot | X_{i,j} - X_j^* |$;

   $N_{SUCC} = N_{SUCC} + 1$;

   $i = i + 1$

   Go to step 2 ;

   otherwise;

   $i = i + 1$

   Go to step 2 ;
4. If $| [F_{av}(T_i) - F_{av}(T_i\text{-}1)] / F_{av}(T_i) | \leq \in_f$ then

   Go to step 5 ;

   otherwise;

   new temperature $T_{i+1} = T_i / [1 + \ln(1 + \delta) / 3 \cdot \sigma_{Fav(Ti)}]$;

   Go to step 2 ;
5. Final result $\mathbf{X}^{BSF}$ and $F(\mathbf{X}^{BSF})$.

Figure 2. Structure of the ANNEAL algorithm.

Figure 1 illustrates behaviour of the generator, based on the equations (2) and (3). Before starting the algorithm one also has to specify lower and upper limit for step size for each variable X_j:

$$ST_j^{min} \leq ST_j \leq ST_j^{max} \qquad (4)$$

Through calculations it was found that the performance of the algorithm depends much more on the ST_{min} than on ST_{max}. Practically, transition generator changes step size according to a local topography of the objective function.

2.3. Penalty function method

The form of the objective function used in the ANNEAL algorithm is:
Minimize

$$F(\boldsymbol{X}) = f(\boldsymbol{X}) + k \cdot C(\boldsymbol{X}) \tag{5}$$

where

$$C(\boldsymbol{X}) = \sum_{i=1} | c_i(\boldsymbol{X}) - b_i | \cdot \delta_i$$

k - constant given by user (depends on particular problem);

δ_i - 1 for constraints that are violated and 0 for nonviolated constraints;

f(**X**) - objective function that is optimized;

$c_i(\mathbf{X}) \leq b_i$ - ith constraint.

The meaning of the terms in the algorithm is following:

T_l - lth value of the control parameter;

$F(\mathbf{X}_i)$ - value of objective function at the point $\mathbf{X}_i$;

$F(\mathbf{X}^*)$ - minimum value of the objective function at the point $\mathbf{X}^*$;

$F(\mathbf{X}^{BSF})$ - "best so far" value of the objective function at the point $\mathbf{X}^{BSF}$;

$F_{av}(T_l)$ - average value of the objective function for a single value of control parameter T_l;

$\sigma_{Fav(Tl)}$ - standard deviation of objective function for the lth value of control parameter T;

L_0 - upper limit of the number of successful transitions;

N_{SUCC} - number of successful transitions so far for a single value of T;

N_{LIMIT} - maximum number of explored transitions for a single value of T;

i - number of transitions explored so far for a single value of T;

j - variable index;

$ST_{i,j}$ - step size for the variable j in transition i.

3. Test problems and results

Three test problems given by Dickman and Gilman [7] were used for testing the algorithm's efficiency. We will emphasize here only important findings and illustrate them

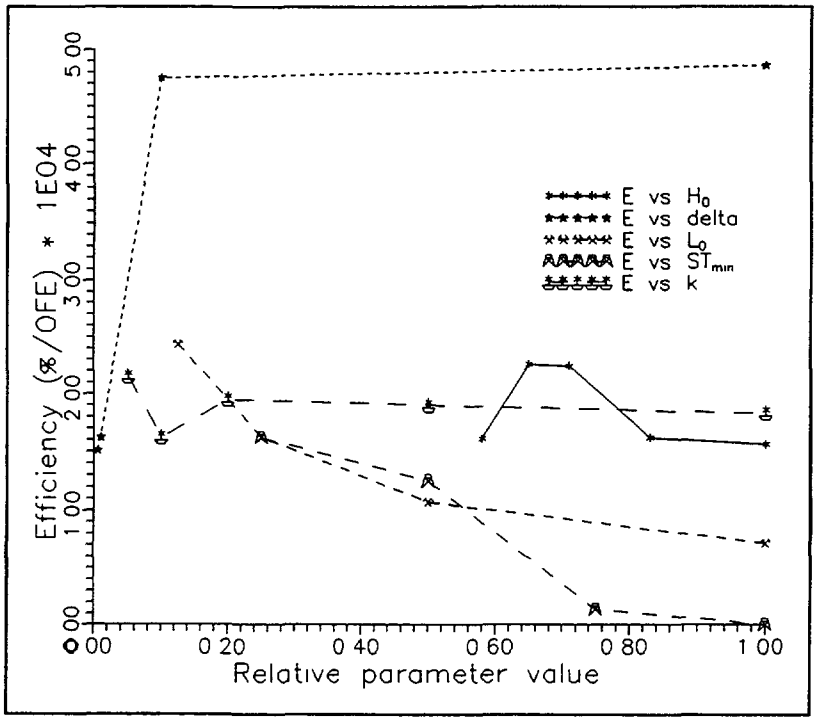

Figure 3. Efficiency of the ANNEAL algorithm vs. relative value of different parameters, expressed in percent. of optimal solutions per objective function evaluation (%/OFE). Each curve represents effect of one parameter on E while others are kept constant.

on the results obtained for one of the problems, which is by its features close to the loading pattern optimization. Problem itself involves minimization of a cubic function of 8 integer variables, on which 10 constraints are imposed. Possible number of combinations is 1.342×10^8, total number of feasible solutions is only 42, and there is only one optimum.

Intervals of input parameters that give reasonable convergence of the algorithm were determined by trial and error. After that, the effect of the change of each parameter on the behavior of the algorithm was studied by keeping all other parameters unchanged. In order to quantify behaviour of the algorithm, two values were monitored. First is the average quality of the final solution given by:

$$q = n_{opt}/N \cdot 100[\%] \qquad (6)$$

where q is a percentage of correct (optimal) solutions. N is the total number of independent runs of the algorithm solving the same problem, and n_{opt} is the number of runs that ended with optimal solution.

Second value important for evaluating efficiency of the algorithm is the average number

of points explored, np (number of objective function calls), while solving the problem with the same input parameters. Since we are dealing with random process, actual results for this two quantities were averaged over a greater number of runs (N=30), for the same problem instance, using different initial starting points, and random number sequences. Using this two values we can define efficiency as:

$$E(\chi_0, \delta, L_0, ST_{min}, k) = q/np \qquad (7)$$

Figure 3 illustrates variation of the efficiency E versus each of the parameters while others are fixed. It is obvious that χ_0 and k do not affect efficiency significantly, compared to other three parameters. Especially important is the effect of ST_{min}, which for larger values drastically lowers the efficiency. One has to bear in mind that although efficiency is very high for high δ values, it is obtained for ST_{min} fixed to the smallest value. The effects of δ and L_0 are similar to those in pure SA algorithms, only that overall efficiency is much higher due to the use of heuristic solution generator.

4. MOCALPS - loading pattern optimization code

MOnte **CA**rlo **L**oading **P**attern **S**earch is the name of the loading pattern optimization code that has as an optimization core ANNEAL algorithm described in the first part, and as a core neutronics simulator MCYC1.5D [8], one-and-a-half dimensional diffusion code developed

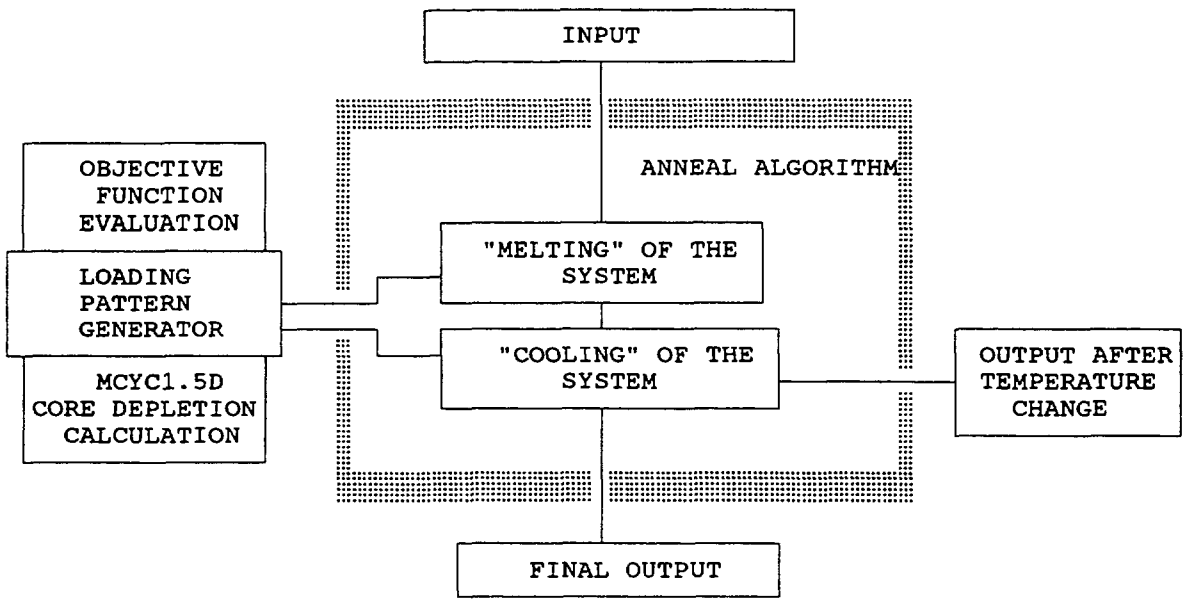

Figure 4. Structure of the MOCALPS code.

for fast scoping fuel management studies. The structure of the MOCALPS code is depicted in Figure 4.

MCYC1.5D was modified to suit the optimization process, while loading pattern generator and objective function evaluator are new parts added to the scheme of the ANNEAL algorithm. In the following sections MCYC1.5D and other parts of the MOCALPS code, apart from the ANNEAL algorithm, are described in short.

4.1. MCYC1.5D Core Neutronics Simulator

In the MCYC1.5D model diffusion equation is solved in two energy groups and one-dimensional geometry, but the results are unfolded to produce the two-dimensional flux/power map. The 1½D calculation starts by determining cross sections for individual fuel assemblies (FAs) which are then averaged over the one-dimensional regions (annular rings). The one-dimensional diffusion equation is solved, producing the one dimensional flux/power distribution map. The innovation of the 1½D model is in the multistep unfolding procedure, which reconstructs the two-dimensional power map. The procedure accounts for all the major physical differences between the cylindrical one-dimensional geometry and the two-dimensional x-y geometry:

1) The average normalized power (NP) within a one-dimensional ring is first split into the two-dimensional NPs for individual Fas based on the ratio of the average and individual k_∞s.

2) The radial importance correction is applied, based on the radial position of the fuel assembly in reference to the two-dimensional position.

3) The difference in local topologies, i.e., the difference between relations of the neighboring rings in the one-dimensional geometry and the neighboring Fas in the two-dimensional geometry is considered next. It leads to the correction, which accounts for the local relation between adjacent fuel assemblies.

4) The peripheral FAs vary among themselves in their neutron leakage, whereas they are treated as equivalent in this respect in the one dimensional geometry. A correction is introduced by an albedo-type relation, to account for such differences.

The unfolded two-dimensional power map obtained by this process is reffered to as the 1½D power map to distinguish it from the results of the standard two-dimensional calculation. The computer code MCYC1.5D incorporates the described model and has also multicycle analysis capabilities similar to the two-dimensional MCRAC code [9]. It employs the same technique of storing and reconstructing the cross sections, which are generated by PSU-LEOPARD code [10].

The code was tested using the NPP Krško data. Several cycles were analyzed. Same cycles were analyzed using the 2D MCRAC code, and the results were compared. Critical soluble boron concentration was always within ± 30 ppm. The highest normalized FA power throughout the cycle differed typically by less than ±10-15%, being somewhat higher for peripheral Fas (which are typically not very important). Difference in EOC burnups for individual Fas was within ±10%. MCYC1.5D is about 30 times faster than MCRAC code.

4.2. Loading Pattern Generator

Loading pattern generator in MOCALPS code incorporates heuristic solution generator and basic logic of filling the core positions with fuel assemblies (Fas). The FA inventory for the optimized cycle is divided in input in groups of 4 and 8 fuel assemblies of same (or similar) neutronic characteristics. The same group cannot be used twice. In case that random process samples an already used FA group to another position, this position is filled with the neutronically closest available FA group. In the loading pattern generator, at the moment, no splitting or merging of used FA groups of 4 or 8 FA which are of the same characteristics is allowed. Also, no other restrictions are imposed in the loading pattern generator, which would further lower the number of possible combinations (loading patterns). If we allow that any FA group can be used on any location, for 1/8 of the NE Krško core this means - at least $(10!)^2$ possible loading patterns (with fixed central position).

4.3. Definition of the Objective Function

It is a difficult job to define the objective function on a single cycle optimization basis. Since ANNEAL algorithm accepts any type of the objective function, we have decided to form the objective function that will be able to represent different designer's goals when

designing loading pattern. In practice, some kind of multicycle scoping analysis or optimization is performed before actual design of the following cycle. This means that most of the important cycle parameters are approximately defined, i.e.: cycle length, number of fresh FA, enrichment of fresh FA, approximate number of burnable absorber rods. In that situation designer's goal is to find a definite loading pattern which is close to defined parameters and satisfies safety constraints.

Even then, there is some space left for optimization. Definite loading pattern can satisfy cycle length, using predetermined number of fresh fuel assemblies, but can use fuel of lower enrichment, or transfer higher reactivity fuel to the following cycle by maximization of burnup for the discharged fuel, therefore minimizing costs for the cycle or leaving some space for saving in future. Also, low leakage can be maximized and number of BAs minimized as well. In the current version of MOCALPS code this goals are combined in a following expression:

$$F(S) = -A_1 \cdot BU_d(S) - A_2 \cdot k_{eff}^{EOC}(S) + A_3 \cdot NFA_{frsh}(S) + K \cdot \sum_i CNS_i(S) \qquad (8)$$

where S represents particular loading pattern, BU_d average discharge burnup, k_{eff}^{EOC} core effective multiplication factor at EOC, and NFA_{frsh} number of fresh fuel assemblies. Last term in equation (8) stands for the sum of constraint penalties. Following penalty functions are involved:

Peaking factor penalty

$$CNS_1(S) = \frac{c_1 \cdot \sum_i (NP_i - NP_{max})}{\sum_i 1}, \qquad \forall i,\ NP_i > NP_{max}$$

Discharge burnup limit

$$CNS_2(S) = c_2 \cdot [BU_d(S) - BU_{max}], \qquad if\ BU_d(S) > BU_{max}$$

MTC or too high boron concentration at BOC

$$CNS_3(S) = c_3 \cdot [cB^{BOC}(S) - cB^{max}], \qquad if\ cB^{BOC}(S) > cB^{max}$$

5. Results

MOCALPS has been tested on the NPP Krško cycle 7 optimization. In Figure 5 real cycle 7 fuel assembly distribution is shown in 1/8 core representation. Weight factors from the

3.4 22826	3.8 13227	3.4 24840	3.8 0 8BPR	3.2 26057	3.8 0 4BPR	3.2 24932
	3.4 24824	3.8 10707	3.4 26154	3.4 13960	3.8 0	3.4 12804
		3.4 12781	3.8 12508	3.35 26427	4.3V5 0	
			3.4 14407	4.3V5 0	3.4 15412	
				3.35 28384		

Figure 5. Real cycle 7 loading pattern design - 1/8 of NPP Krško core.

3.4 22826	3.4 14407	3.4 24840	3.4 24824	3.8 0 4BPR	3.2 24932	3.8 13227
	3.35 28384	3.8 0 8BPR	3.35 26427	3.4 26154	4.3V5 0	3.4 15412
		3.2 26057	3.4 12804	3.8 12508	3.4 13960	
			3.4 12781	3.8 0	3.8 10707	
				3.8 0		

Figure 6. NPP Krško, cycle 7, final result of MOCALPS optimization.

expression for the objective function (8) A_1-A_3 are given values derived from very crude analysis, in which A_1*BU_d represents the return of the investment for the fuel. Assumption is that by burning fuel to the maximum burnup one completely returns money invested in fuel. Cycle length (or precisely k_{eff}^{EOC}) can also be expressed in the same units, using factor A_2, which contains in itself, a ratio of the average core burnup change per reactivity change at EOC, for typical LP. A_3 is the value of fresh fuel assembly. Therefore, objective function can be expressed in dollars. Of course, in this crude model, only relative change of objective function is what we have compared.

Table 1. Comparison of real cycle 7 loading pattern and the one obtained with MOCALPS.

Parameter	Real Cycle 7	MOCALPS Cycle 7
Cycle length (EOC ppm)	30	-21
Discharge burnup (MWd/tU)	35642.	37192.
Max. peaking factor	1.38	1.414
Number of fresh FA	16 x 3.8 w/o 16 x 4.3 w/o	24 x 3.8 w/o 8 x 4.3 w/o
Number of BPRs	48	80
OF[1] (10^3\$)	0 (ref.)	-326

[1] OF - Objective function value according to the equation 8.

Values of constraint penalty factors K, and c_1-c_3 can be put very big, but smaller values can help the algorithm to converge more quickly and to better minima.

The particular test run was made on SG-Crimson UNIX workstation, with a time limit of a 1 hour CPU. About 10000 loading patterns were searched. Figure 6 shows final "best so far" LP after convergence of the algorithm was stopped due to the time limit. Table 1 gives comparison between two LPs in cycle parameters.

6. Conclusions

A new stochastic optimization method for solving integer programming problems of higher dimensions has been built and tested. Its efficiency is significantly higher than efficiency of classical simulated annealing method. The optimization algortihm is coded into a new loading pattern optimization code called MOCALPS, together with a 1.5 dimensional core neutronics simulator. Preliminary results confirm expectations from the testing phase.

Future efforts will concentrate on improvements of loading pattern generator, objective function formulation and on further experimentation with parameters that govern the convergence of the algorithm.

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WESTINGHOUSE FUEL ASSEMBLY DESIGN EVOLUTION

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Abstract

Pressurized Water Reactor fuel assembly designs have evolved significantly over the last decade, as a result of utility needs for enhanced fuel economic, margin improvement, reliability, and operational flexibility. Today's design, VANTAGE 5, already incorporates significant operating and irradiation experience. The core design impacts on reactivity and power distribution control due to longer cycles and/or high discharge burnup fuel management are reviewed. The VVANTAGE 6 fuel design for VVER reactors is also described. Finally, Westinghouse fuel product direction towards meeting even more stringent operational requirements for the 1990's is also discussed.

1. DESIGN EVOLUTION

Fuel development at Westinghouse is a continuous process that builds upon the experience of previous designs and the current needs/drivers. Implementation of advanced fuel concepts over the last few years has contributed to a significant improvement in fuel performance with marked reduction in primary system coolant activity, while improving fuel utilization in PWRs. Several new fuel designs have been introduced since the 1970's. The Optimized Fuel Assembly (OFA) introduced in 1977 met the needs of improved economics and was designed for discharge burnups of 36,000 MWD/MTU. In 1983, the VANTAGE 5 design combined features to address needs on performance, margin, and economics[1,2].

VANTAGE 5 fuel incorporates the following features, including zirconium diboride integral fuel burnable absorbers for enhanced power distribution control and improved economics; intermediate flow mixers for increased thermal-hydraulic margins; axial blankets for improved neutron economy; assembly modifications to increase discharge burnup for improved fuel utilization and availability; removable top nozzles for ease of fuel rod inspection and reconstitution; and debris filter bottom nozzles to decrease the probability of debris-induced fuel rod damage.

Integral Fuel Burnable Absorbers

Longer cycles require installation of large amount of excess reactivity at the beginning of a cycle with highly enriched U235 fuel. Burnable absorbers are then required to control power distributions and moderator temperature coefficients to meet safety limits.

The Westinghouse Integral Fuel Burnable Absorber (IFBA) incorporates the absorber material directly into the fuel rod in the form of a thin coating of zirconium diboride on the pellet surface. IFBA rods provide a smaller residual reactivity penalty than other burnable absorbers. This burnable absorber also improves design flexibility since IFBA rods can be placed anywhere in the core, even under control rod locations, while eliminating the handling of separate components during refueling and fuel storage operations.

With IFBAs, requirements for fuel enrichment are reduced, resulting in an up to 3 percent reduction in fuel cycle costs. If enrichments are held constant, more efficient core designs are possible to achieve higher burnups.

Intermediate Flow Mixer

Intermediate Flow Mixer (IFM) grids enhance the margin to departure from nucleate boiling (DNB) by increasing the flow mixing and turbulence in the upper part of the core. DNB tests have shown that IFM grids increase margin 25 percent.

The implementation of IFMs provides the opportunity for use of this added DNB margin in several ways. For example, it can be used to increase peaking factor limits. Such an increase supports more efficient fuel management strategies with a resultant fuel cycle cost improvement as compared to assemblies without IFMs. In addition, added margin can support plant operation and enhancements or compensate for operating simplifications.

Axial Blankets

Axial Blankets use fuel pellets made of natural rather than enriched UO_2 rather than enriched to create 6 inch zones at both ends of the fuel rods. This forms a blanket across the top and bottom of the fuel, reducing the number of neutrons leaking from the core and thereby improving fuel cycle costs.

Higher Burnup

Discharge burnups have steadily increased over the years. Product enhancements incorporated in VANTAGE 5 fuel address several factors that allows operation in the range of 40,000 MWD/MTU region average discharge burnup.

This increased capability has been accomplished by modifying the overall fuel assembly to provide extra space for fuel rod growth and increased fission gas release resulting from lengthy operation. Top and bottom nozzles have been enhanced to permit a thickness reduction, and a repositioning of the bottom nozzle results in a net increase in space for fuel rod growth during irradiation. In addition, the fuel rods themselves have been lengthened to increase plenum space.

Debris Filter Bottom Nozzle

Eliminating debris-induced fuel damage is a key requirement in enhancing fuel reliability in modern designs. Debris in the coolant can become lodged between the fuel rods and a grid, leading to damage of the cladding by fretting. This mechanism is recognized to cause 70 to 80 percent of the leaking fuel rods since 1983. The Debris Filter Bottom Nozzle (DFBN) uses a pattern of many small holes to reduce the passage of debris into the core. The revised hole pattern has been determined to provide the same hydraulic performance as previous bottom nozzles.

In conjunction with improvements in operating and maintenance practices during reactor outages, the DFBN significantly reduces the probability of debris-induced fuel damage. Therefore, increased fuel reliability can be achieved, which lowers maintenance costs, and helps avoid the possible availability losses that could result from increased activity levels.

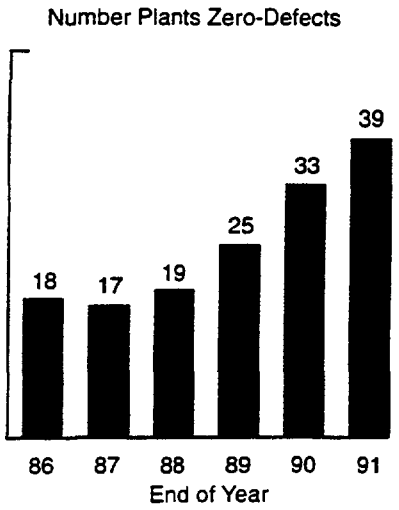

Figure 1 - Westinghouse Fuel Performance Experience

Table 1 - VANTAGE 5 Experience through 1991

Feature	Plants	Regions	Number of Assemblies	Committed Plants
Axial Blankets	22	52	2,900	29
IFBAs	27	50	3,060	36
IFMs	12	21	1,630	17
RTNs	49	113	7,000	53
High Burnup	35	71	4,450	43
DFBNs	44	75	4,920	50

Removable Top Nozzle

A removable top nozzle (RTN) permits easy access to fuel rods, enabling fuel inspection and/or fuel repair. Guide thimbles are attached to the top nozzle with lock tubes that can be removed using specially-designed tools. This nozzle is fully compatible with existing core and plant interface equipment. This feature, although it may never be used, can provide significant savings if used to repair/reconstitute fuel.

