MTR CORE LOADING PATTERN OPTIMIZATION USING BURNUP DEPENDENT GROUP CONSTANTS

by

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A diffusion theory based MTR fuel management methodology has been developed for finding superior core loading patterns at any stage for MTR systems, keeping track of burnup of individual fuel assemblies throughout their history. It is based on using burnup dependent group constants obtained by the WIMS-D/4 computer code for standard fuel elements and control fuel elements. This methodology has been implemented in a computer program named BFMTR, which carries out detailed five group diffusion theory calculations using the CITATION code as a subroutine. The core-wide spatial flux and power profiles thus obtained are used for calculating the peak-to-average power- and flux-ratios along with the available excess reactivity of the system. The fuel manager can use the BFMTR code for loading pattern optimization for maximizing the excess reactivity, keeping the peak-to-average power as well as flux-ratio within constraints. The results obtained by the BFMTR code have been found to be in good agreement with the corresponding experimental values for the equilibrium core of the Pakistan Research Reactor-1.

Key words: MTR, fuel burnup, core loading, core management, optimization, PARR-1

INTRODUCTION

Apart from core physics studies, nuclear research reactors find a wide variety of uses including production of radioisotopes for industrial as well as pharmaceutical applications, neutron activation analysis, boron capture neutron therapy (BNCT) and radiography, *etc*. Currently, there are 291 operational research reactors (including critical assemblies), 246 shut down reactors, 107 decommissioned reactors, 16 under construction and 15 new research reactors planned. These reactors are distributed among 58 countries. There is a fairly even regional distribution of the operational reactors with 64 of them in North America, 65 in Western Europe, 76 in Eastern Europe, 57 in Asia, 18 in Latin America, and 14 in Africa and the Middle East. Eighty-seven of the operational reactors are in 40 developing countries [1]. These systems were initially developed around 1947 at Idaho, USA, for conducting studies for bulk shielding properties of various materials. Their popularity over other types of research reactors stems from the fact that these systems have easy access to the reactor core and that they possess a larger number of irradiation facilities.

Due to the importance of both safety and operational points of views, efforts have been made in the past to carry out studies for the in-core fuel management for various test and research reactors worldwide. In this context, a nuclear analysis and the fuel management code system HANAFMS have been used for the analysis of the Korean high-flux advanced neutron application research reactor (HANARO) [2]. For benchmarking, separately the Monte Carlo based computer code MCNP4A has also been used for this purpose. The calculated results for criticality, power distribution, and control absorber rod value were found to be in good agreement with the corresponding values obtained experimentally. In the Netherlands, a

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thermal hydraulic model has been developed for the Hoger Onderwijs reactor (HOR), a 2 MW pool-type research reactor, and implemented in the computer code SHORT (steady-state HOR thermal-hydraulics) [3]. The theoretical predictions have been found to be in good agreement with the experimental measurements. While performing safety assessment for the FRJ-2, reactivity values were computed with the help of the depletion SUSAN computer code and verified by the ORIGEN computer code [4]. Discrepancies have been reported in the predicted and the corresponding experimental values and have been attributed to the inadequacy of the inverse kinetic method for covering the effects of flux deformation and interaction of the coarse control arm in the event of reactor scram.

Due to diverse uses of research reactor facilities, their varied operation schedules along with typically small sizes and demand for high excess reactivity for experiments and sample irradiation, the task of in-core fuel management becomes much more complex than that of a typical power plant. In the past, many techniques have been proposed for the optimization of the fuel reload patterns for such systems. For example, the multiple cycle interchange approach has been proposed for the HOR reactor system in the Netherlands [5]. In this approach, the first stage starts with the "elite" cluster containing the group of best patterns. Afterwards, the solution space is sampled in gradual manner towards the local patterns employing stochastic acceptance tests. Successive mixed-integer linear programming coupled with backward diffusion calculation has been employed for core loading pattern optimization for VVR-M type research reactors at Kiev, Ukraine [6]. More recently, a novel transition probability matrix based adaptive simulated annealing approach has been proposed for fuel reload pattern optimization which yields much faster convergence rates [7].

From the fuel management point of view, the main problem in the case of MTR systems is to attain high excess reactivity configuration in order to manage the loss of reactivity caused by the leakage and absorption of neutrons for the experiments and irradiation. A typical MTR system has around 25 standard fuel elements in addition to the collective five regulating and shim control elements. The total number of possible independent configurations is more than 10^9 which clearly rules out any enumeration technique. However, the task is somewhat simplified in the situation where the fuel manager has to shuffle or replace only a few fuel assemblies.

In this work, a methodology which enables the fuel manager to carry out search for the optimum loading pattern of a typical MTR system at any stage has been developed based on the use of burnup dependent group constants. The details of this methodology along with its computer implementation are presented here. A comparison of the results obtained by using this approach with the equilibrium configuration of the Pakistan research reactor-1 (PARR-1) is given in this paper.