VANTAGE5 Feature Experience

These advanced design features have provided significant fuel cycle cost benefits, while improving overall fuel reliability and integrity. Overall activity in the reactor coolant is the indication of fuel integrity. As indicated in Figure 1, coolant activity in Westinghouse-fueled plants has decreased from 0.00033 μCi/g (1990) to a current median value of 0.00019 μCi/g, which is significantly lower than the industry-wide level for PWRs.

VANTAGE 5 fuel has achieved extensive operating experience to date. The implementation of VANTAGE 5 fuel features is tailored to each utility's needs. The extensive experience achieved with these features are summarized in Table 1.

2. CORE DESIGN CONSIDERATIONS

Substantial operating experience with high burnup fuel has been accumulated. A total of 35 plants are now operating with fuel which is designed to achieve region average discharge burnups of 40,000 to 45,000 MWD/MTU. The overall economics of long cycles can be favorable, depending on trade-offs between replacement power, fuel cycle costs, and seasonal load variations, among others. Larger reactivity inventories required for longer cycles result in increases in fuel cycle costs, but the use of increased discharge burnup can help mitigate this increase. Therefore, utilities that decided to operate with extended cycles have also typically achieved higher discharge burnup. New fuel designs are already targeting higher discharge burnups and some utilities are moving to 24-month cycle lengths. The use of extended cycle lengths and high discharge burnup fuel management have a major impact on reload core design since it requires higher region average feed enrichments -- 4.5 w/o U235 or higher. The use of high enrichment reload fuel presents challenges in the areas of reactivity control, power distribution control, and fuel rod duty.

High boron concentrations are required to control the initial excess reactivity for longer cycles. In turn, these higher boron concentrations require increased amounts of burnable absorber to maintain moderator temperature coefficient (MTC) within Technical Specifications limits. Westinghouse designs provide efficient solutions to address this issue, with the integral fuel burnable absorber and licensing of a more flexible Technical Specification limit. High boron concentrations also require a re-evaluation of existing accident analyses which are dependent on boron concentration initial conditions (for example, boron dilution events).

Power distribution control becomes more challenging with these fuel management strategies since a larger number of fuel assemblies at higher enrichments are loaded into the core. This creates a more severe reactivity gradient between fresh and burnt assemblies, and care must be taken in the establishment of a core loading arrangement in order to minimize peaking factors. Once again, the use of integral burnable absorbers are used to shape radial and axial power distributions to meet appropriate safety limits. Split fuel batches are often used on a routine basis to help power distribution control. IFMs provide significant benefits in these cases by enabling the increase of peaking factor limits due to increase in DNB margin that they provide.

Fuel rod duty is another important consideration for long cycles and/or high discharge burnups. Fuel rod performance is strongly dependent of the duty experienced during residence in the core. Margin to internal pressure, cladding corrosion, hydriding, and growth limits depends on the design assumptions. Therefore, some rods may approach the licensed peak rod burnup limits, and appropriate design consideration must be made to ensure that leading rod does not surpass the limit.

3. VVER REACTORS

Westinghouse is also developing a fuel assembly design to address the needs of VVER reactor customers. VVANTAGE 6 fuel[3] includes some of advanced features of the VANTAGE 5 fuel, including removable top nozzles, debris resistance bottom nozzles, low pressure drop zircaloy

grids, integral fuel burnable absorbers, and zircaloy guide thimbles. The fuel assembly is being designed to achieve region average discharge burnups of 48,000 MWD/MTU.

The VVANTAGE 6 fuel rod design is based on significant irradiation experience, since it is essentially identical to other Westinghouse products in areas such as rod diameter, cladding thickness, pellet diameter, and pellet-cladding gap. Cladding thickness was selected to maximize uranium utilization and reduce fuel cycle costs while maintaining reliable performance to extended discharge burnups. Cladding material properties have been selected to obtain optimum corrosion performance and high burnup. Fuel rods contain an axial blanket of the natural uranium at the top and bottom 15 cm. Replacement of enriched fuel with this natural uranium eliminates areas of fuel under-utilization at the top and bottom of the fuel assembly and it also increases overall core reactivity. Annular fuel pellets are used in the axial blankets, which also increases the available plenum space in the rod for fission gas release.

The zircaloy grids are designed to minimize pressure drop. The top and bottom grids are fabricated from Inconel-718 to maintain rod support and to limit the axial motion of the fuel rod. The remaining seven grids are fabricated from Zircaloy-4 to improve neutron economy. Axial mixing vanes promote turbulence with a higher degree of mixing, which results in increased margin to departure from nucleate boiling. This design approach is based on the extensive testing and irradiation experience available from the Westinghouse VANTAGE 5 fuel. This similarity provides a broad base of proven technologies that are incorporated in VVANTAGE 6 fuel.

The advanced features incorporated in VVANTAGE 6 fuel provides significant benefits in fuel cycle costs and plant performance margins. For annual cycles, the Westinghouse VVANTAGE 6 - 1000 design is expected to provide about 30% U_3O_8 savings and 25% SWU savings on an equilibrium cycle basis. In addition, this design will have a DNB margin improvement of 20-25% due to the mixing vane design, and a linear peak heat rate generation margin improvement of 15-20% due to increased fuel rod length and application of advanced core design strategies.

4. ADDRESSING NEEDS FOR THE 1990's

PWR fuel operating conditions are expected to become even more demanding in the coming decade, as nuclear power plants continue to require enhanced reliability, greater efficiency, increased flexibility, and reduced costs. Among these demands are:

- increased core operating cycle lengths to higher than 18 months
- region average discharge burnup requirements in the 50,000+ MWD/MTU range
- reactor upratings, leading to higher core temperatures and linear heat rates
- modified reactor coolant chemistry which helps reduce worker radiation exposure but decreases margin to fuel corrosion limits
- system decontamination with the fuel in-place
- increased operational flexibility
- reduction in personnel exposure
- reduction in discharged spent fuel.

A new fuel assembly design, VANTAGE +, was introduced in 1989 to address these needs. VANTAGE + retains the proven features of VANTAGE 5 fuel, but incorporates additional features, such as the ZIRLO™ advanced alloy for fuel rod cladding, integral fuel burnable absorbers enriched in the B-10 isotope, annular axial blanket pellets, and a plenum spring that increases plenum volume.

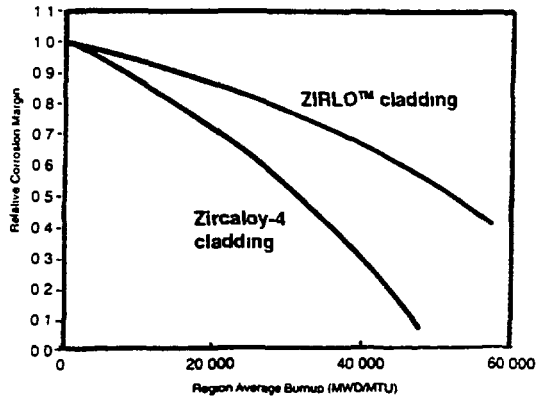

Figure 2 - ZIRLO Corrosion Performance

ZIRLO™ is a specially-developed alloy of zirconium, with niobium, tin, and iron, that has demonstrated its ability to withstand the corrosive effects of high temperature, high-lithium coolant chemistry, and long cycles of operation. The Westinghouse development program in this area involved detailed analysis and testing of a large number of zirconium-based alloys, with elements like niobium, molybdenum, vanadium, copper, manganese, germanium, and tin. ZIRLO™ offered better waterside corrosion performance in out-of-pile, as well as in-pile testing, including exposure of fuel rods in the BR3 Reactor in Belgium to burnups levels of 68,000 MWD/MTU. A demonstration program with ZIRLO™ fuel rods was initiated in the North Anna 1 reactor in 1987. These ZIRLO™ rods have already completed two cycles of operation. Examinations of these rods have demonstrated the excellent corrosion resistance and reduced irradiation growth of ZIRLO™, typically 67% and 50% less than Zircaloy-4. This enhanced margin to corrosion limits is illustrated in Figure 2. The first full reload region with ZIRLO™ began operation in 1991 at the V.C. Summer reactor of South Carolina Electric and Gas Company in the United States.

In parallel with the fuel assembly design development programs, Westinghouse has also established major programs in fuel management and core monitoring methods. As a result of these efforts, the state-of-the-art PHOENIX-P/ANC design methodology has been developed, qualified and used for core design. These methods are capable of modelling the complex three-dimensional features of advanced fuel assemblies and core design strategies with increased accuracy. Significant margin improvements are also provided with BEACON, an advanced system that provides on-line detailed 3D core power distribution information and enables the introduction of direct margin monitoring of Technical Specifications.

The Westinghouse technology program development provides integrated hardware and software solutions to bring benefits to PWR operators. The advanced fuel assembly designs introduced over the last decades enhance the level of nuclear fuel performance and reliability to levels of excellence consistent with nuclear utilities' long range objectives.

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SAFETY-RELATED BOUNDARY CONDITIONS FOR ADVANCED RELOAD DESIGN

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Abstract

In this paper a method is discussed to overcome the discrepancy between the demands for increased fuel management flexibility on the one hand and for permanent operating licenses on the other. By defining safety–related boundary conditions it is possible to determine the safety–related characteristics of reload cores in advance, in spite of the fact that they differ from one another within certain limits. The basis of the boundary conditions is given by the essential mechanical design features of the fuel assemblies (hardware frame) and the concept behind safety analysis and safety–related requirements (software frame) together with the verified limits of the key safety parameters defined by the total amount of explicit analyses carried out during the construction phase of the plant and – possibly – in previous operating cycles.

Key safety parameters denote those input/output parameters of safety analysis which determine the safety–related aspects of core behaviour. With respect to reload safety evaluation, only those safety parameters are relevant which may vary significantly from reload to reload.

Safety analysis is a two–dimensional array structured by requirement categories and areas of analysis. Primary (external) design criteria are of direct relevance to safety. They define safety margins to failure and determine the range fixed by the operating license. Derived (internal) design criteria are only of indirect relevance to safety. They simplify verification efforts, but do not determine the range fixed by the operating license. Within reload safety evaluation, in general, it is sufficient to demonstrate that the safety–related input parameters are within the verified limits.

The application of these safety–related boundary conditions to in–core fuel management is discussed for an exemplary equilibrium core of the PWR 1300 MW characterized by a number of features typical for advanced reload design. Safety evaluation demonstrates the feasibility of the envisaged fuel management strategy. Moreover, it helps to identify, if necessary, hardware modifications indispensable or recommendable prior to realization of challenging loading schemes.

1 Introduction

Advanced in–core fuel management is a symbiosis of economic optimization and response to varying utility requirements regarding cycle length and capacity factor. This implies a large variation bandwidth in future reload cycles. An individual reload cycle is characterized by a number of variable parameters like reload fraction, reload enrichment, type of fuel as well as number, type and spatial distribution of burnable absorbers. Depending on these variable boundary conditions, a new loading pattern has to be prepared for each cycle. The core design determined by it is one of numerous core design modifications. Moreover, due to general advances made in science and technology, one has to incorporate also fuel assembly design modifications into the spectrum of possible reload cores. Such foreseeable modifications relate to design changes, use of other materials and fuels as well as to changes in specifications.

Due to the German Atomic Law, utilities need permanent operating licenses. A license, however, requires definition in sufficient depth of the object to be licensed. Therefore it can only be granted if future reload cores are sufficiently defined.

2 Concept of Safety–Related Boundary Conditions

2.1 Overview

In this contribution a method is presented to overcome the discrepancy between these conflicting aims. The basic approach of the method is to define safety–related boundary conditions. This set of boundary conditions makes it possible to determine the safety–related characteristics of reload cores in advance, in spite of the fact that they differ from one another within certain limits.

The basis of the boundary conditions is given by the following three elements:

- the essential mechanical design features of the fuel assemblies (hardware frame),
- the concept behind safety analysis and safety–related requirements (software frame),
- a summary of data containing the verified limits of the key safety parameters as defined by the total amount of explicit analyses carried out during the construction phase of the plant and – possibly – in previous operating cycles.

In this context, key safety parameters denote those input/output parameters of safety analysis which determine the safety–related aspects of core behaviour. With respect to reload safety evaluation, only those safety parameters are relevant which may vary significantly from reload to reload.

All reload cores which meet the present safety–related boundary conditions are to be regarded as being acceptable with respect to safety and, therefore, to be equivalent, i.e. they differ in their safety–related characteristics to such a small extent that a transition from one to another does not represent a substantial modification. This is a mandatory prerequisite for a permanent operating license.

2.2 Structure of Safety Analysis and Data Flow

The safety analysis of the reactor core is based on a variety of detailed analyses interconnected in multiple ways. These analyses may be performed or verified in part already during construction of the plant; the remaining part, however, only for the actual core loading.

Since the safety–related requirements differ according to the particular load condition being analyzed, the set of possible detailed analyses must be classified first according to requirement categories. The following categories are to be considered for this:

- normal operation;
- operational malfunctions and accidents:
  - transients,
  - LOCA, external events.

The respective detailed analyses to be performed for each requirement category are thematically combined to the following areas of analysis:

- neutron physics,
- thermal hydraulics,
- system dynamics,
- fuel rod design,
- fuel assembly structure design,
- LOCA analysis.

Thus, the safety analysis has a two–dimensional structure consisting of requirement categories and areas of analysis.

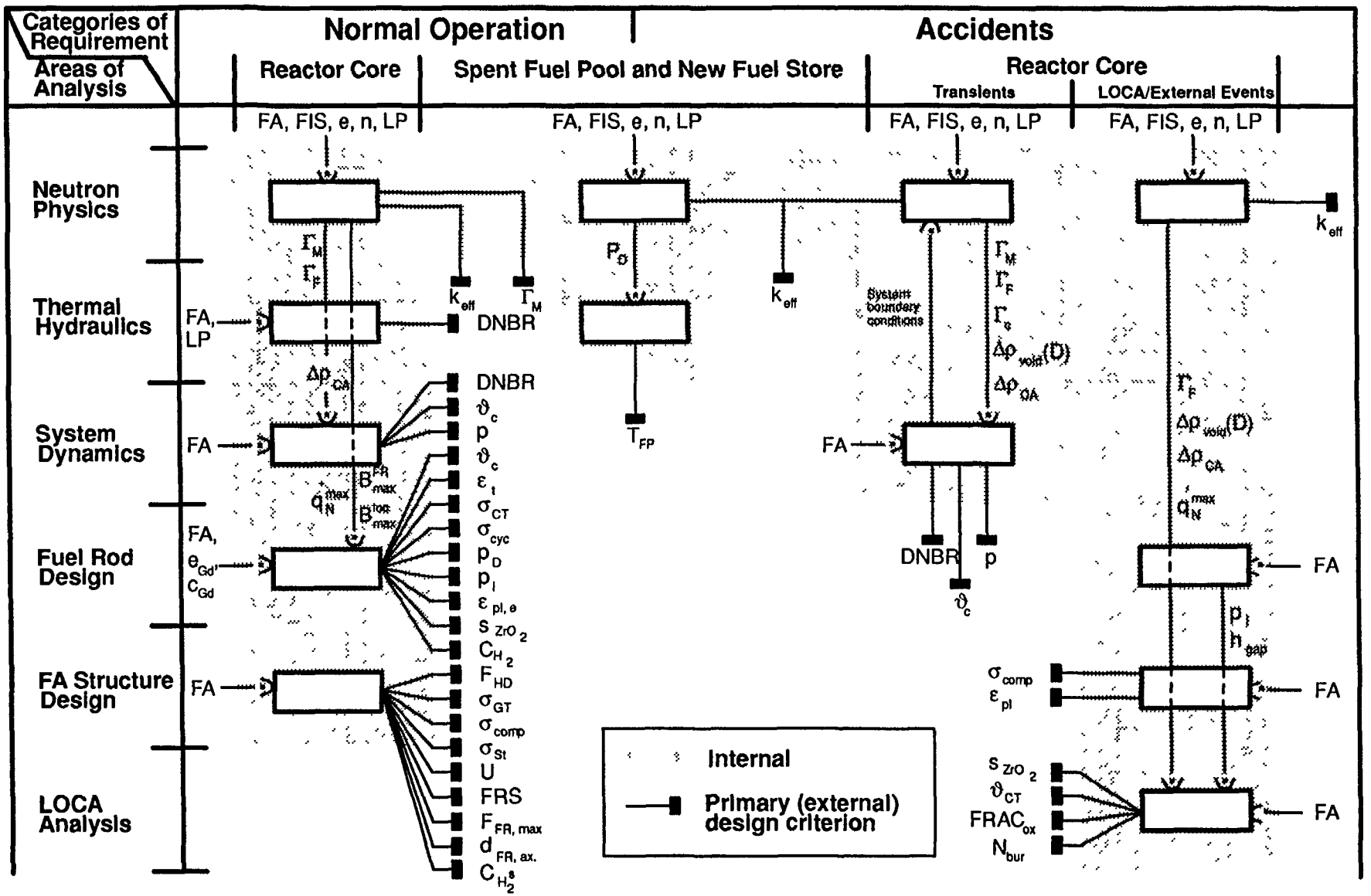


Fig. 1 Structure of Safety Analysis for Reload Cores (symbols explained in nomenclature appendix)

Fig. 1 shows the data flow between the individual structural elements. Besides the aspects of the reactor core the diagram includes the aspects of the spent fuel pool and new fuel store. In each area of analysis input data are processed into output data via computational models and then evaluated with respect to their safety–related relevance.

Input data are either the result of the specific core design or operating conditions (external input data) or have been obtained from preceding areas of analysis (internal input data). Correspondingly, output data either serve directly to prove the acceptability with respect to safety (external output data) or are required as input for subsequent areas of analysis (internal output data).

2.3 Safety–Related Requirements and Design Criteria

The acceptability of a reactor core with respect to safety is assured if certain safety–related requirements are met. Such requirements are established for each requirement category and each area of analysis. The requirements result from the applicable codes, standards, guidelines, regulations etc. Thus the variation bandwidth of possible design modifications is limited.

The safety–related requirements are specified and quantified by design criteria. They define permissible ranges of values (bounds) for safety–related output parameters.

Only those design criteria which relate to external output parameters are of direct relevance to safety. These primary (external) design criteria are defined in a conservative way compared to anticipated technological failure limits and thus define safety margins to failure. They determine the range fixed by the operating license.

Derived (internal) design criteria, i. e. those which relate merely to internal output parameters, are only of indirect relevance to safety. This relevance is due to the fact that the internal output parameters are at the same time input parameters for subsequent analyses and as such must lie within permissible boundaries. Thus, the derived design criteria serve to simplify verification efforts, but do not determine the range fixed by the operating license.

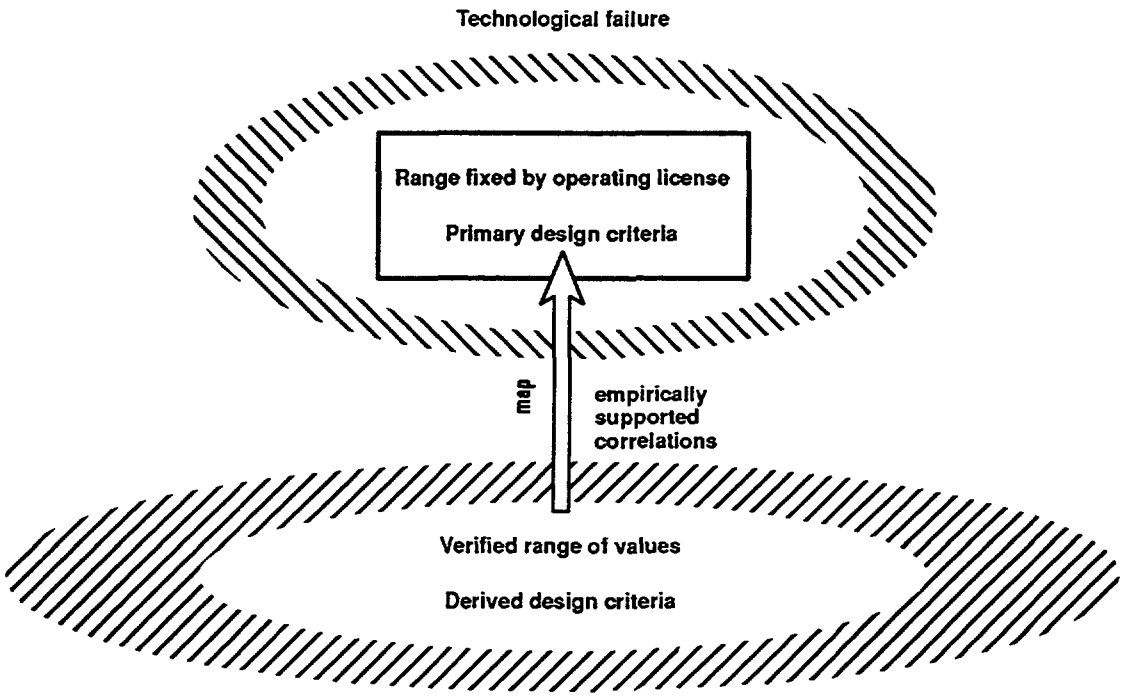


Fig. 2 Concept of Reload Safety Evaluation

2.4 Reload Safety Evaluation Concept

The objective of safety analysis is to prove, for a given set of input parameters, that the design criteria are met. This is done explicitly by calculation for each area of analysis during the licensing phase of the plant for the first core as well as for exemplary reload cores.

Input and output parameters of each analysis are interconnected via empirically supported correlations. Within the scope of these correlations, the entire set of design criteria of an area of analysis defines at the same time a permissible range of values for the input parameters of that area of analysis. Moreover, the total amount of explicit analyses defines a verified range of values. This situation is illustrated schematically in Fig. 2.

The verified range of values, in general, will be considerably smaller than the actually permissible range of parameter values. This is due to the fact that worst–case analyses covering the entire range of possible input parameter values have not been carried out in all cases. Every new analysis extends the verified range of input parameters and of derived (internal) design criteria as well. The primary (external) design criteria are not affected by this procedure.

Within reload safety evaluation it is not necessary to repeat all analyses. In general, it is sufficient to demonstrate that the safety–related input parameters are within the verified limits. In general, consequently, new explicit analyses which prove compliance with the primary design criteria are necessary only if these verified limits are exceeded.

Detailed design documents summarizing the results of reload safety evaluation are prepared for each reload cycle and submitted to the authorized inspector. On the basis of the documents, the authorized inspector has to check whether the reload core design complies with the safety–related boundary conditions. If this is the case no further verifications are required.

3 Application to In–Core Fuel Management

In the following section the application of safety–related boundary conditions to advanced in–core fuel management is discussed. The example reactor selected for this purpose is the Siemens PWR 1300 MW, the reload core presented is an equilibrium core used for long–term considerations.

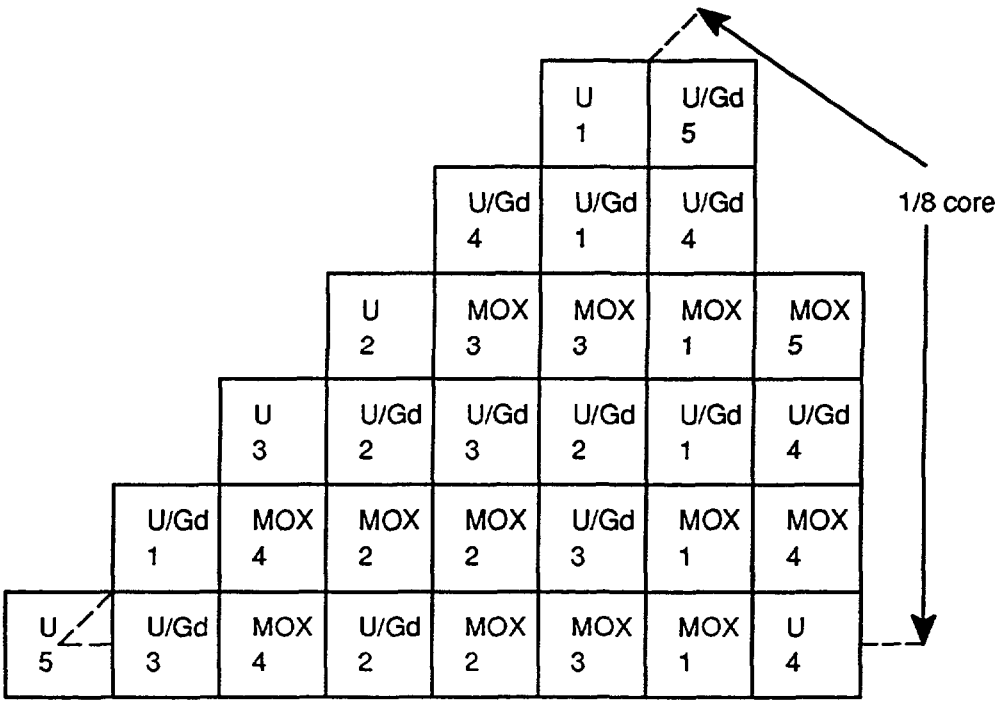


Fig. 3 Exemplary Equilibrium Cycle of a PWR 1300 MW (193 FA): Schematic Loading Pattern

3.1 Characteristics of Exemplary Cycle

The schematic equilibrium loading pattern is given in Fig. 3. In each fuel assembly (FA), the first line denotes the type of fuel, the second line the irradiation period.

The example core illustrated above is characterized by a number of features typical for advanced reload design. These features particularly include:

- high reload enrichment (4 w/o U235, equivalent Pu_{fiss} content in MOX fuel assemblies),
- full–low–leakage strategy,
- usage of Gd_2O_3 burnable absorbers and
- a high fraction of MOX fuel assemblies (45 % of FA inventory).

The most important characteristics of the MOX fuel assemblies are given by the carrier material (tails Uranium) and the design with 4 water rods (flooded cladding tubes) to minimize the heterogeneity of power density distribution within the MOX FA.

The design of the equilibrium cycle is suitable for annual refueling. The cycle length amounts to approximately 315 efpd, including a stretch–out operation of about one month. This implies a high discharge burnup at given reload enrichment (cf. section 3.3).

3.2 Safety Evaluation Procedure

The present contribution summarizes highlights of the safety evaluation procedure. The items are arranged according to the different areas of analysis.

3.2.1 Neutron Physics

Neutron physics analysis has to be carried out cycle by cycle, as it is not possible to show any correlation between the (incalculably many) possible modifications of the loading pattern and the key safety parameters to be observed by neutron physics.

The safety–related requirements refer to inherent safety of the core and to a sufficient shutdown margin /1/. Inherent safety is ensured by a negative moderator temperature coefficient at the full power, Xenon equilibrium state. A sufficient shutdown reactivity has to be provided for the cases of reactor trip as well as for cold shutdown and long–term maintenance of subcriticality. Shutdown margin analysis has to comply with limits of 1 % resp. 5 %, depending on the case investigated /2/.

The detailed three–dimensional depletion calculations do not only yield parameters like critical boron concentration or cycle length, but also (in connection with an off–line pin–by–pin evaluation processor) the local power density and burnup distribution over the whole reactor cycle.

Besides demonstrating compliance with the primary (external) design criteria, selected reactivity coefficients and integrals are checked. This simplifies reload safety evaluation for other areas of analysis.

3.2.2 Thermal Hydraulics

The safety requirements for normal operation and anticipated operational occurrences demand the maintenance of fuel cladding integrity under all associated conditions. The corresponding primary (external) design criterion is to avoid departure from nucleate boiling (DNB) with 95 % probability and 95 % confidence level /3/, taking into account the loss of flow event as design basis and considering deviations of operational variables from their nominal values.

Due to the design characteristics of a Siemens PWR, the loss of flow transient is the limiting DNB transient. Explicit analysis of this transient is carried out for each reload cycle. The result is a cycle–specific limit

value for the steady–state DNBR (including allowance for uncertainties). Reload core design directly uses the 3–D power density distribution to demonstrate compliance with the steady–state DNBR limit.

It is a characteristic feature of modern Siemens pressurized water reactors that they are equipped with an in–core monitoring system and optimized axial power shape control. Among other parameters, the steady–state DNBR is used and monitored.

As a result of the enhanced monitoring facilities in Siemens PWRs, small offset ratios in axial power density distribution can easily be met. Consequently, radial power peaking factors can be raised substantially (in the present case to values significantly above 1.60) without any restrictions on operational flexibility.

Thus, the combination of cycle–specific DNBR analysis and advanced monitoring system allows to reduce over–conservative safety margins and facilitates economic fuel management.

3.2.3 System Dynamics

The reference safety analysis of system behaviour is carried out using a conservative set of input parameters. Therefore reload safety evaluation can concentrate on a check of the relevant reactivity coefficients and integrals.

A somewhat special case is given by transients involving recriticality of the reactor core (steam line leak), where Siemens uses a specific methodology, the concept of fictitious supercriticality at zero load /4/.

Within the scope of this methodology, the relevant key safety parameters minimum DNBR and maximum centerline fuel temperature are correlated with the eigenvalue at cold zero power, $\Delta\rho_{FUEN}$, thus defining a key parameter substitute. The correlations yield a $\Delta\rho_{FUEN}$ limit compliance with which rules out DNB as well as centerline fuel melting. Within reload safety evaluation for individual cycles only a simple eigenvalue calculation ($\Delta\rho_{FUEN}$) is required.

3.2.4 Fuel Rod Design

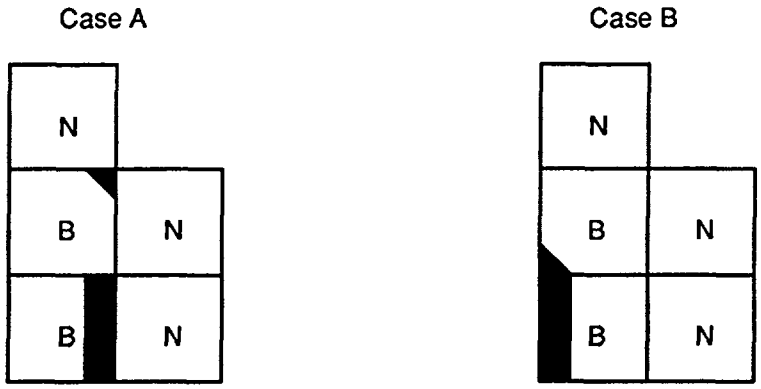
The fuel rod is subjected to a number of loads during normal operation and anticipated operational occurrences. It is necessary to limit these loads to ensure mechanical integrity of the rod. In fuel rod design, therefore, several key safety parameters are involved. These parameters result from different types of analysis (hot channel analysis, stress analysis, analysis of long–term behaviour). The bulk of them is verified by checking against verified burnup and linear heat rate limits.

For the state of the art for fuel rod design, waterside corrosion is the limiting phenomenon, and the thickness of the outer oxidation layer on the cladding tube (s_{ZrO2}) is the most restrictive key safety parameter. Corrosion is a complicated mechanism which depends not only on burnup but is also significantly influenced by the individual power history of the fuel rod. For this reason, the correlation between s_{ZrO2} and fuel rod burnup is rather weak, and the only way to avoid unnecessary restrictions on fuel management is given by explicit core–wide pin–by–pin corrosion analysis.

The result of corrosion analysis, i.e. the corrosion distribution may be evaluated with respect to safety by two different methods. The deterministic approach is to demonstrate compliance with an upper corrosion limit value for all fuel rods. Within the more sophisticated statistical approach, the corrosion distribution is folded with the defect probability distribution (defect probability as a function of s_{ZrO2}). In that case, the allowable expectation value of corrosion–induced fuel rod damage is less than one pin per reactor cycle.

Using the above methodology, discharge burnup may be further increased by optimization of the core loading pattern against corrosion (and not just for power density). This is schematically illustrated in Fig. 4.

Optimization against corrosion leads to higher radial power peaking factors than optimization just for power density. In Siemens PWRs, however, the required margin is provided by thermal–hydraulic analysis. Optimization against corrosion, therefore, is an adequate tool for further improvement in economic fuel utilization.



N: new fuel assembly
B: fuel assembly with high burnup
black: region of maximum burnup within fuel assembly

Fig. 4 Optimization Strategies for Power Density (Case A) and against Corrosion (Case B)

3.2.5 Fuel Assembly Structure Design

Fuel assembly structure design is a task to be carried out for a given type of fuel assembly and not for an individual reload design. The necessary calculations are performed in advance in the course of development of new FA types.

3.2.6 LOCA Analysis

In Germany, LOCA analysis has to meet a specific 10 % fuel rod defect limit /1/. Compliance with this primary (external) design criterion is demonstrated in two steps. First, a defect threshold is determined for each type of fuel rod. The analysis takes into account pre–accident cladding corrosion , thus resulting in threshold values depending on burnup.

The second step to be carried out for each reload consists of an evaluation of the number of fuel rods with a linear heat rate above the respective limit values. For this purpose an extremely conservative bounding distribution is constructed by expanding the calculated fundamental power density distribution. The expansion factor is defined such that the maximum power density of the bounding distribution is identical to the setpoint of the limitation system.

3.3 Results of Safety Evaluation

The characteristic features of the exemplary equilibrium cycle have a systematic influence on some key safety parameters. The results of reload safety evaluation can be summarized as follows:

The high reload enrichment and the low reload fraction lead to discharge burnup values beyond 50 MWd/kg for batches with 5 irradiation periods. The influence on fuel rod internal pressure and waterside corrosion can be minimized by proper rod design (sufficient free volume for fission gas) and by the choice of commercially available corrosion–resistant cladding material.

The full–low–leakage loading pattern in combination with the optimization strategy against corrosion yields rather high radial power peaking factors, in spite of the extensive usage of Gd_2O_3 burnable absorbers. Explicit DNBR analysis, however, demonstrates a sufficient margin for operational flexibility.

The large fraction of MOX fuel assemblies in the core results in a strongly negative moderator temperature coefficient at end of cycle ($\Gamma_M < -70$ pcm/K) and in a small absolute boron worth ($\Gamma_c = -5.5$ pcm/ppm at

begin of cycle). This implies a high, but tolerable reactivity release in the course of subcooling transients ($\Delta\rho_{FUEN}$) and extensive requirements with respect to borating systems. Depending on the hardware equipment and technical specifications of the individual plant, an increase in refueling boron concentration and/or boron inventory of the borating safety system may be required.

The safety evaluation of the example core demonstrates the feasibility of the envisaged fuel management strategy. Moreover, it helps to identify, if necessary, hardware modifications indispensable or recommendable prior to realization of challenging loading schemes.

4 Conclusion

The concept of safety–related boundary conditions discussed in this contribution has the capability of handling a large variety of possible reload cores. It provides substantial flexibility in reload safety evaluation procedure, ranging from explicit analysis (whenever required to avoid unnecessary restrictions) to mere check of relevant input parameters (whenever possible to simplify verification efforts). Moreover, it contains a detailed specification of the range determined by the operating license as well as a comprehensive checklist of the items to be treated within the documents prepared for an individual reload cycle. Thus, it is a powerful tool for utilities, core designers, authorized inspectors and licensing authorities.

At present, the concept has been introduced for 10 Siemens pressurized water reactors. The accumulated operating experience amounts to approximately 50 reactor cycles.

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Appendix

Nomenclature

FA	Type of fuel assembly
FIS	Fissile material in new fuel assemblies
e	Enrichment of new fuel assemblies
e_{Gd}	Enrichment of fissile material in new Gd–rods
c_{Gd}	Gd_2O_3 concentration
n	Number of new fuel assemblies
LP	Loading pattern
k_{eff}	Effective multiplication factor
DNBR	DNB ratio
$q'_N{}^{max}$	Maximum LHGR during the cycle for steady–state operation at rated power
$B_{max}{}^{FR}$	Maximum fuel rod burnup
$B_{max}{}^{loc}$	Maximum local burnup
ϑ_c	Centerline fuel temperature
ε_t	Transient tangential strain due to fast positive power changes
σ_{CT}	Stress in cladding tube
σ_{cyc}	Cyclic stress under dynamic load
p_D	Fuel rod design pressure
p_i	Fuel rod internal pressure
$\varepsilon_{pl,e}$	Equivalent plastic strain under tensile load
s_{ZrO2}	Thickness of outer oxidation layer of cladding tube
c_{H2}	Average H_2 absorption in cladding tube
F_{HD}	Hold–down force
σ_{GT}	Stresses in guide thimbles
σ_{comp}	Axial compressive stresses in guide thimbles
σ_{st}	Stresses in structural components excluding guide thimbles
U	Cumulative usage factor
FRS	Fuel rod support in spacer grid
$F_{FR,max}$	Axial compressive load on fuel rods
$d_{FR,ax.}$	Axial clearance of fuel rods
$C_{H2}{}^S$	Average H_2 absorption in structural components
ε_{pl}	Permanent deformation of spacer grids in fuel assemblies with control assemblies
Γ_M	Moderator temperature coefficient of reactivity (HFP, Xe eq.)
Γ_F	Fuel temperature coefficient of reactivity (HFP, Xe eq.)
Γ_c	Boron worth
$\Delta\rho_{void}(D)$	Void reactivity as a function of moderator density
$\Delta\rho_{CA}$	Control assembly net worth (hot zero power)
p	Coolant pressure
h_{gap}	Gap heat transfer coefficient
ϑ_{CT}	Cladding tube temperature
$FRAC_{ox}$	Fraction of oxidized Zirconium
N_{bur}	Number of burst fuel rods
P_D	Decay heat power
T_{FP}	Spent fuel pool temperature

OPTIMIZATION OF AXIAL GADOLINIUM LOADING IN A VVER-440 TYPE REACTOR

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Abstract

The present paper deals with the elaboration of a code system for the Gadolinium axial distribution optimization in a VVER-440 reactor. The work follows the original ideas of Dressnum and Lee in connection with the applications of conjugate gondearts method to the pattern solution but some modifications have been made in order to take into account the spectual interaction between Gadolinium and fuel burnup and the objective function and state equations are descarried in terms of neutron yield cases section. A parameterized library has been created on the basis of WIMS-D14 spectral core calculational results by means of polinomical and exponential dependences fitting to this set of values. The contained distribution for the axial Gadolinium loading reduced the power peaking factor in a 12% extent in association with a uniform distribution loading.

1. Introduction.

In the frame of the work about the use of gadolinium as a burnable poison for PWRs some optimal control studies have been carried out.

Several of them treat the problem of gadolinium optimal axial distribution to optimize core performance throughout a cycle and to avoid axial oscillations in the core power distribution.

In reference [1] Drumm and Lee show a method to solve this kind of problem. Their optimization scheme is based on Pontryaguin's maximum principle, with the objective function accounting for a target power distribution. The state and the resulting Euler-Lagrange equations are solved iteratively using the conjugate gradients method to find the optimal search direction and the first-order perturbation theory to estimate the search lenght.

They applied this method to a PWR reducing the power peaking factor in 12.8 % [1].

The work presented here follows the general formulation of the space-time optimal control problem solved by Drumm and Lee in their work, but in our case the method is applied to the core performance optimization of the VVER-440 type reactor.

In the original work, the spectral interaction between the fuel burnup and gadolinium spectral effects is neglected, introducing several corrections into cross sections to match as closely as possible preliminary results with more accurate ones. In the present work this spectral interaction is accounted for which slightly complicates our task.

In our case the objective function as well as the burnup and power constraint equations were described in terms of $\nu\Sigma_f$, instead of Σ_f. This fact causes a litte difference between our equations and the original ones [1].

To solve the problem we developed a program system written in Turbo Pascal 5.5 for IBM PC/AT or compatible computers and using it we have reduced the power peaking factor in the range of 12% to 16% according to the desired power distribution shape used.

2. Gadolinium burnable absorber optimization.

The method developed by Drumm and Lee allows to handle distributed parameters in space-time optimal control problems.

In the gadolinium loading optimization problem some variables depend on the space and time, but others as the multiplication factor K_{eff} and the soluble boron concentration Cb are spatially integrated and are therefore independent of space. On the other hand several variables are given at the beginning of the cycle (BOC) and therefore depend only on the space.

Here the state variables are the neutron flux $\Phi(z,t)$, the neutron current $J(z,t)$, the fuel burnup $E(z,t)$, the integral power $P(z,t)$ and the soluble boron concentration $Cb(t)$. The control variables are the gadolinium pins per assembly $\alpha_1(z)$ and the initial concentration of gadolinium in the poisoned pins $\alpha_2(z)$ in weight percent units.

The problem is presented in one-group one-dimensional diffusion approximation for slab geometry. It is assumed that the cross sections are functions of the average fuel burnup, moderator and fuel temperatures, the soluble boron concentration, the number of pins per assembly and the concentration of gadolinium in the poisoned pins at the BOC.

The general task is to minimize the following objective functional [1]

$$J = \int_0^T \int_0^H \{ L[\underline{x}(z,t), \underline{y}(t), \underline{u}(z)] \} \, dzdt \qquad (2.1)$$

subject to a set of differential and algebraic equations

$$G(\underline{x}, \underline{x}_t, \underline{x}_z, \underline{y}, \underline{u}) = \underline{0} \qquad (2.2)$$

where

T: cycle length.

H: core height.

$\underline{x}(z,t)$: n dimensional vector of state variables that depends explicitly on both time and space.

$\underline{y}(t)$: m dimensional vector of state variable that depends explicitly on only the time variable.

$\underline{u}(z)$: p dimensional vector of control variable that depends explicitly on only the space variable.

G: q dimensional vector of constraint equations with boundary and initial conditions specified in such a way that for a given control vector the state of the system can be uniquely determined.

The x_t and x_z are defined as

$$x_t = \delta x/\delta t \qquad (2.3)$$

$$x_z = \delta x/\delta z \qquad (2.4)$$

and the function L is a piecewise continuous function.

To determine the optimal control a q dimensional Lagrange multiplier row vector $\Gamma(z,t)$ is considered and the term $\Gamma \cdot G$ is introduced into the objective function. According to the calculus of variations the first order variation of the augmented functional must be zero at the optimal solution [1].

After integration by parts [1], the resulting Euler-Lagrange equations are given by the formula

$$\Gamma \cdot \delta G/\delta x - \Gamma_t \cdot \delta G/\delta x_t - \Gamma_z \cdot \delta G/\delta x_z = -\delta L/\delta x \qquad (2.5a)$$

$$\int_0^H \{ \Gamma \cdot \delta G/\delta y \}\, dz = -\int_0^H \{ \delta L/\delta y \}\, dz \qquad (2.5b)$$

with the final conditions for the Lagrange multipliers vector

$$\left[\Gamma \cdot \delta G/\delta x_t \right]_T = 0 \qquad (2.5c)$$

The terms involving perturbations in the control variables determine the gradient [1] and is given by

$$g(z) = \int_0^T \{ \delta L/\delta u + \Gamma \cdot \delta G/\delta u \}\, dt \qquad (2.6)$$

The conjugate gradient algorithm can be used to calculate the optimal search direction.

The initial step is to estimate a reasonable value of the control $\underline{u}_1(z)$, then the next step is to solve the constraint and Euler-Lagrange equations for the state variables and Lagrange multipliers respectively. Then the first conjugate gradient direction is taken as the negative gradient direction

$$\underline{s}_1 = -\underline{g}_1 \qquad (2.7)$$

The subsequent conjugate direction can be determined by the recursion formula

$$\underline{s}_{k+1} = -\underline{g}_{k+1} + \beta_k \cdot \underline{s}_k \qquad k=1,2.. \qquad (2.8)$$

where

$$\beta_k = < \underline{g}_{k+1} , \underline{g}_{k+1} > / < \underline{g}_k , \underline{g}_k > \qquad (2.9)$$

The optimal search length calculation is based in first-order perturbation theory [1] that is a consistent choice, since Lagrange multipliers and the search direction are known only to first order. The perturbations in the state variables are calculated for a perturbation in the control given by the relationship

$$\delta \underline{u}_k = \epsilon_o \cdot \underline{g}_k \qquad (2.10)$$

where ϵ_o is an estimate of the optimal search length. In principle the value for ϵ_o is arbitrary, since the perturbation in the state variables scale linearly with the chosen value, howewer, in certain cases, the value of ϵ_o must be restricted to avoid the violation of the first-order assumption.

Linearization of the system state equation about a reference state yields [1]

$$\delta \underline{G}/\delta \underline{x} \cdot \delta \underline{x} + \delta \underline{G}/\delta \underline{x}_t \cdot \delta \underline{x}_t + \delta \underline{G}/\delta \underline{x}_z \cdot \delta \underline{x}_z +$$

$$\delta \underline{G}/\delta \underline{y} \cdot \delta \underline{y} + \delta \underline{G}/\delta \underline{u} \cdot \delta \underline{u} = \underline{0} \qquad (2.11)$$

Once the boundary and initial conditions are given, the perturbations in the state variables can be determined for a given perturbation of the control variables.

Since the first-order perturbation in the state vectors are directly proportional to the search length, the optimal search length ϵ is calculated by direct search [1] minimizing

$$J(\epsilon) = \int_0^T \int_0^H \{ L[\underline{x} + \epsilon \cdot \delta \underline{x}, \underline{y} + \epsilon \cdot \delta \underline{y}, \underline{u} + \epsilon \cdot \delta \underline{u}] \} \, dzdt \qquad (2.12)$$

In spite of the state equations have been linearized, this equation conserves the nonlinearity of the objective function, which is an essential condition for convergence.

Finally the new estimate of the optimal control is given by

$$\underline{u}_{k+1} = \underline{u}_k + \epsilon_k \cdot \underline{S}_k \qquad (2.13)$$

The iteration is then continued until the convergence criterium is satisfied.

In our case the $\underline{x}$ vector is given by

$$\underline{x} = \begin{bmatrix} \Phi(z,t) = \text{(neuton flux)} \\ J(z,t) = \text{(neutron current} \\ E(z,t) = \text{(fuel burnup)} \\ P(z,t) = \text{(integrated power)} \end{bmatrix} \qquad (2.14)$$

where the integrated power is defined by

$$P(z,t) = \int_0^z \{ \nu\Sigma_f\Phi \}\, dz' \qquad (2.15)$$

and

$\nu\Sigma_f = \nu\Sigma_f(z,t)$ is the neutron yield cross section.

The $\underline{y}$ vector is

$$\underline{y} = \begin{bmatrix} Cb(t) = \text{(soluble boron concentration)} \end{bmatrix} \qquad (2.16)$$

and the control vector $\underline{u}$ is

$$\underline{u} = \begin{bmatrix} \alpha_1(z) = \text{(Gd pins by assembly)} \\ \alpha_2(z) = \text{(Gd pins initial concentration)} \end{bmatrix} \qquad (2.17)$$

The function L is given by

$$L[\underline{x}, \underline{y}, \underline{u}] = C \cdot \left[(Kq - Qd)^4 + w_a \cdot Kq \right] \qquad (2.18)$$

where

$$C = 1.0/H \cdot \int_0^H \{ \nu\Sigma_f\Phi \}\, dz \quad \text{(mean power)} \qquad (2.19)$$

$$Kq = \nu\Sigma_f\Phi/C \quad \text{(power peaking factor)} \qquad (2.20)$$

$Qd = Qd(z,t)$ is the desired power distribution shape

w_a is a weighting factor that determines the relative importance to be placed on minimizing the axial offset. The parameter w_a is chosen to be positive in the half of the core where the average power is greater than the core average and negative in the other half.

The constraint vector $\underline{G}$ consists of

$$G_1 = \Phi_z + J/D = 0 \qquad \text{(Fick's law)} \qquad (2.21a)$$

$$G_2 = J_z + (\Sigma_a - \nu\Sigma_f)\cdot\Phi = 0 \qquad \text{(Diffusion equation)} \qquad (2.21b)$$

$$G_3 = E_t - \nu\Sigma_f\cdot\Phi = 0 \qquad \text{(Burnup equation)} \qquad (2.21c)$$

$$G_4 = P_z - \nu\Sigma_f\cdot\Phi = 0 \qquad \text{(Power normalization)} \qquad (2.21d)$$

with the following boundary and initial conditions

$$\Phi(0,t) = 0 \qquad (2.22a)$$
$$\Phi(H,t) = 0 \qquad (2.22b)$$
$$E(z,0) = E_o(z) \qquad (2.22c)$$
$$P(0,t) = 0 \qquad (2.22d)$$
$$P(H,t) = P_h(t) \qquad (2.22e)$$

where

$$P_{h(t)} = \int_0^H \{ \nu\Sigma_f\Phi \}\, dz \qquad (2.23)$$

The Σ_a absorption cross section in the diffusion equation (2.21b) is given by

$$\Sigma_a = \Sigma_{a0} + \Sigma_{aGd} + \Sigma_B + D\cdot Br^2 \qquad (2.24)$$

where

$$\Sigma_{a0} = \Sigma_{a0}(E,Tm,Tf) = \Sigma_{a0}(z,t) \qquad (2.25)$$
Assembly averaged absorption cross section without gadolinium.

$Tm(z,t)$: moderator temperature.

$Tf(z,t)$: fuel temperature.

$$\Sigma_{aGd} = \Sigma_{aGd}(E,\alpha_1,\alpha_2) = \Sigma_{aGd}(z,t) \qquad (2.26)$$
Assembly averaged gadolinium absorption cross section.

$$\Sigma_B = \Sigma_B(Cb,Tm) = \Sigma_B(z,t) = \theta(z,t)\cdot Cb(t) \qquad (2.27)$$
Soluble boron absorption cross section required for criticality.

$$\theta(z,t) = \theta(Cb,Tm) \qquad (2.28)$$
Function that relates the soluble boron concentration $Cb(t)$ to the boron absorption cross section.

D: diffusion coefficient.

Br^2: Radial buckling.

The $v\Sigma_f$ neutron yield cross section is given by

$$v\Sigma_f = v\Sigma_{f\theta}(E,Tm,Tf,Cb) + v\Sigma_{fGd}(E,\alpha_1,\alpha_2) = v\Sigma_f(z,t) \qquad (2.29)$$

where

$v\Sigma_{f\theta}$: Assembly averaged neutron yield cross section without gadolinium.

$v\Sigma_{fGd}$: Gadolinium neutron yield cross section to account for the spectral interaction between fuel burnup and gadolinium in this cross section.

The axial dependence of the temperature is computed from

$$Tm(z,t) = Tm(0) + [Tm(H) - Tm(0)]/Ph(t)\cdot\int_0^H v\Sigma_f\cdot\Phi\cdot dz \qquad (2.30a)$$

$$Tf(z,t) = Tm(z,t) + (T_F - T_M)\cdot Kq \qquad (2.30b)$$

where

$Tm(0)$: inlet moderator temperature

$Tm(H)$: outlet moderator temperature

T_F: axial average fuel temperature

T_M: axial average moderator temperature

Moreover, in the original work [1] the objective function L, as well as the burnup and power constraints, were written in terms of fission cross section, while in this paper the formulation is in terms of the $v\Sigma_f$.

On the other hand in reference [1] the spectral interaction of fuel burnup and the gadolinium burnable absorber was not explicitly accounted for. As will be seen in the next section we considered this effect in an explicit way. It leads to the fission yield cross section dependence on the control variables, thus the actual gradient formula will be slightly different in our case.

From (2.5) the Euler-Lagrange equations here are given by

$$\Gamma_1/D - \Gamma_{2z} = 0 \qquad (2.31a)$$

$$-\Gamma_{1z} + (\Sigma_a - v\Sigma_f)\cdot\Gamma_2 - v\Sigma_f\cdot(\Gamma_3 + \Gamma_4) = -\delta L/\delta\Phi \qquad (2.31b)$$

$$(\delta\Sigma_a/\delta E - \delta v\Sigma_f/\delta E)\Phi\cdot\Gamma_2 - \delta v\Sigma_f/\delta E\cdot\Phi\cdot(\Gamma_3 + \Gamma_4) - \Gamma_{3t} = -\delta L/\delta E \qquad (2.31c)$$

$$- \Gamma_{4z} = 0 \tag{2.31d}$$

$$\int_0^H \{ \Theta\Phi\cdot\Gamma_2 \}\, dz = - \int_0^H \{ \delta L/\delta Cb \}\, dz \tag{2.31e}$$

and the final condition on Γ_3 is determined as

$$\Gamma_3(z,T) = 0 \tag{2.31f}$$

Introducing the following variable change

$$\Gamma_1 = D\cdot\Gamma_{2z} \tag{2.32a}$$

$$\Gamma_2 = \Phi^* \tag{2.32b}$$

$$\Gamma_3 = - E^* \tag{2.32c}$$

$$\Gamma_4 = - P^* \tag{2.32d}$$

the equations (2.31) may be written as

$$- \left[D\cdot\Phi^*{}_z \right]_z + (\Sigma_a - \nu\Sigma_f)\cdot\Phi^* = - \delta L/\delta\Phi - \nu\Sigma_f\cdot(P^* + E^*) \tag{2.33a}$$

$$E^*{}_t = - \delta L/\delta E - \delta\nu\Sigma_f/\delta E)\Phi\cdot(P^* + E^*) - (\delta\Sigma_a/\delta E - \delta\nu\Sigma_f/\delta E)\Phi\cdot\Phi^* \tag{2.33b}$$

$$P^*{}_z = 0 \tag{2.33c}$$

$$\int_0^H \{ \Theta\cdot Cb\cdot\Phi\cdot\Phi^* \}\, dz = - \int_0^H \{ \delta L/\delta Cb\cdot Cb \}\, dz \tag{2.33d}$$

with the final and boundary conditions given by

$$E^*(z,T) = 0 \tag{2.33e}$$

$$\Phi^*(0,t) = 0 \tag{2.33f}$$

$$\Phi^*(H,t) = 0 \tag{2.33g}$$

The left side operator in equation (2.33a) is singular, but its right side is not zero. For an equation of this type to have a solution, the right side must be orthogonal to the homogeneous solution of the singular operator [1], which is the flux in this problem. From equation (2.33c) follows that the Lagrange multiplier P* is a function only of the time variable. Following the above considerations the value of P* must be calculated

$$P^*(t) = - 1.0/Ph(t)\cdot \int_0^H \{ \delta L/\delta\Phi + \nu\Sigma_f\cdot E^* \}\cdot\Phi\cdot dz \tag{2.34}$$

P* plays the role of an eigenvalue for the power normalization, like does the boron concentration in the diffusion equation [1].

The singularity of the left side operator forbids to find the solution by direct inversion. It must be obtained by first discretizing (2.33a) and then replacing one of the rows of the singular matrix by the discretized form of the equation (2.33d). The resulting nonsingular system can now be solved directly with a single outer iteration using some well known numerical method.

When the state variables and the Lagrange multipliers are known the gradient may be computed from the equation (2.6) obtaining in our case

$$q_i = \int_0^T \{ [\, 4\cdot(Kq - Qd)^3 + w_a + E^* + P^* \,]\cdot\delta v\Sigma_f/\delta\alpha_i + [\, \delta\Sigma_a/\delta\alpha_i - \delta v\Sigma_f/\delta\alpha_i \,]\cdot\Phi^* \}\cdot\Phi\cdot dt, \quad i = 1,2 \qquad (2.35)$$

since the function L as well as the three last constraint equations depend on the control variables.

In a similar way the linearized set of state equations are

$$-\left[D\cdot\delta\Phi_z\right]_z + (\Sigma_a - v\Sigma_f)\cdot\delta\Phi = -(\delta\Sigma_{a0} + \delta\Sigma_{aGd} + \delta\Sigma_B - \delta v\Sigma_f)\cdot\Phi \qquad (2.36a)$$

$$\delta E_t = \delta v\Sigma_f\cdot\Phi + v\Sigma_f\cdot\delta\Phi \qquad (2.36b)$$

$$\int_0^H \{\, \delta v\Sigma_f\cdot\Phi + v\Sigma_f\cdot\delta\Phi \,\}\cdot dz = 0 \qquad (2.36c)$$

with the initial and boundary condition given by

$$\delta\Phi(0,t) = 0 \qquad (2.36d)$$

$$\delta\Phi(H,t) = 0 \qquad (2.36e)$$

$$\delta E(z,0) = 0 \qquad (2.36f)$$

The equation (2.36a) is similar to (2.33a) for Φ^*, and therefore the way to solve it is the same. In this case the eingenvalue is the perturbation in the soluble boron concentration, which is fixed according to

$$\delta Cb(t) = \frac{-\int_0^H \{(\delta\Sigma_{a\theta} + \delta\Sigma_{aGd} + Cb\cdot\delta\theta - \delta\nu\Sigma_f)\cdot\Phi^2\}\cdot dz}{\int_0^H \theta\cdot\Phi^2\cdot dz} \qquad (2.37)$$

To find the optimal search length first the perturbation equations are solved for a small perturbation of the control variables to determine the resulting perturbation in the state variables. Afterwards the optimal search lenght can then be computed by direct line search by minimizing expression (2.12).

The derivatives of the objective function L with respect to Φ, E and Cb can be computed from the definition equation (2.18).

The optimization solution is obtained as follows: First, an axial distribution of gadolinium is assumed, and the state equations are solved over a number of discrete time intervals. Then the Euler-Lagrange equations are solved from the EOC to the BOC, since the Lagrange multipliers boundary conditions are specified at the EOC. When the state variables and the Lagrange multipliers are known then the optimal search direction can be computed using the conjugate gradients method and the optimal search lenght can be determined after the first-order perturbation calculation is done. Finally the new estimate of the control variables are computed using equation (2.13). The whole procedure is repeated until further improvement is not possible.

3. Cross section library generation.

The first step to solve the optimization problem is to calculate a cross section library able to describe the physical properties of the gadolinium poisoned assembly with enough degree of accuracy.

As stated above we used the 1G-1D diffusion approximation to model the reactor core. It means that the cross section library must include the values of the diffusion coefficient D, the absorption cross section Σ_a and the fission yield $\nu\Sigma_f$. It is assumed, that in general, these parameters depend on the averaged fuel burnup E, the moderator and the fuel temperatures (Tm, Tf), the soluble boron concentration Cb, the number of gadolinium pins per assembly α_1 and the initial concentration of gadolinium in the poisoned pins α_2.

Of course, to take into account these dependences in detail is a too expensive task. Howewer, from physical considerations it is possible to reduce the amount of calculations to be done, making a set of suitable assumptions. Logically, this must be done in such a way that the more relevant features are preserved.

In our case it is assumed that Σ_a and $\nu\Sigma_f$ for the assembly are the result of the superposition of the material properties of an unpoisoned fuel assembly and the gadolinium rods. The unpoisoned fuel assembly cross sections will depend on the averaged fuel burnup, the moderator and fuel temperatures and the soluble boron concentration. On the other hand taking into account the high absorption of gadolinium it is assumed that its cross sections are only function of the fuel burnup, the number of gadolinium pins per assembly and its initial concentration. Similarly, it is also assumed that the boron effect is additive in the Σ_a cross section (Σ_B) and is independent of the burnup and the fuel temperature. This assumption is not made in the case of $\nu\Sigma_f$. Finally, it is assumed that the diffusion coefficient is constant in all the space and all the time.

From these assumptions the cross section library generation could be reduced to the calculation of two simpler spectral units. The first one is an unpoisoned fuel cell, which was burnt until 30200 MWd/TU. Afterwards at several selected burnup points some state calculations were performed for a set of values of the moderator and fuel temperatures. Similarly, at an intermediate burnup point some calculations were carried out for different values of the soluble boron concentration and the moderator temperature to determine the boron effect.

The second spectral geometry consists in a set of supercells, each of them with a different amount of poisoned pins per assembly as well as a different initial concentration of gadolinium. These supercells, with a central poisoned pin, are burnt in a same way as the unpoisoned fuel cell. The gadolinium cross sections are calculated then as the difference between the corresponding supercell and the unpoisoned fuel cell.

The values of Σ_{a0}, Σ_{aGd}, Σ_B, $\nu\Sigma_{f0}$ and $\nu\Sigma_{fGd}$ that take part of equations (2.24) and (2.29) were calculated using this scheme. In particular the term $\nu\Sigma_{fGd}$ accounts for the spectral interaction between the fuel burnup and gadolinium.

All the spectral calculations were carried out using the code WIMS-D/4 [2]. The geometry and composition data used correspond to the VVER-440 type reactor [3]. The enrichment of 3.6 at% of uranium was the selected one for the unpoisoned fuel cells. In all the cases the calculations were performed in an equivalent cylindrical geometry using the S_8 transport approximation and 36 energy groups [4]. This energy structure is able to handle gadolinium burnup problems. The poisoned fuel pins were divided in 6 different material regions. The Dancoff and Bell factors were computed from reference [5] as a function of the resonance energy and the water density. During the burnup 37 lattice calculations were carried out and in the supercell cases the POISON option was used. The selected ranges of the independent parameters are

E = [0, 30200] MWd/TU
T_m = [200, 300] °C
T_f = [385, 950] °C
C_b = [0, 6] g/Kg
α_1 = [3, 7] pins
α_2 = [3, 7] wt%.

When the cross section values were available the final step for the library generation was the fitting of some polynomial and exponential dependences to this set of values [6]. The basis data of this library are the resulting fitting parameters.

4. Obtained results.

A program system has been written to solve this problem in Turbo Pascal 5.5 for IBM PC/AT or compatible computers. The diffusion equation is solved by a simple factorization method [7]. The Euler-Lagrange equation for Φ^* and the perturbation equation for $\delta\Phi$ are solved as pointed out in the section 2 using a factorization method too [8]. The burnup and adjoint burnup equations are integrated by means of the improved Euler method, while the burnup perturbation equation is solved using the Euler method [9]. To find the optimal search length ϵ an accelerated direct search is combined with the modified Fibonacci search technique [10].

Using this system we have obtained a power peaking factor reduction from 1.55 to 1.36 setting $Qd(z,t) = 1.0$ for a reactor of 250 cm core length and a cycle length of 300 days. In this calculation 50 spatial meshes and 20 burnup steps were used. The starting guess for the gadolinium loading was chosen to be uniform with $\alpha_1(z) = 5$ pins and $\alpha_2(z) = 5$ weight percent. When $Qd(z,t) = 0.9$ was used the optimal solution found had a maximal power peaking factor of 1.31.

5. Conclusions.

A program system for the axial gadolinium distribution optimization has been developed.

The developed program system allows to determine the optimal gadolinium loading for the optimization of the core performance of a VVER-440 type reactor.

For the utilization of the program system a parameterised cross section library has been created using WIMS/D-4 spectral code.

The calculated cross section library is able to describe the material properties of the gadolinium poisoned fuel assemblies. In spite of the feasibility of improvements in the fitting functions, the obtained library adequately describes the physical behaviour of the fuel burnup in presence of gadolinium.

The obtained reduction in the power peaking factor is similar to that reported by Drumm and Lee in the original work.

The performed work constitutes an unavoidable step in order to achieve the necessary experience for solving three dimensional reactor optimization problems.

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EXPERIENCE IN RELOADING TECHNIQUES

(Session 4)

PWR OPERATION AND RELOADING: EDF EXPERIENCE AND DEVELOPMENTS

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Abstract

The large experience accumulated by EDF in PWR operation and reloading for about fifteen years required reliable and industrial techniques. Presently, about 54 units of 900 MWe and 1300 MWe PWR's are being operated through various fuel managements (three-batch cycle, four-batch cycle, plutonium recycling).

EDF has developed two sets of automatized computational sequences with automatic generation of input data and core calculations for both, the Loading Pattern (LP) optimization and initialization of input data (fuel reshuffling), and for reload related calculations (safety evaluation, start-up physics tests prediction, operating data).

As far as the LP search is concerned, it consists in a technique of "trial and error" based upon knowledge and which is under very severe constraints. Then, reload values prediction and core following are performed with codes and calculational methods which have a high level of qualification and calibration over the large experience of in-core measurements.

With respect to these different points, continuous efforts are done aimed at improving the overall reloading methods. Developments are being achieved at different levels.

Because of load following perturbations, on-line and off-line core power distribution followings are evaluated with fast nodal CAROLINE code. This one is derived from the 3D design COCCINELLE code developed by EDF, and whose main features are 3D core calculations with optimized numerical schemes and fast resolution techniques, fuel thermal and neutronic feed-back effects modelling (pin by pin).

As an alternative to LP manual design used currently, EDF has examined two possible approaches : expert system and optimization package.

As far as automatic sequences are concerned, a new technique of automatic generation of input files was evaluated but priority has been given to improvements in physics by more 3D extensive calculations with the new COCCINELLE code.

INTRODUCTION

Technical and economical PWR performances are strongly influenced by fuel management, e.g. fuel utilization and core design. Because of the large number of standardized french PWR units, this question is of considerable importance for EDF. A large experience has been accumulated in PWR operation and reloading for about fifteen years which required reliable and industrial techniques. Furthermore, efforts are going on to improve the accuracy and performances of our tools for a better optimization of loading patterns.

The various conditions and specific characteristics of French PWR operation are first presented. Then, the accumulated reloading experience and technique in service are described. Finally, the different axes of development in progress are discussed.

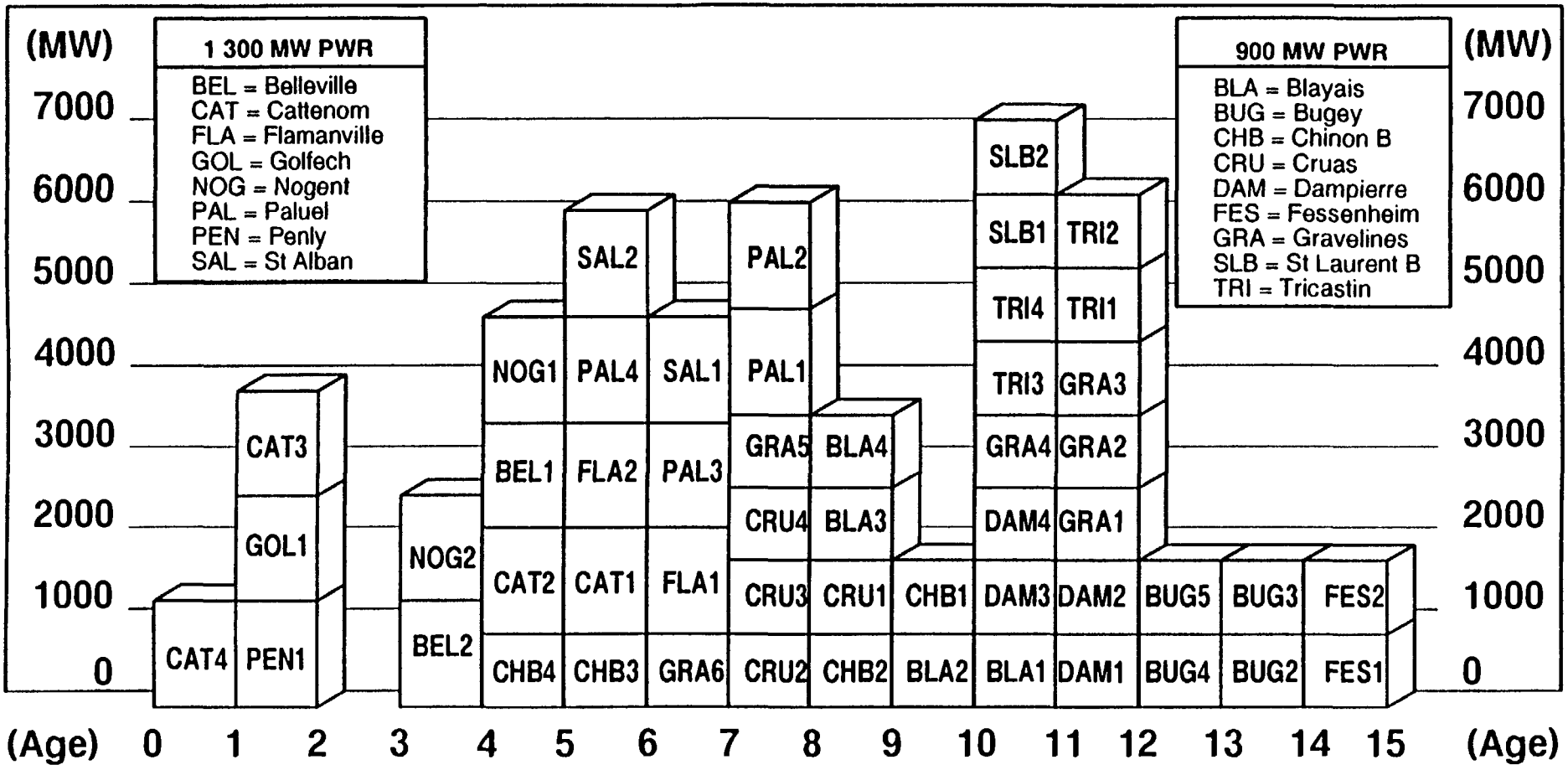


FIG. 1. Power histogram for PWR units (52 units, 426 reactor-years).

1 - MAIN CHARACTERISTICS OF FRENCH PWR OPERATION

1.1 - Number of units in operation

EDF presently operates two standardized types of PWR : thirty four 900 MW - class and twenty 1300 MW - class PWR's (Figure 1). The 900 MW class PWR units are of 3 - loop nuclear reactor design with a core composed of 157 fuel assemblies whose active length is 12 feet high. The 1300 MW - class PWR units are of 4 - loop nuclear reactor design with a larger core (193 fuel assemblies) and higher fuel active length (14 feet).

1.2 - Fuel assembly design

Fuel assemblies are of the 17 x 17 rod array design. They are delivered by different manufacturers, mostly by FRAGEMA, but also partly by ANF and SIEMENS - KWU for the 900 MW - class PWR units.

Advanced design with zircaloy grids is generalized. For MOX fuel assembly, three zones of different plutonium content are designed with the lowest plutonium concentration in the peripheral rods to limit flux increase at the interface with an UO_2 fuel assembly.

Reutilization of repared or restaured fuel assemblies, reloading of lead or experimental assemblies where some rods are extracted at the end of fuel cycles, need to take care of pin by pin power perturbation due to rod replacement. This perturbation is minimized if the damaged or extracted burnt fuel rods are replaced by a fresh rod with slightly enriched uranium

1.3 - Fuel management schemes

After first operation with 12 months three-bath fuel management (3.25% uranium enrichment), economical optimization has led to fuel management evolutions for 900 MW - class units.

Extended length fuel cycle

For some of them, extended length fuel cycles with 3.45% (64 feed assemblies) and 3.7% U235 (52 feed assemblies : three - batch loading scheme) was experienced. Burnable poison (gadolinium oxide) was used to compensate initial excess of core reactivity and flatten the core power distribution. Fuel reshuffling schemes were designed to meet the radial power peaking factor limits. Modification in the safety bank locations was possible to increase the shutdown margin because of original empty locations reserved for future plutonium recycle.

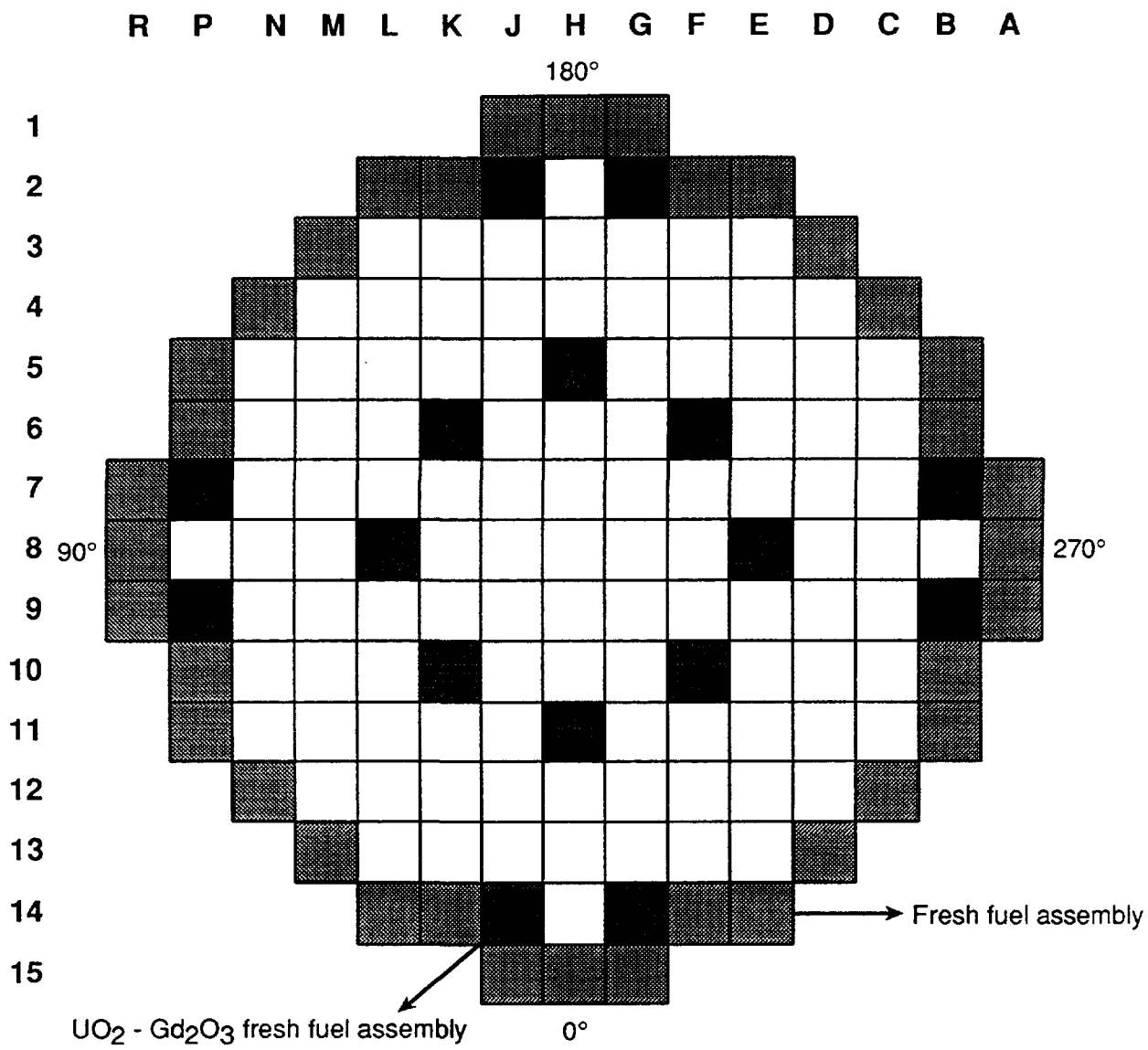
A typical three batch hybrid core loading pattern is shown on figure 2. Sixteen fuel assemblies are poisoned with eight gadolinium rods. This fuel management leads to an equilibrium cycle length of 330 EFPD and average discharge burn-up of 39 GWd/tU.

Quarter core fuel management

Taking advantage of a large feed-back experience on fuel behaviour, discharge burn-up has been increased up to 47 GWd/tU with four-batch fuel management, keeping the same 3.7% enrichment (40 fuel assemblies per reload). Further experiments about high burnt and cycling loaded fuel rod behaviour during accidents are requested by the Safety Authorities to increase the limit up to 52 GWd/ tU. A large part of the 900 MW - class PWR units is now operated with this fuel management, which does not require, any more poisoning of fuel assemblies. A fuel cycle length of 280 EFPD is obtained. The same control bank configuration is kept. An opportunity of vessel fluence reduction is given by reshuffling burnt fuel assemblies at the periphery of the core, in the appropriate position, to deliver lower power level in front vessel flux peak.

Plutonium recycling

Five 900 MW - class PWR units are loaded with MOX fuel. The reload contains 32 UO_2 fuel enriched by 3.25% and 16 MOX fuel assemblies whose plutonium content is adjusted, to get energy equivalence with 3.25% uranium fuel, because of isotopic fluctuations. This corresponds to a 30% recycling ratio and



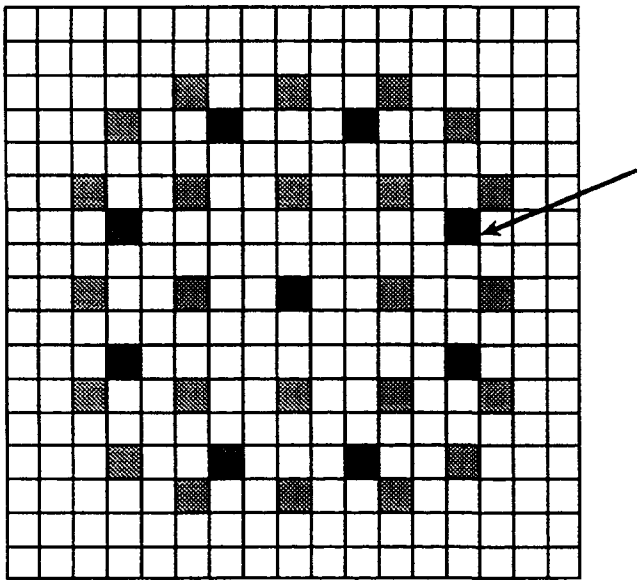
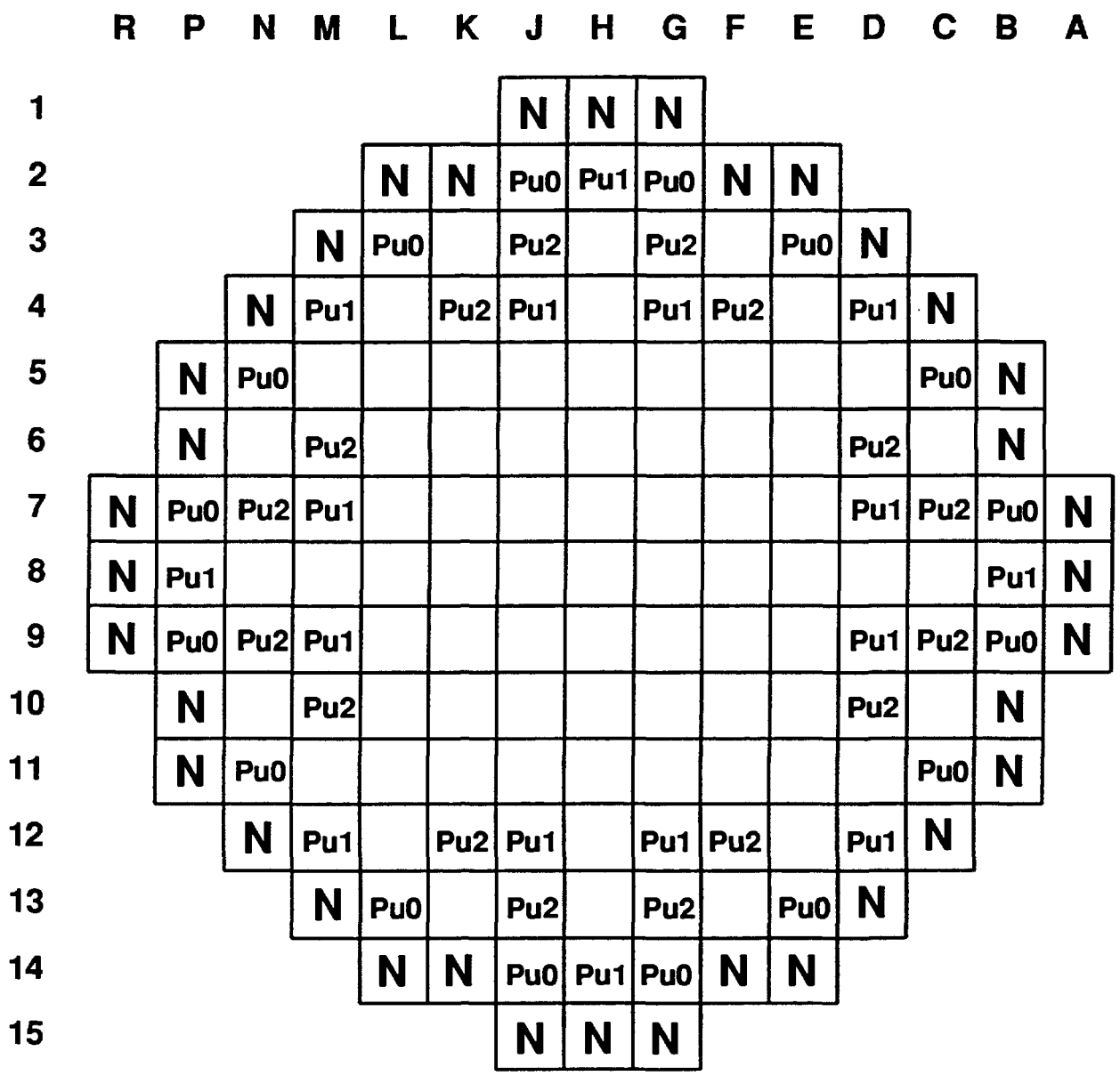


FIG. 2. Hybrid loading pattern for 3.7% fuel management (52 fresh assemblies).



N : Fresh UO2
Pu0 : Fresh MOX FA
Pu1 : ONE CYCLE SPENT MOX FA
Pu2 : TWO CYCLES SPENT MOX FA

FIG. 3. Plutonium recycling (30%) reloading pattern equilibrium cycle.

equilibrium cores contain 48 MOX fuel assemblies. A typical loading pattern is given on figure 3. Four safety control bank have been added to get enough shutdown margin, because of boron acid and rod worth decrease.

1300 MW - class PWR units

For 1300 MW - class PWR units, studies of new fuel management schemes have also been performed : 3.6% U235 four-batch fuel management, from 3.7% to 4.1% U235 three - batch fuel management with gadolinium poisoning. This last fuel management is very similar to the 900 MW - class PWR extended cycle fuel management experienced in the 1980's. Economical studies showed that an important advantage is derived from an extension of fuel cycle length up to 18 months, particularly for the 1300 MW - class cores, due to the recent increase in shutdown period durations. The goal is to improve plant availability and facilitate schedule of plant outages.

1.4 - Reactor operation flexibility

Because of the large increase of the share of nuclear power in the national electric generation capacity, schedule of overall plant outages has to be optimized. Furthermore, a high degree of reactor operation flexibility is needed for load follow and network frequency control, mostly with the FRAMATOME G-mode of operation. Early shutdown or stretch-out operation at the end of natural cycle length are frequently used to adjust the fuel cycle length to the planned shutdown period.

The number of feed assemblies per reload can slightly be reduced (because of moderator temperature coefficient safety related constraints) or increased (to extend the cycle length). We are looking for a higher level of flexibility, especially for increasing the number of feed assemblies in the four - batch fuel management scheme.

1.5 - Organization for reload calculations

The EDF Generation Division comprises a centralized "Nuclear Fuel Calculations and Core Analysis' team, located in Paris, which works, in close relation with the Nuclear Power Plants, with reliable, efficient and automatized tools to compute loading patterns and perform all the calculations associated with reloading reports. Calculations are submitted to IBM and CRAY computers operated by the computing center of EDF's Research and Development Division.

The main feature is that French PWR's fuel management is defined by referring to standardized schemes (three - or four - batches). Early shutdown or stretch-out operation at the end of cycle are used, together with the number of feed assemblies flexibility, to get an overall optimization of the outages for all the EDF units.

2 - EDF INDUSTRIAL AND RELOADING EXPERIENCE AND TECHNIQUES

2.1 - Operational and safety related constraints

A Loading Pattern (LP) must meet very severe constraints to be considered valid. The main aim is to flatten the core power map to minimize the power peak. The key parameters limiting values allowing the choice of a loading pattern are of different types.

The hot pin power (Fxy) in the ARO (All Rods Out) core configuration, and also in seven other configurations with control rods completely inserted in the core, must be minimized. Each rodded configuration consists in different number and type of control rods. These limitations come from hypothetic abnormal conditions (such as excessive load increase, boron dilution, rod withdrawal etc...) which are simulated to design the Δt core protection set points. As far as the EDF 1300 MWe PWR's are concerned, the LP radial peaking factor constraint is less strong because the reactor is on-line controlled by monitoring the DNBR (Departure from Nucleate Boiling Ratio) and LOCA Fq (Loss of Cooling Accident core hot point), using a new digital core protection system delivering higher margins. These two physics parameters, are computed both with the axial power distribution measurements by multiexcore detector and with all the radial power peaking factors Fxy(z) calculated versus core elevation (z) for each loading pattern.

The end of cycle Reactivity Shutdown Margin (RSM), which is computed with all the control rods (regulation and safety rods) inserted but the most antireactive which is supposed stucked at the top of the core, must remain above a minimum value to ensure an antireactivity reserve in case of a primary cooling due to credible steamline break.

The Moderator Temperature Coefficient , computed at beginning of cycle with zero power and ARO, must not be too positive so that it could be made negative using control rods insertion during normal operation.

The maximum assembly burn up must be under a limit which is imposed by the Safety Anthorities while waiting for experimental results for accidental behaviour.

Finally, a constraint is becoming more important in the LP search : it is the minimization of the vessel fluence during a cycle. This leads to locate burnt fuel assemblies at the periphery of the core, what is an opportunity of the four - batch fuel management for 900 MW - class PWR's (burnable poisons are not necessary).

The constraints values are not the same according to the type of fuel management : uranium management, plutonium management, 4-cycle, 3-cycle, extended length fuel cycle. The Fxy and RSM constraints are specific to the LP, the other parameters being generically controlled by the type of management (enrichment, cycle batching).

In this way, the LP search is under some key parameters limiting values which make it particularity restricting. This in an important difference between the French PWR's and what is done in other countries, especially in the USA where the nuclear reactor operate mostly in base load and are then under less severe constraints.

2.2 - Codes and calculational methods

EDF (Design and Construction Division SEPTEN, Research and Development Division DER) has elaborated a large set of codes and calculational methods in order to manage all the nuclear power plants.

The neutronic codes used are LIBELLULE (1D), JANUS (2D) and COCCINELLE (3D). They allow core and assembly diffusion calculation, a pin by pin option (being implemented in COCCINELLE), and then evaluate the main physics parameters characteristics of a given fuel management type. These codes use input neutronic data libraries which are set up with the APOLLO transport cell code elaborated at the French Commissariat à l'Energie Atomique (CEA) according to a sophisticated calculational procedure CNBIBLIO developed at EDF. This procedure links up a transport APOLLO calculation, a transport diffusion equivalence HERMES calculation and a JONAS calculation in order to dispose of a pin fine power distribution that could be superposed to the core homogeneous power distribution computed by JANUS or COCCINELLE. This is the calculational way to reach the core hot point factor (pin power) which is an important value to validate an LP.

The conditions of use for these codes are defined from the experience accumulated on all the reactors until now (Reference 2). These codes and methods, which have the same physical model to account for neutronic feed-back effects, are well experienced on a wide range of PWR core conditions. The same standard calculation route is used for cores with plutonium recycling but with finer meshes in the finite differences diffusion calculation. What can be underlined is that same standard calculational procedure is used for each reload. The qualification of computing results to the experimental measurements constitues an important milestone in the industrial use of the management scheme.

2.3 - Input burnup data

The LP computing, for a N + 1 cycle, needs to know some of the features of the end of cycle N. More particularly, we have to dispose of the N cycle length at the time of shutdown for refueling. We have also to dispose of the assembly average burnups at the end of cycle N. These burnups are computed, all the cycle long, by the means of flux maps recorded on the reactor every month. The LP calculational method is implemented generally two or three weeks before the plant shutdown. So, at this date, we have to estimate the real cycle length and also assembly average irradiations with a slight band of variation around these estimations in order to take into account possible operating deviations which would change the shutdown date. The computed LP must be valid for the extreme values of the band and for the estimated length cycle. If any

assembly happens to break when unloaded or loaded, then another assembly, stored in the spent fuel pool, is used, which shows close neutronic characteristics (burnup, enrichment). If it is not possible to find a spent fuel assembly of similar neutronic characteristics, then symetrical fuel assemblies have also to be discharged and all replaced by spent fuel assemblies, the loading pattern being reoptimized.

2.4 - Loading pattern optimization

Once the end of cycle assembly burnups are given (i.e the assembly reactivities) the LP computing may start.

For this, library (data basis) is available containing a set of LP got from the experience accumulated on different types of fuel management. From this library, an initial LP is chosen which is close to the operating conditions and the management type of the considered plant. On this initial LP, a "trial and error" technique is repeated by the engineer until the results meet the constraints. The experience acquired by the engineer lets him choose the assembly permutations to be done. The reshuffling is processed automatically by TRIDENS software.

For each iteration, the LP is evaluated with diffusion codes which allow to compute the physics parameters which have to meet the key parameters limiting values. As long as a parameter does not meet the criterium, the LP is considered as not valid, and a new iteration must be processed. For a standard management, a few iterations are enough to get good results. For a management whose operating has been disturbed (long stretch-out period or/and leaking fuel assemblies) a larger number of iterations may be necessary.

In case of problems during the reloading period for a new cycle, the computing of a new LP may be achieved in one or two days.

Taking benefit of a CRAY computer, fast calculations can be implemented at low cost. When the LP has been set out, EDF submits it to the fuel vendor (FRAGEMA, others) and the confrontation of EDF and vendors results ensures the quality of the LP.

When the fuel loading is being implemented on the site, the MAC software establishes automatically the assemblies loading sequences in the core from the basis of the LP computed by the Nuclear Fuel Calculations and Core Analysis. Once the fuel is loaded in the core, a magnetoscopic recording allows to verify the conformity of the core loading to computed LP.

2.5 - Fully automatic sequences for reload calculations

EDF has developed an automatic calculational scheme both for the LP optimization and calculation of the physical characterics of the reload. As far as the LP search is concerned, two steps are automatized at present : the initialization of input data and the different neutronic diffusion calculations allowing LP validation.

The burnup distribution is generated by the TRIDENS software which allows the assemblies reshuffling from a matrix given by the engineer. Then, to test the validity of the LP, the software RDP (Rechreche De Plan), developed at EDF, allows to link up different calculations : a core evolution to the end of cycle in order to get the irradiation radial distribution and the ARO radial peaking factors for the whole cycle length ; critical calculations with rods inserted to get the radial peaking factors in various rodded configuration ; and finally the RSM computing at the end of the cycle.

Once the LP is found, a whole set of calculations has to be implemented to establish three types of reports : a safety report, which is sent to the Safety Authorities, a Start-up physics test prediciton report and an operating data report, these reports containing the reference values for the plant operating.

All the calculations related to these reports are achieved and linked up automatically by DIAPASON software which runs on IBM but calculations involving the neutronic codes are submitted to the CRAY computer. The data files to initialize a DIAPASON computing are generated by TRIDENS which can process different types of file (irradiations, densities, assembly homogeneous, 1D, 2D, 3D, pin, by pin...). There are about 29 calculations to elaborate the physics fundamental characteristics of the concerned cycle, 42 safety calculations, 17 calculations for start-up physics test prediction and 35 calculations for the plant operation. This automatic

sequence of calculations is managed by a set of rules which define what type of calculation has to be achieved at a given time according to a hierarchy which has been established before. Some calculations may be executed in parallel others must be sequentially processed. The engineer can ask for the whole calculations serie or for individual calculations.

During the progress of this automatic sequence, a table containing intermediate and final values is filled-up. These values are used as input for the next calculation or as output data. Then, listings of results are issued for each calculation in order to validate it. The last step of this whole reload scheme consists in the automatic edition of the various reports by using tables containing results values. Tables and figures are also automatically edited according to the same process.

3 - DEVELOPMENTS

Developments are being implemented to improve the environment of the whole management scheme. These developments concern both the calculational codes in use and the methods to process them.

3.1 - Input burnup data

The burnup follow, as it is done at present by EDF, is based upon a methodology which adjusts the average irradiations, computed with a neutronic 2D/3D code to pseudo-experimental irradiations derived from the successive power maps recorded on the core at full power. This burnup follow does not take into account potential perturbations in rodded assemblies, but operating periods with rod insertion are limited. The operating modes that are scheduled to be achieved on the futur reactors (X mode for the 1500 MWe N4 reactor) will be featured by a deep and frequent insertion of rods. So EDF will consider the elaboration of a new burnup follow methodology adapted to these operating modes for which rods are often inserted. To evaluate precisely the contribution of 3D calculations to burnup follow, EDF has chosen the CAROLINE simplified (Reference 3) version of the core neutronic 3D COCCINELLE code. This code allows on-line / off-line core burnup follow from measurements of operating parameters that are acquired on site.

The definition and qualification of a burnup follow methodology must lead to choose a computing system able to operate efficiently on site. To do this, it is necessary to evaluate the incidence, on the core physics parameters calculations, of the irradiations distribution which has been obtained, in this way, on a relatively long operating period (a few months) with important power perturbations variations as far as load follow is concerned. Two things have to be precised : the calculational method that will be chosen and the user's specifications of the on-site software taking into account its computing environment.

At present, a procedure, which define on-line / off-line burnup follow, is being achieved. Detailed studies of measurements recorded on the CRUAS 2 reactor operated with intensive load follow are in progress.

3.2 - A new 3D design code

In order to improve the operating flexibility of reactors managed according to many various fuel managements, an industrial tool has been developped at EDF, which allows detailed calculation of the power issued in any point of the nuclear fuel in various situations. Such is the COCCINELLE software function which solves, in 3D, the neutron diffusion equation coupled with an axial thermohydraulics module (Reference 4). This code allows to compute both square feet PWR cores and hexagonal feet advanced reactors. Calculations may be achieved both on homogeneous (assembly) and heterogeneous (pin by pin) geometry. They take into account neutronic feed-back modelling (fuel temperature, water density, boron, xenon). More particularly, a new pin by pin neutronic feed back modelling has just been achieved and is being qualified. This new pin by pin function will enable the operator to undertake a better analysis of fuel pin burnup follow. Moreover, a new samarium evolution modelling has been introduced in the code in order to take into account, in a more physical way, this poison at beginning of a new cycle. In the near future it is also planned to introduce in the code a new baffe and reflector modelling. In parallel, COCCINELLE, whose role will be to be used more and more in an industrial environment, particularly important efforts are made to lower the CPU time according to the computer (CRAY) on wich this software is processed. The choice of proved resolution methods (Finite Differences) for which vertorization can be developed has been implemented for standard calculation while new nodal type methods are being achieved and qualified for their specific efficiency.

3.3 - Loading pattern optimization

Until now, the LP search has been done at EDF in a manual way and the LP validation in an automatic way : the whole LP calculation is not entirely automatized. However, EDF is interested in a solution to fully automatize the LP calculation (optimization). As an alternative to the manual design used currently, there are two possible approaches : expert systems and optimization packages.

The expert system solution was the first approach EDF has undertaken with the help of the Computing and Applied Mathematics Division. The idea was to incorporate into the expert system the rules that define the LP validity. As far as the Fxy were concerned, the integration of the guidelines has been done, unlike the RSM for which the integration proves more difficult. Some successful tests have been achieved on a well defined fuel management type. They allowed a first selection of a LP's set which, however, had to be analyzed with standard core neutronic calculations in order to ensure a complete safety validation. To improve this expert system, it should have been implemented the possibility to compute other types of fuel management (plutonium recycling, extended length fuel cycle management ...).

The second way that has been explored to automatize LP optimization is under evaluation. EDF is interested in FORMOSA optimization software, developped at North Carolina State University, under the responsability of Pr Turinsky. This software uses the so-called Simulated Annealing method to optimize the LP and the Generalized Perturbation Theory to compute the physics parameters (cycle length, discharge burnups, radial peaking factors ...). After EDF having tested this software on a realistic case of a French reactor, FORMOSA has proved not to be able to take into account all the severe constraints EDF has to validate an LP. The collaboration between EDF and NCSU goes on in order to lift the limits of the software concerning these constraints. FORMOSA proves to be operational as it is and its potentiality of adaptation is real .

3.4 - Automatic sequences

A reload calculation (neutronic feasibility, safety, operating ...) is made of an automatic sequences scheme of mainly 1D and 2D calculations. EDF makes this scheme evoluate firstly as far as the physics definition is concerned. For this, 3D calculations, achieved with 3D COCCINELLE code, are progressively implemented in automatic calculations sequences. They will allow, at first, to have 3D realistic irradiations with fine axial modelization which will be used in safety studies. They replace some 1D and 2D calculations but they cannot substitute completely to them because some decoupling key parameters are used in a simplified synthesis 2D/1D method. In the near future, an important job will consist in defining what types of 3D calculations will be implemented and according to what methodology in the automatic sequences scheme for the high heterogeneity MOX cores. The introduction of 3D calculations sequences requires an important work, particularly for the computing environment because the number of concerned files is relatively important and the relations between calculations may be very complex.

At the same time, the TRIDENS software has been developped to generate input files automatically. It can deal with various types of files (1D, 2D, 3D, burnups, densities, experimental burnups ...). In the near future, a new functionnality will be developped in order to work with 2D/3D pin by pin files, which is necessary because of the introduction of 3D calculations in the automatic sequences scheme.

In the near future, EDF is interested in the possibility of implementing on work station the automatic sequences scheme which runs, at present, on IBM computers, the calculations themselves being processed on CRAY. This implementation could improve the operating flexibility of the whole scheme and bring down the file stockage costs. However, the feasiblity and the interest of such a solution are to be proved.

CONCLUSION

While working on more and more specific fuel managements, EDF must go on ensuring a high degree of quality in managing the plant reloadings.

Aiming at this, a whole set of codes and methods allowing better core behaviour modelling have been developped, and for which the experimental feedback allowed to define user's conditions.

The computing of a fresh fuel reload is largly automatized. Firstly, the LP search is made of a "manual" optimization step (optimum assembly reshuffling) and an automatized validation step (safety constraints respect). Secondly, once the LP has been computed, the calculations needed for the reload are sequenced in an entirely automatized way. Then it ends by automatic plotting and drafting for the reports which will contain the reference values to plant operating.

All this automatic scheme is constantly improved by EDF. At term, developments concern many points of this scheme. First of all, the definition of input burnups will be precised by taking into account, in a better way, the plant realistic operation, particularly for the N4 future reactors. Moreover the core modelling will be improved by the progressive introduction of a finest core description (3D, pin by pin). The use of efficient numerical methods (nodal methods) will allow a gain of calculational time and cost, while keeping the same precision in results. As far as the LP optimization is concerned, EDF should take benefit from a deeper knowledge of the optimization techniques that are developped outside, the aim being to implement a fully automatized LP search.

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PROGRESS OF IN-CORE FUEL MANAGEMENT AT THE QINSHAN NUCLEAR POWER PLANT

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Abstract

For the in-core fuel management of Qinshan NPP (PWR, 300 MWe), the first one in the mainland of China, the relevant computer code packages have been developed and validated. The calculated results for the first cycle of the NPP indicated a good coincidence with the experiment data. The consideration of the reloading plan and scheme of the NPP has been described. A complete database about operation parameters and fuel accounting is planning to be created.

I. Introduction

Qinshan Nuclear Power Plant (QNPP) is the first PWR type nuclear power plant designed and built by our country. Its thermal power is 966 MW, and electric output 300 MW. Its initial startup began in November 1991. From then, the operation power has been raised step by step, with success in generating electric power on Dec 15, 1991.

The reactor core is loaded 121 fuel assemblies, see Fig. 1. The fuel rods in each assembly are in the form of 15 X 15 array. Each fuel assembly consists of 204 fuel rods and 21 water holes. The water holes can be mounted in-core flux measurement tube, control rods or burnable poison tubes as requirement. 30 in-core measurement tubes are inserted in the central holes of 30 fuel assemblies, respectively. The rest 20 holes of each assembly can be used for mounting 20 control rods or varying number of burnable poison tubes. There are 37 control rod clusters in the core. Each cluster has 20 control rods. According to their different functions, they are further classified into 6 groups. Each group contains 4 to 8 control rod clusters, see Fig. 2.

II. Organization for In-core Fuel Management

With QNPP's power rising and future normal operation, the task of in-core fuel management will be getting more and more important. Because our nuclear power plant is just beginning to be developed, our practical experience on in-core fuel management for nuclear power plant is limited. In order to ensure the safety of the QNPP operation, our government and related authority always attach great importance to the in-core fuel management of our first nuclear power plant. In the implementation of the in-core fuel management for QNPP, the following measures is being adopted:

a. A technical coordination group which is composed of the experienced experts from operation organization and several nuclear power institutes is in charge of the in-core fuel management for QNPP.

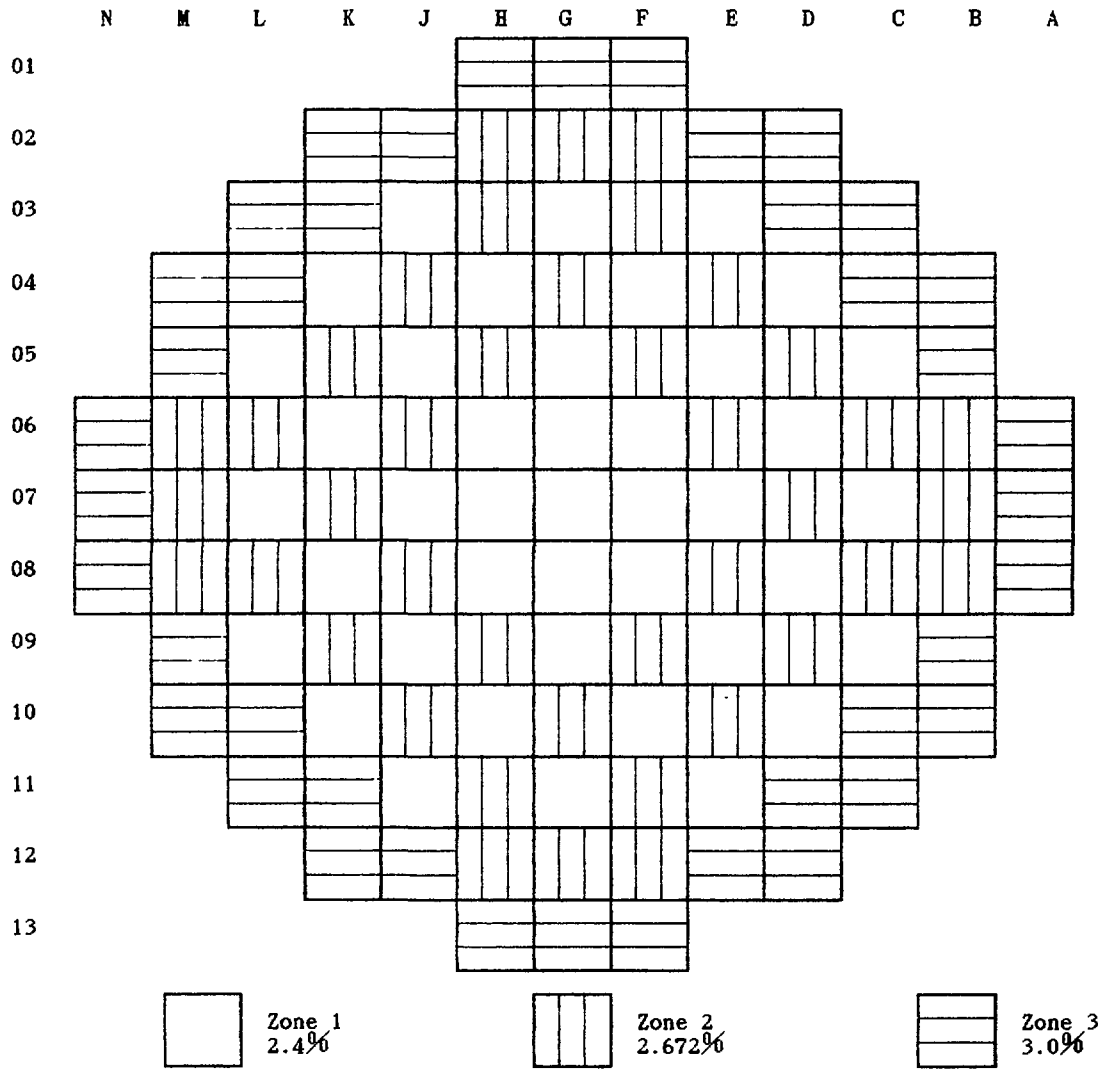

Fig. 1 The initial core loading pattern.

b. The computer codes for in-core fuel management and for treatment of experiment data have been developed.
c. The validity of the code packages is being verified. The comparison between the results computed by different code packages, and if possible, by experiment measurements has been carried out at the beginning of the first cycle.
d. The refueling plan and refueling scheme design will be performed.
e. A complete data base about the operation parameters and the fuel accounting is planning to be created, in order to quickly provide information for safe operation and refueling design at any time.

According to the necessity of the development of nuclear power in our country, Nuclear Power Institute of China (NPIC) and Shanghai Nuclear Engineering Research and Design Institute (SNEI) have separately developed their own codes for in-core fuel management of nuclear power plant. In the design of QNPP, the code package developed by SNEI made a big contribution. The code package made by NPIC was proved its validity in the verification computation of the first cycle of QNPP. It will be selected as a primary means of the future QNPP's in-core fuel management.

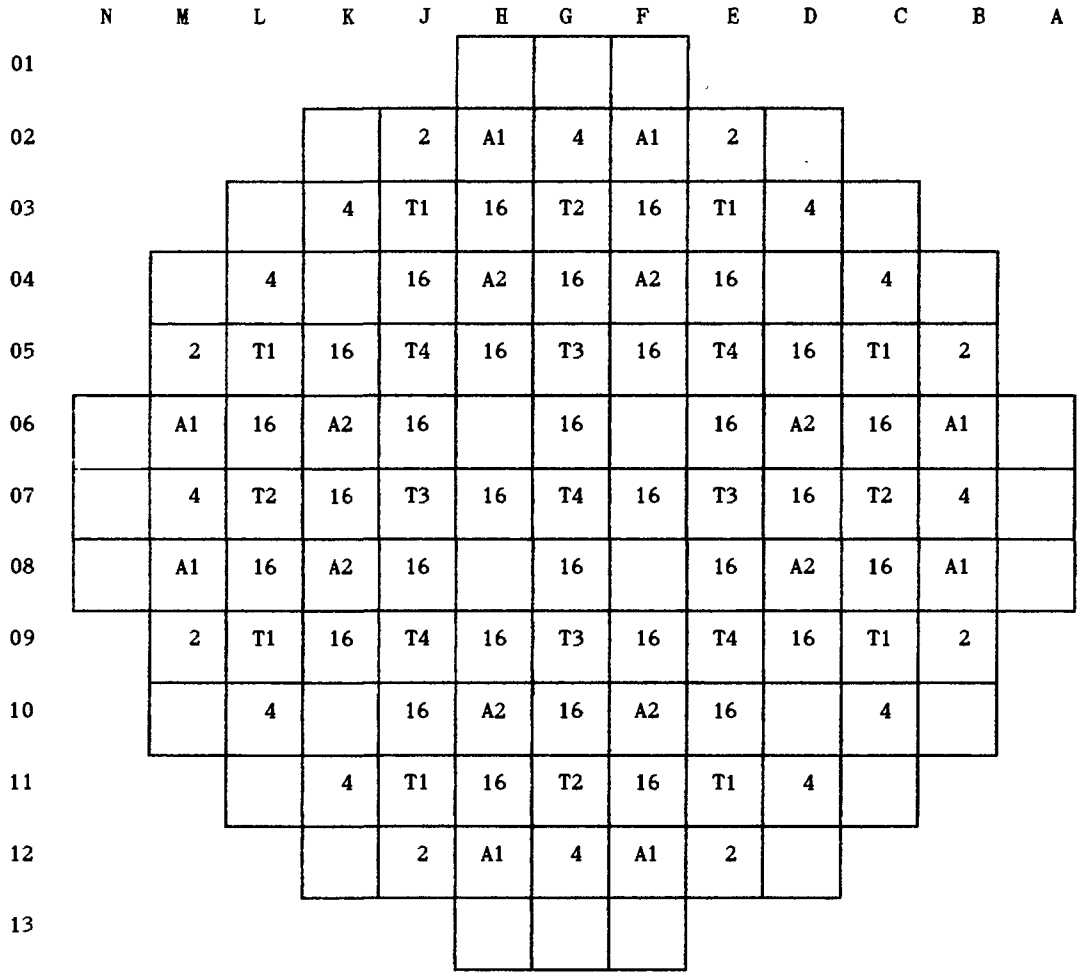

Fig. 2 The configuration of control rod bundles and burnable poison tubes in the core.

Scram rods : A1 8 clusters, A2 8 clusters.

Regulating rods : T1 8 clusters, T2 4 clusters,

T3 4 clusters, T4 5 clusters.

The figures denote the number of burnable poison tubes.

III. Comparison of Computations and Some Experiment Ressults

The comparison between the results obtained by two separate code packages shows that most of them coincide with each other.

Assembly computation results are basically consistent. As examples of the comparison between assembly computation results obtained by two code packages, the normalized power distribution in fuel rods within assemblies of 2.4% and 2.672% enrichments, with and without burnable poisons, are given in Fig.3 and Fig.4. The maximum deviation of normalized power distribution in the fuel rods is 4.9% for assembly without burnable poisons, and 8.5% for assembly with burnable poisons. The comparison of the local power peaking factors in three types of assemblies calculated by two codes is given in Table 1. The maximum deviation of them is 3.2%. The different

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1.010	0.998	<---------- By NPIC code					
1.010	0.984	<---------- By SNEI code					
0.995	0.997	1.009					
0.990	1.025	1.015					
1.024	1.014	1.034	-----				
1.016	1.040	1.066	-----				
-----	1.042	1.046	1.046	1.026			
-----	1.041	1.080	1.027	1.015			
1.026	1.034	-----	1.026	1.020	-----		
1.036	1.039	-----	1.016	1.000	-----		
0.987	0.994	1.007	0.991	0.987	0.993	0.969	
1.030	0.989	0.973	0.951	1.013	1.001	0.992	
0.970	0.970	0.971	0.969	0.966	0.960	0.953	0.947
1.011	0.986	0.997	0.959	1.016	0.994	0.961	0.993

Fig 3. Normalized power distribution of fuel rods within assembly with 2.4% enrichment.

Condition: Hot, full power, no control rods, clean core at BOL

Number of burnable poisons : 0

1.134	1.108	<---------- By NPIC code					
1.045	1.100	<---------- By SNEI code					
1.092	1.076	1.026					
1.053	1.032	1.017					
1.093	1.055	0.960	-----				
1.117	1.014	0.929	-----				
-----	1.037	0.926	0.891	0.913			
-----	1.043	0.914	0.894	0.929			
1.055	0.984	-----	0.915	0.914	-----		
1.063	0.959	-----	0.917	0.912	-----		
1.029	1.007	0.958	0.975	0.973	0.953	0.998	
1.054	1.024	0.966	0.976	1.001	0.954	0.999	
1.033	1.029	1.019	1.013	1.012	1.017	1.029	1.039
1.065	1.048	1.058	1.014	1.052	1.022	1.075	1.114

Fig 4. Normalized power distribution of fuel rods within assembly with 2.672% enrichment.

Condition: Hot, full power, no control rods, clean core at BOL

Number of burnable poisons: 16

cross section libraries used in the two code packages and their different treatment methods of cross section are principal source for these deviations.

4-group, 2 and 3-dimensional computations obtained the core power, burnup and moderator temperature distributions at every burnup step. The reactor physical characteristics at BOL were also computed.

Some Keff computation results under different core conditions obtained by two code packages are listed in Table 2. They are basically in agreement with each other.

Table 1. The local power-peaking factors of 3 types of assemlies under hot, full-power, clean core, no boron

	No BP		8 BP tubes		16 BP tubes	
Enrichment (%)	NPIC code	SNEI code	NPIC code	SNEI code	NPIC code	SNEI code
2.4	1.046	1.0803	-------	-------	1.136	1.1377
2.672	1.049	1.0682	1.078	1.1047	1.134	1.1170
3.0	1.052	1.0870	1.077	1.0760	1.132	1.1323

Table 2. Keff calcultion results by two codes

Core condition *	NPIC	SNEI
HZP, without BP, 0 ppm, clean core control rods out, T_m = 280℃	1.2338	1.2217
HZP, with BP, 0 ppm, clean core control rods out, T_m = 280℃	1.1641	1.1550
HFP, with BP, 0 ppm, clean core control rods out, T_m= 302℃	1.2128	1.2125
HFP, with BP, 1048 ppm, clean core control rods out, T_m = 302℃	1.0088	1.0000
HFP, with BP , 850 ppm, clean core control rods out, T_m = 302℃	1.0038	1.0000

* HZP = hot zero power, HFP = hot full power.

Table 3. Critical boron concentrations at BOL

Core condition	Computation	Experiment
HZP, with BPs, clean core all control rods out, T_m = 280℃	1305	1310
HZP, with BPs, clean core all control rods in, T_m = 280℃	463	----
HFP, with BPs, clean core all control rods in, T_m= 302℃	1024	----
HFP, with BPs, eqilibrium X_e all control rods in, T_m = 302℃	955	----

Table 4. The reactivity coefficients at BOL

Reactivity coefficients	Core condition	Calculation	Experiment
pcm/℃	HFP, with BPs T_m=302℃, 1100ppm no rods, clean core	-12.1	-11.47
	HFP, with BPs T_m=302℃, 0 ppm no rods, clean core	-55.1	
pcm/1%RP	Hot, clean core, no rods T_m=302℃ different power ranges		
	2% to 25% rated power	-13.8	
	25% to 50% rated power	-13.1	-13.0
	50% to 75% rated power	-12.6	-12.2
	75% to 100% rated power	-12.6	
pcm/ppm	HZP, clean core T_m = 280℃		
	1214ppm to 1173ppm	-10.65	-10.0
	1173ppm to 1068ppm	-10.87	-10.1
	1068ppm to 800ppm	-10.89	-10.3
	800ppm to 670ppm	-11.20	-10.6

Table 5. Reactivity worth of Control rods under HZP, at BOL

Boron (ppm)	Inserted rods	Keff	ΔKeff(%)	
			Calculation	Experiment
1214		1.010783	1.07	1.09
	T4	1.000054		
1173	T4	1.003311	0.48	0.45
	T4,T3	0.999435		
1068	T4,T3	1.007933	1.25	1.14
	T4,T3,T2	0.994466		
800	T4,T3,T2	1.016454	3.32	3.09
	T4,T3,T2,T1	0.979290		
670	T4,T3,T2,T1	0.990031	3.93	3.82
	T4,T1,A1,A2	0.951358		

Table 6. Some burnup calculation results of the first cycle

Days	Burnup MWD/TU	Boron ppm	Power peaking factor Pxy	Assembly of Pxy	Axial node number*
0.00	0	1214.00	1.2034	D11**	8
2.23	60	850.00	1.1529	G11	7
37.18	1000	811.35	1.1495	G11	7
74.37	2000	761.22	1.1593	F10	7
111.55	3000	706.93	1.1836	G09	7
148.74	4000	646.46	1.2034	G09	8
185.92	5000	580.14	1.2191	G09	8
223.10	6000	507.90	1.2318	F08	8
260.29	7000	430.59	1.2412	F08	10
297.47	8000	348.81	1.2481	G07	13
334.66	9000	263.00	1.2516	G07	14
371.84	10000	174.30	1.2511	G07	14
409.02	11000	83.35	1.2476	G07	14
442.63	12000	0.00	1.2411	G07	15

* Total number of axial nodes is 19, counting from bottom to top.

** The measured hot assembly

The critical boron concentrations under different conditions at BOL are shown in Table 3. We can see that there is a perfect consistency between the results of computation and experiment. The computed value is almost the same as measured under HZP condition, the difference is only 5 ppm.

From Table 4, we can see that the computed reactivity coefficients at BOL are good consistent with measured results.

The reactivity worth of control rods under HZP condition at BOL are shown in Table 5.

In Table 6, some results of the first cycle burnup computation are listed. Under HZP condition, the computed location of power-peaking is exactly the same as measured (D11 assembly).

Under HZP condition, with control rods out, computed value of the core power-peaking factor is 2.399, and experiment value is 2.388.

The normalized power distribution maps in the core plane under some different conditions are given from Figure 5 to Figure 7.

IV. Refueling Plan and some considerations

According to schedule, the first refueling of QNPP will be performed at the end of 1993.

As seen from Fig. 1, the initial core loading contains 3 types of fuel assemblies with U-235 enrichments of 2.4%, 2.672%, and 3.0%. It belongs to a 3-zonal outside-to-center scheme. 40 assemblies with highest enrichment

0.9743	0.8794	1.1151	1.0440	1.1645	1.1308	0.7490
0.9088	0.8281	1.0207	1.0328	1.1558	1.1927	0.8259
0.8797	1.0720	1.0067	1.1673	1.0351	1.1024	0.6185
0.8281	0.9739	0.9836	1.0998	1.0686	1.1155	0.6446
1.1156	1.0069	1.1233	1.0475	1.1515	0.9098	
1.0207	0.9836	1.0864	1.0715	1.1401	0.9490	
1.0448	1.1679	1.0477	1.2169	1.1319	0.6859	
1.0328	1.0998	1.0715	1.1882	1.2034	0.6842	
1.1955	1.0358	1.1520	1.1321	0.7607		
1.1558	1.0686	1.1401	1.2034	0.7975		
1.1318	1.1033	0.9103	0.6862			
1.1927	1.1155	0.9490	0.6842			
0.7497	0.6191	<---- By SNEI code				
0.8259	0.6446	<---- By NPIC code				

Fig. 5 Normalized power distribution of assemblies under HZP, clean core, all control rods out.

1.2360	1.1637	1.2437	1.1933	1.1476	1.0299	0.6729
1.2268	1.1450	1.2302	1.1862	1.1346	1.1013	0.6957
1.1637	1.2458	1.2189	1.2049	1.0920	0.9683	0.5400
1.1450	1.2318	1.2105	1.1895	1.0900	0.9761	0.5571
1.2437	1.2189	1.2226	1.1479	1.0645	0.8121	
1.2303	1.2105	1.2066	1.1425	1.0516	0.8266	
1.1933	1.2049	1.1479	1.1095	1.0097	0.5710	
1.1862	1.1895	1.1425	1.0961	1.0241	0.5871	
1.1476	1.0920	1.0645	1.0097	0.6533		
1.1346	1.0900	1.0516	1.0241	0.6678		
1.0299	0.9683	0.8121	0.5710			
1.0413	0.9761	0.8266	0.5871			
0.6729	0.5400	<---- By SNEI code				
0.6957	0.5571	<---- By NPIC code				

Fig. 6 Normalized power distribution of assemblies under HFP, equilibrium xenon at middle of core lifetime

1.1951	1.1611	1.2030	1.1961	1.1365	1.0352	0.6970
1.2476	1.1822	1.2353	1.2064	1.1294	1.0261	0.6871
1.1611	1.2025	1.2103	1.1803	1.1156	0.9747	0.5664
1.1822	1.2421	1.2328	1.1899	1.1051	0.9637	0.5527
1.2030	1.2103	1.1903	1.1588	1.0594	0.8270	
1.2353	1.2328	1.2060	1.1575	1.0412	0.8177	
1.1961	1.1803	1.1588	1.0934	1.0048	0.5893	
1.2064	1.1899	1.1575	1.0807	1.0041	0.5799	
1.1365	1.1156	1.0594	1.0048	0.6628		
1.1294	1.1051	1.0412	1.0041	0.6539		
1.0352	0.9747	0.8270	0.5893			
1.0261	0.9637	0.8177	0.5799			
0.697	0.5664	<---- By SNEI code				
0.687	0.5527	<---- By NPIC code				

Fig. 7 Normalized power distribution of assemblies under HFP, equilibrium xenon, at EOL .

are loaded in a zone on the periphery of the core. There are 13 fuel assemblies with enrichment of 2.4% in the central zone. 28 assemblies with enrichment 2.4% and 40 assembles with enrichment of 2.672% are loaded in the mid-zone in a scatter pattern. With this initial loading, a flattening power distribution can be obtained. Our ultimate goal is to achieve an equilibrium refueling pattern. The planning refueling scheme is as follows.

At EOL of the first cycle, apart from one which has lowest burnup, all assemblies with initial enrichment of 2.4% will be removed from the core. Their vacancies will be occupied by the assemblies with initial enrichment of 2.672%, and the assemblies with initial enrichment of 3.0% will be moved to the mid-zone, then 40 fresh assemblies with enrichment of 3.4% will be loaded in the outer zone. In this 3-zonal refueling pattern, after a complete fuel cycle (i.e. 3 reactor cycles), an equilibrium refueling pattern is expected to be achieved. After equilibrium refueling, the mean maximum burnup of withdrawn fuel assemblies is about 30000 MWD/TU. The plan equilibrium refueling period is about one year. The fuel burnup data at EOL of the first cycle are listed in Table 7.

Table 7. The burnup of fuel assemblies at EOL of 1st cycle

Core life-time (day)	for 0 ppm boron	442.6
	for 25 ppm boron	432.2
Average assembly burnup (MWD/TU)		11632
Average burnup of withdrawn assembly (MWD/TU)		13252
Withdrawn assemblies	highest burnup (MWD/TU)	13886
	lowest burnup (MWD/TU)*	12376

* Keeping in the core for 2nd cycle utilization.

In the refueling design, the most important criterion which must be satisfied is that the thermal limitations (DNB limit and coolant enthalpy up limit) can not be exceeded. In a reloaded core, the worst power distribution during the whole core lifetime can not produce any problems on the operation safety. In other words, the power peak in the core should always be as low as possible. Under the essential premise, it is desired that we make the withdrawn fuel assemblies reach their maximum burnup, so as to obtain a better economical efficiency at lower fuel cost.

The development of QNPP in-core fuel management will be from the trial-and-error method to the final optimal method. In view of the operational safety, we will use the power peaking factor as our object function, that is, the reloaded core must have the lowest power peak. We are planning to apply a direct search technique in determining the reload core configuration. If possible, some once-burnt and twice-burnt fuel assemblies can be rotated to take account of the burnup unuiformity in fuel assemblies in order for a flattening power distribution in the reload core.

Apart from partition-loading pattern, another approach for flattening core power is appropriate choice of configuration of burnable poisons and control rod programming. On radial power flattening, the burnable poisons in

assemblies located in the outter zone should be of centripetal configuration. On axial power flattening, through shortening axial size of burnable poison rods and selecting an improved control rod programming, we can obtain a better core power distribution.

In the future, the important task accompanied by the in-core fuel management for QNPP will be accumulation of the experience of optimization refueling and the development and modification of the computation codes for optimal refueling. In addition, we shall make researches on the low-leakage loading pattern in QNPP.

FRAGEMA'S IN-CORE FUEL MANAGEMENT PRACTICE AND ASSOCIATED TECHNIQUES TO OPTIMIZE QUALITY AND EFFICIENCY OF SAFETY EVALUATIONS

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Abstract

Since the beginning of the eighties, the fuel utilization and discharge burnup in Pressurized Water Reactors (P.W.R.) have been considerably increased while improving fuel reliability and performance, and maintaining the high level of safety of nuclear power plants. This trend is accompanied by a large diversification in the manufactured fuel assembly product, and also in the strategies of in-core fuel management. In France, FRAGEMA, which supplies most of the fuel reloads, is also in charge of the associated safety evaluations.A safety evaluation involves many interconnected calculations with different codes.The complexity of this task has required to design an expert system to perform these evaluations with entire reliability. This paper describes our experience in this area of activity through the last ten years.

Introduction

A safety evaluation, whose main aim is to ensure that the applicable plant safety limits as defined in the Safety Report are respected, involves many interconnected calculations with different codes. Its basis is the reference studies contained in the Safety Report. Due to the large diversification of the in-core fuel managements and/or the specific features of each reload composition, the reference studies have to be modified. As a result, a large set of safety evaluations has to be performed. FRAGEMA has introduced the use of the "knowledge-based system" techniques and tools to manage easily these calculations.

1. Reference studies

The Safety Report is submitted by the utility to the Safety Authorities so that an authorization decree for plant construction and operation can be granted. It includes the applied safety rules, the accident studies and the technical specifications. It is based on engineering studies relying upon hypotheses which must be verified throughout plant lifetime and which center on the unit operating mode, the state of the plant systems and the core characteristics. To avoid penalizing an entire standardized plant series through extreme operating conditions, the hypotheses are chosen to adequately cover all the states of the different plant units.The typical reference condition for PWR's is the third core annual reloading with a batch average burnup of about 33 GWd/t(U) and an enrichment of about 3.25% U235.

In practice, the operation of a nuclear power plant (N.P.P.) is subject to a number of constraints whose effect is to more or less drastically modify the hypotheses of the Safety Report accident studies and the management characteristics considered in these studies. These constraints include :

- management flexibility (cycle extension or shortening);
- the introduction of a new fuel design;
- changes in the number of assemblies reloaded;
- changes in the feed enrichment;
- changes in system characteristics;
- changes in operating conditions.

Table 1 : Number of safety evaluation calculations

900 MWe - EDF	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991
3.25 % 1/3 core	2 A	5 A	8 A	11 A	11A 1 G	5 A 11 G	6 A 13 G	6 A 17 G	5 A 14 G	5 A 4 G	1 A 2 G	1 A 2 G	3 A
3.25 % 1/4 core										1 G	3 A 3 G	4 A 3 G	2 A
3.45 % pyrex					1 A								
3.45 % gadolinium					1 A	2 G	1 G						
3.7 % 1/3 core								1 G	1 G	1 G			
3.7 % 1/4 core									2 G	7 G	10 G	12 G	18 G
M3 power increase										1 G	2 G	1 G	
MOX 3.25 % 1/3 core									1 G	2 G	4 G	4 G	5 G
1300 MWe - EDF													
3.10 % 1/3 core								3 G	5 G	7 G	11 G	7 G	13 G
3.10 % 1/4 core													1 G
EXPORT	1 A	2 A	1 A	2 A	2 A	1 A	1 A	3 A	4 A	4 A	5 A	4 A	4 A
TOTAL	3	7	9	13	16	19	21	30	32	32	41	38	46

A : Mode A; G : Mode G

Each reloading is therefore accompanied by a safety evaluation, whose importance is dependent on the deviations from the hypotheses of the generic studies and on the margins which may exist in these studies. The first step in this evaluation is to compare the neutronic parameters of the new reload with the values used in the Safety Report and to verify that the hypotheses adopted for these safety studies are not compromised by N.P.P. operation or state. Should the recalculated parameters prove more penalizing than the generic parameters or some hypotheses no longer be verified, then the second step is to perform a complete safety analysis for the relevant accidents. When major changes arise, from the operation or nature of the reload, and exert a perceptible impact on the accident study hypotheses, rather than adding to the reload calculation load it is preferable to re-analyse the affected accidents in generic studies and to draw up a new list of key parameters (Reload Safety Analysis Checklist - R.S.A.C.), whose justification will be found in an addendum to the Safety Report.

2. Safety evaluation studies

When a pressurized water reactor is shut down at the end of a cycle (now once per year in France) refuelling takes place. This operation involves unloading of some assemblies, usually the most-spent, and their replacement by the same number of fresh fuel assemblies. Thus, the core is arranged in a new loading pattern with optimum distribution of the differently burnt assemblies, so that the most uniform possible power distribution is achieved. To guarantee safety for the next cycle by verifying a number of neutronic criteria related to the safety of the plant and to ensure the latter operates according to the technical specifications, FRAGEMA generally supplies a safety document to their customers. This document summarizes the results of the studies arising from this new loading scheme.

One safety evaluation represents more or less 50 sets of linked calculations, obtained from 3 main codes and numerous utility programs.These results are yielded by core neutronics computer programs which simulate core behaviour under varying conditions. The models are used to study accidents and specific operating modes. The design engineer has to generate the data files for these programs, carry out job batching (a CDC 960 and a CRAY Y-MP) and analyse the results of these calculations. He must also perform intermediate operations so that new parameters for an input file can be obtained. The final calculated values are derived from FRAMATOME's methodology and associated uncertainties, and then compared with the key parameters summarized in the R.S.A.C..The complex interlocking of these jobs and the time spent analysing the listing make these studies lengthy.

3. Automation of safety evaluation calculations

In France the 54 PWR's (CPY 900 MWe and PQY 1300 MWe) ordered in the 1970s were brought on line in an average time of 15 years and with a peak period around 1985. Table 1 shows the management of E.D.F. power generating capacity in terms of reloads from 1979 to the end of 1991. This rise in the number of on-line units translates into a rapid increase in the number of safety evaluation calculations : 2 in 1979, 21 in 1985, 46 in 1991. This increase in work load, the tightening of contractual deadlines and the repetitivity of the calculations led to planning of the progress of the tasks associated with a reload safety evaluation calculation, as early as 1982.

The two computer-based "tools" used up to the mid 1980's were updates and procedures dedicated to some particular tasks. These standard updates contained the set of files needed to run a reload calculation. They were accompanied by a hand-written user's manual describing the link between the inputs and the outputs of the different codes and also enabling the final values to be obtained. This approach had the merit of obliging all the engineers to use the same calculation methods. Given the diversification and increase in the number of the safety evaluations

in 1985, the technique of the standard input files and the rules to obtain the final values appeared limited. The time spent on these studies was highly variable, depending on the special features of the study. Finally, a checker spent a long time verifying the calculation. So, FRAGEMA developed an expert system (HADES) to fully automate study execution.

4. Diversity of the safety evaluations

In France, this diversity is reflected first of all in the reactor control mode. By satisfying the neutronic limits dictated by the safety criteria, this mode enables the target of the plant operating requirement to be achieved.

Mode A

Historically, this was the first reactor control mode. The load variations are performed without disturbing axial power distribution; the power defect is accommodated by the boron and core control is performed by the R.C.C.A's. To date, 6 units are operated in this control mode in France.

Mode G

About 65% of the electrical energy produced in France in 1985 was generated by nuclear power stations, essentially PWRs. Since 1983, French nuclear units have had to adapt to this new situation and no longer operate only in base load, which is the usual operating mode in most other countries. It is to meet this requirement that FRAMATOME developed a new operating mode known as "Mode G".

Four groups of control rods having a neutron absorption capability lower than that of the standard "black" control rods are used to adjust the output while causing less perturbation of the axial and radial flux than would be the case if standard control rods were employed. The boron concentration is adjusted only to compensate for the slow reactivity changes due to the xenon effect. In Mode G, the fine reactivity adjustments are made by means of an "R" or regulation group of control rods, which control the temperature of the reactor coolant. The variations of xenon reactivity induced by remote frequency control of the power grid are also compensated for by the R group. To date, 28 CPY 900 MWe plants are operated in this control mode.

For the PQY 1300 MWe reactors, the protection system has significantly advanced from the 900 MWe design. The Integrated Numerical Protection System (S.P.I.N.) receives data from the axial splitting of the ex-core chambers into sections. Based on the core measurements, the software is used to continuously determine all the physical parameters factored into the calculation of DNBR and of linear power on the one hand, and the margins from the core physics limits on the other. To date, 20 PQY 1300 MWe units are equipped with this type of protection.

These control modes, originally developed for a 1/3 core reload management, serve as a basis for the safety report accident studies. They are considered by the utility E.D.F. as a "must", to be retained and improved if possible when the fuel management strategies change.

Secondly, the diversification of the safety evaluations is also due to the diversity of fuel managements. In France, the deployment of new fuel managements is accompanied by generic studies, whose aim is to determine the optimum loading patterns for achieving the goals set by the customer. These reshuffling schemes must allow for the operational adaptability mentioned hereabove, together with cycle flexibility : groups of units are affected by these managements, so it is vital to have unit outages available on demand (-1 month to +2 months about the natural

cycle length). An overall strategy of increase to equilibrium, which combines natural cycle lengths, early shutdown and stretchouts is determined by joint agreement with the customer.

The studied loading patterns particularly define the strategy for positioning fresh fuel assemblies in the core. From this set of loading patterns, bounding kinetic coefficients are determined. These serve as a basis for accident re-analysis. Accordingly, each new fuel management may result in a more or less far-reaching amendment to the initial Safety Report and in the issue of a new list of key parameters. This approach limits the safety evaluation calculations to a comparison with a list of key parameters. The generic safety study, valid for a given management mode, is conducted once and for all.

Extended cycles

These cycles are characterized by an increase in the enrichment (3.45% U235) and the number of loaded fresh fuel assemblies (64), and by the use of discrete or lumped burnable poisons. The lumped burnable poison is gadolinium. The latest fuel management scheme is a further increase in enrichment (3.7% U235) accompanied by a reduction in the number of fresh fuel assemblies (52). Two gadolinium-bearing assembly designs differing in the position of the absorbing rods in the assembly were inserted into the reactor.

Mox managements

Spent fuel reprocessing is enabling the recycling of fissile products arising from the U235 fission chain, particularly plutonium. Given the plutonium specific features : higher neutron absorption and less delayed neutron production, the associated generic studies are extensive and there are more accident re-analyses. The first MOX reload was inserted in 1987 and since then the share of this type of fuel has steadily increased.

Quarter-core reload managements

Economic considerations are leading the utility to increase the reload split by adopting quarter-core reloads. This new management mode is currently implemented in the CPY 900 MWe series and will soon be extended to the PQY 1300 MWe series.

In parallel, N.S.S.S. changes may have impacts on the safety evaluation studies. This was the case for the 4.5% power uprating of some CPY units and it is also the case for the reactor coolant average temperature decrease in PQY units in 1992.

This description reflects the diversity and the quantity of the reload safety evaluation studies performed for the French N.P.P's.

Export situation

The export situation is generally different : each unit is unique in terms of reload safety evaluation calculations. The loading patterns are performed on a case by case basis in response to clearly defined goals. As a result, it is generally not possible to conduct generic studies to bound a particular case. Any accident re-analyses will be conducted during the reload safety evaluation.

5. HADES, an expert system for evaluating reload safety

The French context, through the quantity of data and their repetition, was favourable for the development of tools with the aim of optimizing quality and efficiency of the safety calcula-

tions. Thus FRAGEMA has developed an expert system for complete automation of engineering studies. By using the "knowledge-based system" techniques and tools, a significant automation threshold has been crossed, mainly through the all-purpose approach, by which all the old tools were integrated into a complete system. In this way, the updates and procedures of the first period were integrated into the expert system. To-day, the system performs the following tasks :

- automatic generation of input files for the different neutronics computer codes,
- management of job batching on an optimized basis,
- automatic data selection by means of output reduction codes,
- fully automatic connection to the computer network containing the codes,
- plotting of figures and drafting of the specific safety document for submittal to the customer,
- creation of a data base containing all the intermediate values specific to the study.

Its scope covers standard reloads and standard fuels, whose types are stated in the input data files. Nevertheless, through its design the system can also be used for special-purpose studies (fuel management : cycle sequencing), insofar as the required changes are made either in the input files produced by the system or in the knowledge bases, for more sophisticated functionalities.

An expert system commonly contains the knowledge of a person in a given area, the most important part of the system is contained in the "convert" module, in which the computing methods and neutronics code knowledge are coded. The simplest rules of this part are those which convert a data unit from a calculation into an input for another file. The most complex are capable of generating waves of variable-length input data from data batches originating in several jobs. The design methods may change with time, either through improved calculations/ measurement comparisons or through fuel management diversification. By representing this knowledge in rule form, the bases can easily be incremented. The rules make it possible to determine inputs called "variables", which are placed in "skeleton" files already containing the fixed inputs. With this method, it is possible to parameter any input or input pack when required and to retain clear legibility of the skeleton files.The system design allows its knowledge base to be easily expanded. This was clearly demonstrated in 1987 after the first hand-made safety evaluation for MOX fuel. All the knowledge was transferred into the expert system and the following safety evaluations were conducted by the system. Since then, the system has stood the test of time.

6. HADES development technical data

6.1. Hardware and Software

The system is run on SUN 3 and 4 workstations, with the UNIX operating system. The system and its connected files occupy a disk space of 20 Megabytes. The computers are a CDC 960 and a CRAY Y-MP. The software programs needed to run it on the workstation are :

- K1 : expert system development software. SUN-FRAMENTEC version
- FTP/TELNET : communication software
- for maintenance : C and FORTRAN compiler, MAKE utility programs and SCCS (Code source management).

The system comprises 26 C programs (externals K1 functions and multiwindow user environment), 11 outputs reduction codes (HACKERS), 30 knowledge bases and 26 skeleton files used by the bases. The set of knowledge bases represents about 7000 rules and facts of order one.

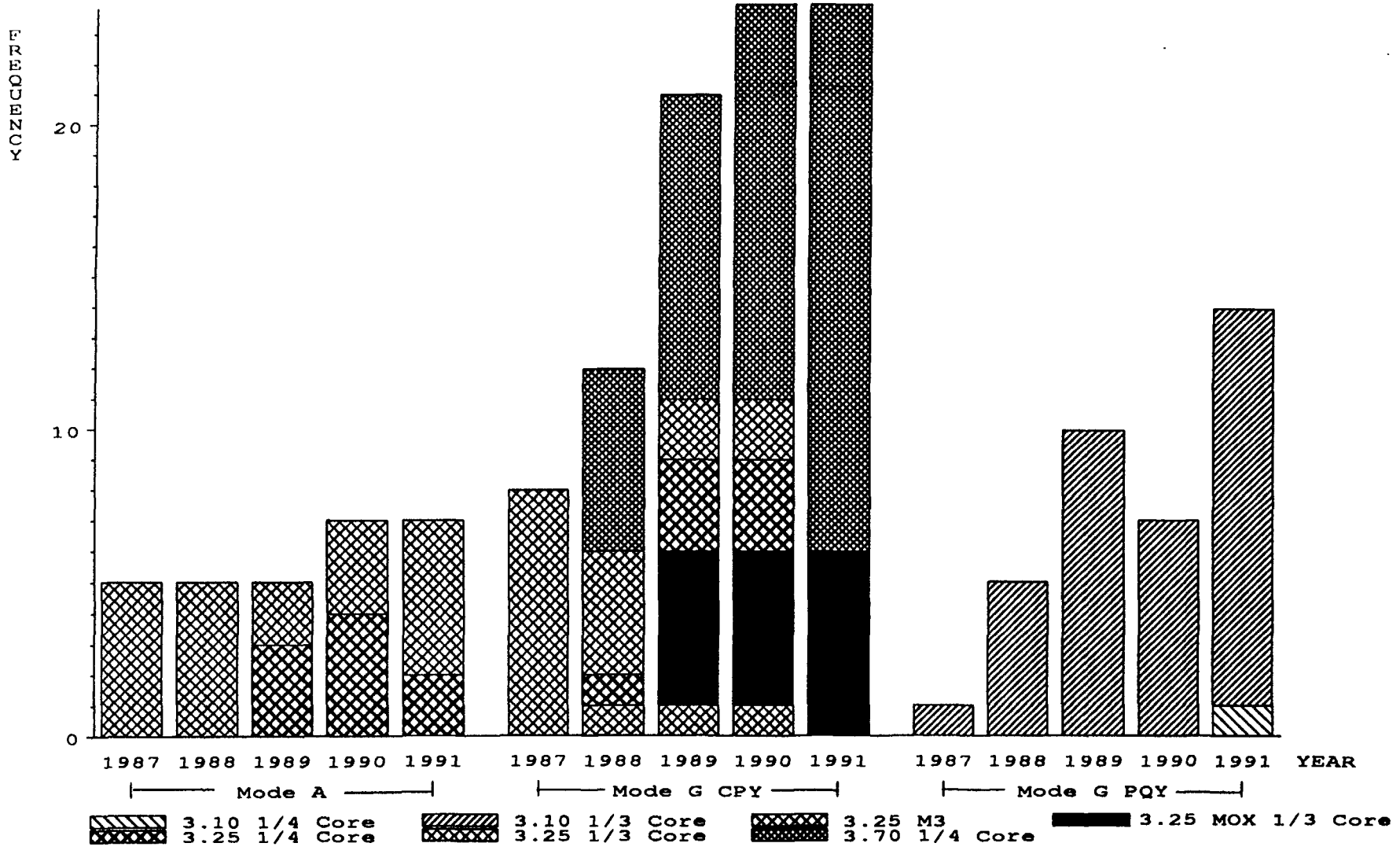

Figure 1 : Reloads calculations processed by HADES

6.2. Environment

In parallel with the knowledge bases, a multiwindow environment with dials has been developed. This relieves the user of the need to know the specific language of the expert system development tool used.

A dozen dials are available for selecting the calculation phase, the operation and control options, starting or stopping, or editing of a file. The user can also safeguard the memory, edit it, change it or reload it, with the latter keeping the same format as the data reduction code outputs. This operation is not routine, but gets around any specific problems which may arise. Note that the system has a large degree of autonomy and can run without a user. It is the sole judge of the actions it may undertake. It possesses full capability for connecting up to the computer network and running calculations. It is capable of analysing many of the operating errors and calls on the user only for solving unusual problems. The environment's main role is to control and drive the system.

6.3. Maintenance

Close attention was paid to the maintenance unit. An expert system has the great advantage of being readily modifiable for incrementing its base.

Nevertheless, for a big system, the factors related to the application operating warranty are vital if unwanted effects are to be avoided. The unit installed uses a source code control system (SCCS) which keeps a record of all the changes made to the modules.

A UNIX procedure ensures proper work by driving SCCS, the "Make" compilation and automatic updating tool, the editing programs and the test cases. The test cases are studies in which the calculations are not really started, operating with pre-defined reduced output files. A series of test cases can operate the system in configurations which are wide-ranging and representative of the possible studies. All the documents and files are generated and compared with the previous reference.

The changes made to the system are activated by a method manager, who centralizes all the proposals and upgrades and generates a document modification package. This package is then transmitted to the knowledge base administrator, who analyses its feasibility and generates a maintenance sheet. The modifications are then incorporated by one of the two people in charge of the package. Note that the entire set of maintenance sheets (about 250 to date) and all the technical documentation are also run on the workstation so that a record of the changes can be kept. To date, the system is capable of performing 11 different calculations. Figure 1 shows the distribution of these HADES calculations versus different parameters (reactor control mode, fuel managements,...).

7. Conclusion

The development of an expert system was necessary to cope with the increase in the number and diversity of the reload safety evaluations. The use of the HADES system has allowed a considerable decrease in reload safety study time. The different calculations are performed with a high confidence level, due to the standardization of methods, the safeguarding of knowledge and through continual comparison between the values calculated instantaneously and those safeguarded in the data base. Running since February 1987, it has been extended many times, both in its scope and in its internal computer techniques (network, test, on-line help,...). To date, it has been used to conduct about 180 reload studies.

The HADES tool is ready to easily incorporate all the major modifications in fuel management and/or the safety evaluations in the near future.

ALPS: AN ADVANCED LOADING PATTERN SEARCH PROGRAM

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Abstract

In-core fuel management continues to be affected by trends toward longer cycles, increased discharge burnup, increasingly complex fuel assembly features such as axial blankets and reduced-length burnable absorbers. Obtaining efficient core loading patterns which meet several, often conflicting, design criteria is becoming difficult to perform manually. To address this need, Westinghouse has developed the Advanced Loading Pattern Search program, ALPS, a self-contained fuel management tool. The computer program establishes a variety of loading pattern candidates, based on the existing fuel inventory of spent and fresh fuel assemblies, and burnable absorber configurations defined by the user. The possible loading pattern candidates are established based on a reactivity distribution target and an appropriate matching of available fuel and burnable absorber configurations. An extensive search algorithm evaluates thousands of alternatives, and selects the candidates that meet specific design criteria. This search approach offers several advantages over conventional methods, as no specific knowledge and/or constraints are fixed in the algorithm.

ALPS includes an innovative approach to take into account the reactivity effects of three-dimensional features. Such an approach permits accurate and fast representation of 3D features, using a two-dimensional core physics model. The software is now operational on a workstation computer, with an advanced graphical user interface to streamline input setup and output review.

1. INTRODUCTION

As fuel management evolves and advanced fuel assembly designs are developed, obtaining economical loading patterns which meet design criteria is a more difficult task to be performed manually. The Westinghouse advanced loading pattern search computer program, ALPS, represents the latest generation of automated methods for generation of such loading pattern.

2. METHODOLOGY AND CAPABILITY

The loading pattern generation techniques included in ALPS is an evolution of the LPOP [1] computer program developed at Westinghouse. The ALPS methodology incorporates several improvements derived from the extended usage of LPOP in reload design applications.

The methodology begins with the generation of a target reactivity objective, which is determined from the transformation of a target beginning-of-cycle power distribution through the Backward Diffusion Method. Combinations of available fuel assemblies and burnable absorber configurations are then established that meet the reactivity objective. This matching process seeks to minimize both the global (core-wide) and local (subgroups of assemblies) reactivity mismatch. Several hundred unique loading patterns are typically generated from reactivity matching.

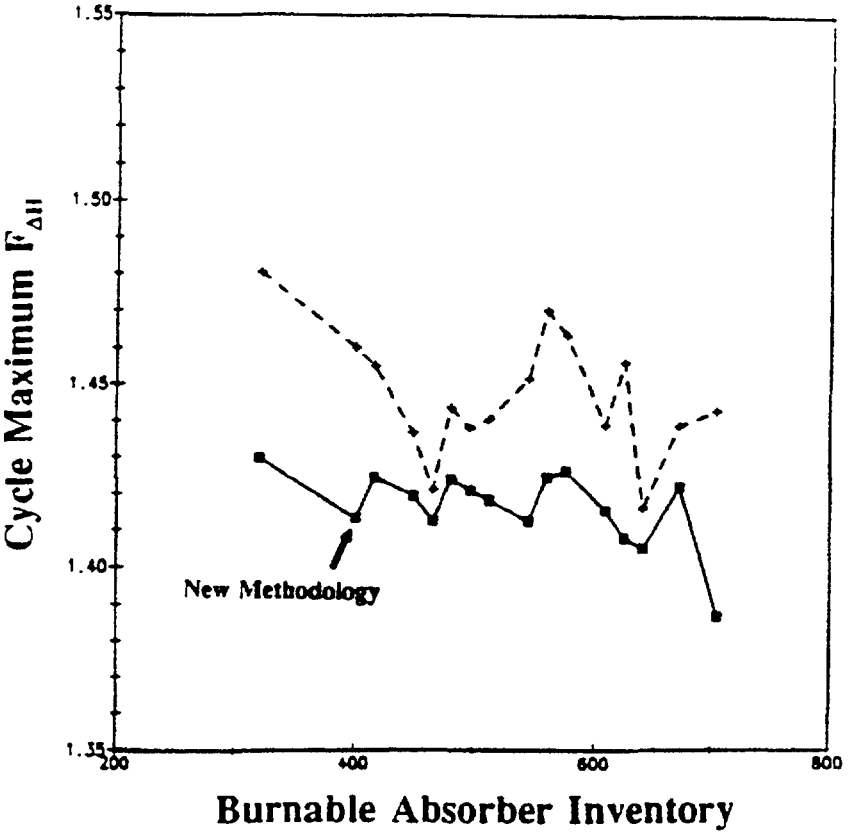

Figure 1 - ALPS Peaking Power Improvement
Westinghouse 193 Fuel Assembly Core

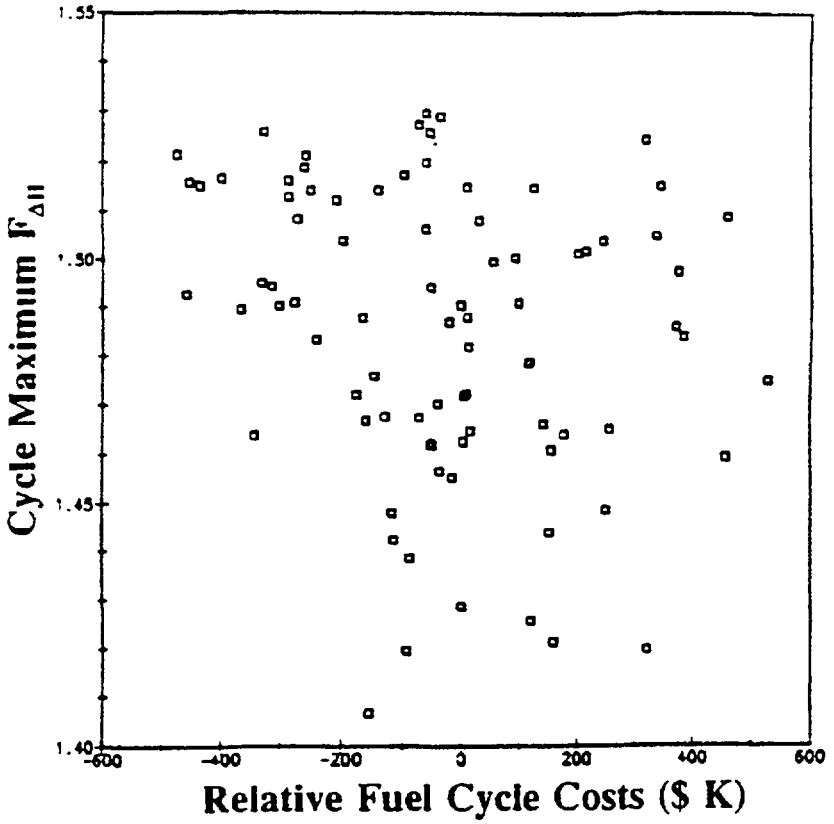

Figure 2 - Typical ALPS Loading Pattern Evaluation
Westinghouse 193 Fuel Assembly Core

An improved shuffling sequence has been added to ALPS with the objective function to further reduce the power peaks of the loading patterns generated by reactivity matching. Power peaks for these loading patterns through the cycle life are calculated using the accurate and fast-running steady-state mode of the two-dimensional SPNOVA [2] routine, which utilizes the core-wise Green's function matrix. Typically, 50,000 to 400,000 loading pattern candidates are generated and analyzed by ALPS depending on the core size. The 100 resulting loading patterns with lowest power peaks are then re-analyzed and depleted with the more detailed two-dimensional nodal method. These patterns are further evaluated with respect to moderator temperature coefficient (MTC), power peak ranking, number of burnable absorbers, cycle length, and relative fuel economy.

The strengths of the improved loading pattern search methodology in ALPS are shown in Figure 1. ALPS loading pattern search results are compared to LPOP results for a typical 193 assembly core, operating on an 18-month cycle, using low leakage fuel management and Wet Annular Burnable Absorbers (WABAs). For a given BA inventory, ALPS is generally able to find patterns with power peaks about 1% to 3% lower than LPOP. This translates into more optimum fuel management and also productivity improvement, as the designer spends less time manually fine-tuning the selected loading pattern.

The ALPS program incorporates a unique and innovative model permitting three-dimensional fuel features, such as axial blankets and reduced-length burnable absorbers, to be accurately represented with its extremely fast two-dimensional calculational routines. The model permits the designer to simulate 3D effects in designs ranging from the first transition cycle of axial blankets to the equilibrium cycle. Only three user input parameters are required to specify three-dimensional product features. Extensive benchmarking to the Westinghouse PHOENIX-P/ANC system for actual plant designs confirms the performance of this model.

ALPS has several unique features that make it more practical and that permit it to be easily used as a self-contained fuel management tool without the additional work of core model set up. It has its own master library of energy-group constants for all Westinghouse fuel assembly and burnable absorber (BA) types. The library has evolved to incorporate new Westinghouse features, such as Integral Fuel Burnable Absorbers (IFBAs), with higher B-10 loading and the use of IFBAs and discrete BAs within the same assembly. An added ALPS feature is the capability for the user to specify a loading pattern for dimensional analysis. Both full power and zero power MTC branch calculations, and branch calculations with control rods inserted can be performed at specified times during the cycle depletion. The program can also generate and read its own data file containing fuel inventory and core parameters. Coupled with its stand-alone capability, multiple cycle fuel management strategies and "what-if" scenarios can be efficiently and reliably examined.

The implementation of this software on a workstation computer system enabled the development of an advanced graphical user interface. This user interface provides additional productivity improvements due to enhanced man-machine interaction computing. These modifications enable the quick generation of loading patterns for the advanced fuel management strategies in operation today. This capability can help utility decision-making in optimizing margin-versus-economics tradeoffs best fit their specific operational considerations.

3. PHYSICS PERFORMANCE

The ALPS core modeling physics data and spatial depletion model have been extensively benchmarked with the Westinghouse PHOENIX-P/ANC system and plant measurements.

Table 1: ALPS Qualification Summary
Steady-State, Normal Operating Conditions
Comparison With PHOENIX-P/ANC Predictions

	ALPS Performance	Criteria
End-of-Cycle Reactivity		
Mean difference (ppm)	-0.6	
2σ (ppm)	±29.7	±30
Assembly Average Power (>1.0)	1.20% RMS	< 2% RMS
Assembly Peak Pin Power (>1.3)	1.99% RMS	< 3% RMS

Reliable and consistent performance with the design model are key to establishing useful enrichment estimates and cycle length estimates that meet requirements.

ALPS steady-state performance has been established by comparison with the detailed design model ANC for twenty-three actual plant cycles. The designs include all current Westinghouse fuel products features, such as IFBA. Twelve of the benchmarked cycles contained axial blanket fuel and several transition cycles were represented. The results summarized in Table 1 confirm that ALPS predictions agree well with ANC results. The key design parameters for fuel management calculations are end-of-cycle reactivity, assembly average powers, and assembly peak pin powers.

The statistics for all relevant parameters are consistently small and satisfy the performance criteria required by Westinghouse core designers. The statistics remain nearly the same if only cycles with axial blanket fuel are considered. These results indicate that the methodologies incorporated in ALPS, including the "3D effects" model, provide the required accuracy for use in fuel management applications.

4. DESIGN EXPERIENCE

The ALPS loading pattern search methodology makes the program capable of scanning through a large number of loading pattern candidates quickly with each one of them spatially analyzed and depleted. The methodology covers a wide spectrum of loading patterns and examines interesting and innovative patterns not typically considered by manual processes or by existing loading pattern search approaches, including Perturbation Theory and Expert Systems.

The ALPS program develops several loading pattern candidates that meet key design requirements, such as power peak, MTC, and cycle length, and provides a relative economic evaluation of each. As shown in Figure 2, the final result provided to the user contain a variety of evaluated loading patterns, rather than just one with no alternatives. The engineer can then select those patterns which meet the design constraints.

ALPS is used for nearly all fuel management studies and loading pattern development efforts performed at Westinghouse, and is also used by several utilities. Designers have found several benefits from use of the ALPS program. ALPS is able to identify more economical loading patterns than can be derived with manual loading pattern search techniques alone. There are several instances where ALPS found loading patterns which lowered fuel cycle costs by several hundred thousand dollars compared to manually derived loading patterns. The convenient and efficient multi-cycle capability in ALPS has also allowed designers to quickly and quantitatively determine the impact of current cycle fuel management strategy on future cycles. In all cases ALPS has improved productivity and reduced cycle time for fuel management work.

5. CONCLUSIONS

ALPS is a state-of-the-art fuel management tool used to search quickly, reliably, and accurately for loading patterns. The core modeling physics data and spatial depletion model developed and implemented at Westinghouse have been demonstrated to be sufficiently accurate to provide reliable enrichments presentations, cycle-length estimates, and power distribution predictions. The ALPS stand-alone capability allows the tool to be used independently of any design model to make the difficult task of in-core fuel management more efficient. An ALPS development program is in place focused on further improving ALPS performance and incorporating optional user-specified design constraints to the loading pattern search algorithm.

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IN-CORE FUEL MANAGEMENT OF THE SLOVAK POWER ENTERPRISE (SEP) VVER-440 REACTORS: PRESENT STATUS AND FUTURE CONCEPT

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Abstract

In the near future the nuclear fuel costs will be more than 45% of the total operating costs of the NPP Bohunice. From this point of view the in-core fuel management is a key point of our interest.
The Slowak Power Enterprise (SEP) have now more than 40 reactor-years experiences in reloading techniques for VVER 440. An old fuel design with a restricted number of enrichments allow to use only an out- in reload scheme or its slight variation. During the normal operation in NPP Bohunice there is ddischarged 1/3 of the core with an average burnup of 32 000 MWd/tU. It is SEP's strong interest to improve the present in-core fuel management.
This aim can be achieved only by using of an advanced fuel with new structural materials and optimized design solutions.

1. Introduction:

Within the ČSFR there are two power utilities: SEP covers the Slovak Republic and CEZ covers the Czech Republic. These two utilities have integrated transmission lines and a join dispatching center - CSED.

S E P:

The Slovak Power Enterprise (SEP) is the utility responsible for the operation of the power plants and and transmission of electricity within the Slovak Republic including four WWER - 440 nuclear PWRs (Pressurized Water Reactors) at Jaslovske Bohunice in operation and four WWER- 440 nuclear PWRs under construction at Mochovce.

The costs for nuclear fuel except the costs for the fuel cycle back-end are more than 26 % of the total operating costs of NPP Bohunice. We expect that this amount will increase to more than 45 % after including the costs for the fuel-cycle back - end. From this point of view the in core management is a key point of our interest.

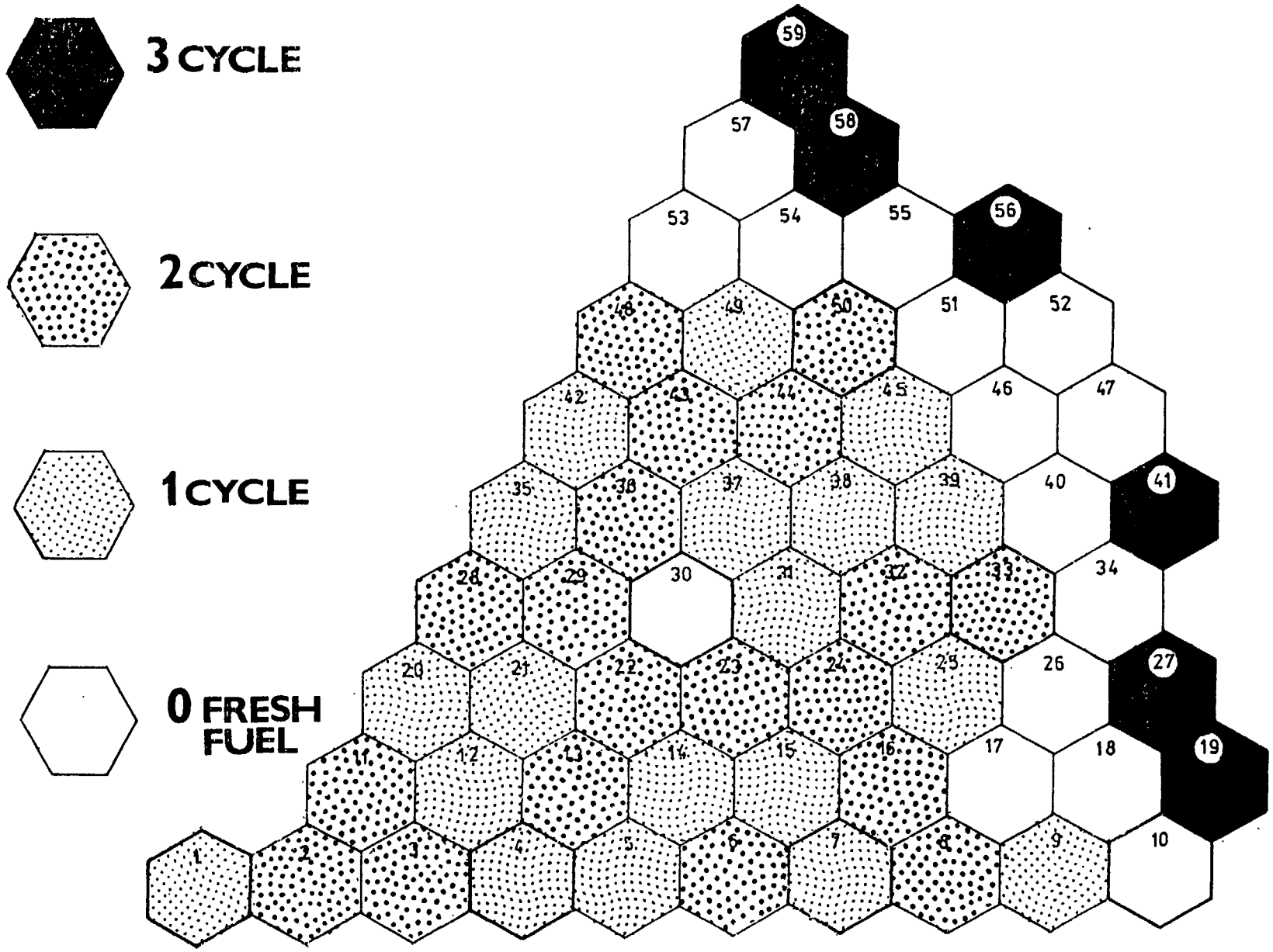

FIG. 1. Loading pattern of Bohunice 4, cycle No. 7.

2.Present status:

During the normal operation in NPP Bohunice there is 1/3 of fuel discharged from the core at the end of the cycle. This amount consists of 3.6% and 2.4% enrichment assemblies (average enrichment 3.35%). The design of reloading supposed to load the fresh fuel at the out of core. This design of core brought to SEP higher power rate at the periferial assemblies, high neutron leakage and extremly high fluence of neutrons to the reactor pressure vessel.

From the 1985 there is used low leakage reloads. This solution brought to SEP higher effectivness of the one way fuel cycle and decrease of the abovementioned disadvantages. In view of the fact that the present fuel design was not improved significantly in the last 15 years these low leakage reloads are relized in limited scale (see Fig 1.)

The assemblies are loaded in core during 3 years and reach average burnup 32000 MWd/tU.The lenght of one cycle is aproximately 290 full power days for all four units in NPP Jaslovské Bohunice. This solution was established with the aim to provide the annual outage for maintenance to the same period of calendary year,to avoid the double planed outages during winter.

3.Future concept - advanced reload patterns

Up till now there is only one supplier of fresh fuel for ČSFR it is The Commonwelth of Independent States (CIS). The fuel is relatively reliable,especialy for V-213 model but with low neutronic efficiency.

It is a SEP's strong interest to improve the present in

- core fuel management with aim to achieve:
  - safe, reliable, simple and flexible operation
  - lower fuel cycle cost

Table: 1

No. :	1	2	3	4
Part of core	1/3	1/4	1/5	1/3
No.of fresh assemblies	110	96	78	96
Burnup [MWd/ tU]	32 000	34 000	42 000	47 000
Aver. enrichment [%]	3.35	3.38	4.00	4.2
Cycle length [month]	12	12	12	18
EFPD [days]	290	290	290	290

These aims can be achieved by:

- use of proven structural materials for grids, instrumentation tube, axial blankets,integral fuel burnable absorbers, removable top nozzles
- use of proven design solutions as optimized design for improved thermal margin,spring and dimple fuel rod support, coil springs for plenum support etc.

We expect, that the new fuel design will have the features similar to the western fuel types. At the same time must be ensured full compatibility with the in-core instrumentation as well as with the present Russian fuel.

SEP in cooperation with other companies made studies to estimate the cost benefits of an improved fuel cycle. These studies include also the costs for the fuel cycle back-end. Table 1 shows the results of reload studies for VVER 440 reactors.

It is robably necessary starting from case 3 to use burnable absorbers. Till now there is no utility which operates VVER 440 unit or some vendor of fuel with the experiences in adopting of an other fuel design as it is offered by the Russian supplier. Several LWR fuel vendors are offering to develop a new VVER 440 fuel but we see the major difficulties in the following:

- development and qualification of new computer codes
- development of an hexagonal LWR fuel design
- compatibility with the present fuel in-core instrumentation and safety systems
- new plant safety analyses

To overcome these difficulties in adopting of new fuel it is necessary to observe the utmost diligence. It is recommended to adopt the following procedures :

1st step: the use of advanced fuel design for the first core in 2 unitss of NPP Mochovce (which is under construction)

2nd step: to implement new advanced reload techniques in these blocks

3rd step: start with the lead test assembly programs in one of the present operating units

4th step: to introduce new advanced reload technics for the unit mentioned in the 3rd step
block

4. Summary:

The original in-core management strategy of SEP's VVER reactors consisted of a loading scheme where the fresh fuel loaded at the core periphery. This loading scheme or its slight variation is used till now. Advanced in-out fuel management strategies as in western reactors can be used for VVER reactors only after introducing of a new fuel assembly design. SEP have to make any effort to achieve the level of western in - core fuel management.

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