TYPICAL MTR SYSTEM

In this work, the PARR-1 has been taken as a typical MTR system. Following the worldwide "Reduced enrichment for research and test reactor (RERTR)" Program, the PARR-1 has been upgraded from 5 MW HEU (over 90% enriched uranium) core to 10 MW LEU (~20% enriched uranium) system [8]. The upgraded core is composed of 24 plate-type standard fuel elements, five control fuel elements which are identical for both shim and regulating control elements, and four graphite reflectors. It is essentially a demineralized-water cooled, moderated and reflected system with the reactor core sitting at 7 meter water depth in a two sectioned pool. A downward flow rate of 950 m^3/s is required at full power to remove the heat generated by the nuclear fission chain reaction. The design specifications of PARR-1 are briefly listed in tab. 1 [9].

Table 1. Some of the design parameters of the upgraded LEU PARR-1 core [9]

Parameter	Value
Reactor type	Pool, MTR
Steady-state power level [MW]	9
Fuel (enrichment, %)	U_3Si_2-Al (19.99)
Grid plate	6 9
Core geometry	6 5
Lattice pitch [mm]	81.00 77.11
Fuel element dimensions [mm]	79.63 75.92
Fuel plates in core (a) standard fuel elements (SFE) (b) control fuel elements (CFE)	 24 23 5 13
Thickness of plates [mm] inner, outer	1.27, 1.50
Total width of plate [mm]	66.92
Total length of plate [mm]	625.0
Fuel meat dimensions [mm]	62.75 0.51 600
Thickness of clad [mm] inner, outer	0.38, 0.495
Thickness of side plates [mm]	4.5
Water channel thickness [mm]	2.10
Coolant flow rate [m^3/h]	900
Coolant inlet temperature [°C]	38
Coolant inlet pressure [bar*]	1.703

*1 bar = 100 kPa

The experimental facilities around the reactor core consist of six radial beam tubes, one tangential through-tube, graphite thermal column, independent rabbit station, a hot cell, dry gamma irradiation cell and bulk irradiation area. A special feature of the up-

graded PARR-1 core is the central flux trap facility where high neutron flux is available for the irradiation purpose. The core configuration of the PARR-1 system is given in fig. 1 [10].

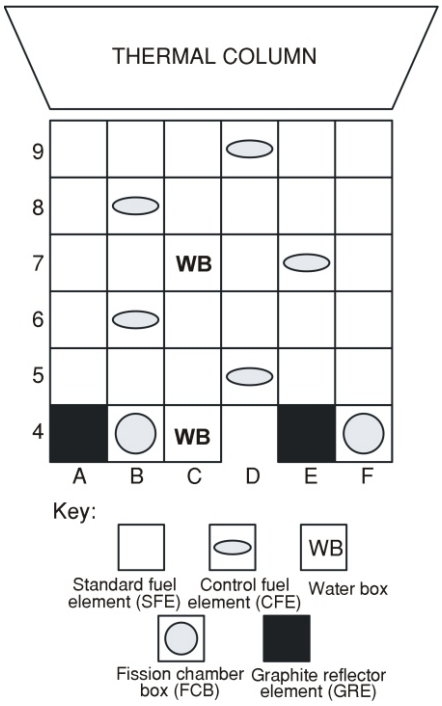

Figure 1. Schematics of the upgrades LEU PARR-1 core

BURNUP DEPENDENT GROUP CONSTANT GENERATION

The main complexity of the MTR fuel management arises from the fact that the core size being small leads to steep flux gradients, which results in large disparity in the burnup of even closely placed fuel and control elements. Frequent scheduled shuffling of fuel elements aggravates this problem. One needs to keep track on the burnup history of each fuel and control element in order to optimize the loading pattern. The existing procedure requires core composition calculations at each stage which is a complicated processing, keeping in view the frequent fuel shuffling as well as steep flux gradients. The proposed methodology is based on the generation of burnup dependent group constants and using them in detailed 2-D-diffusion theory calculations. For the group constant generation, the WIMS-D/4 computer code has been used in this work [11]. The WIMS-D/4 is a general purpose lattice code for the generation of cell averaged multigroup constants using transport theory calculations. For this purpose, it uses a 69-group nuclear data library of UK origin which has 14-fast, 13-resonance, and 42-thermal groups. The fast groups have equal lethargy spacing. In the resonance range, the lower region has specifically been chosen to match important resonances. The upper boundary of thermal range has been arranged in such a way that thermal up-scattering can be neglected.

The super-cell used for standard fuel element (SFE) is shown in fig. 2 [10]. The extra-region takes into account the structural material (aluminum) and the water-gap between two fuel elements. The resulting group constants have been generated for a wide enough range of burnup values to match the history of all fuel elements. A fourth order polynomial has been fitted to each of the group constant so that the maximum difference between the fitted and the actual values remains below 1% error-limit throughout the burnup range.

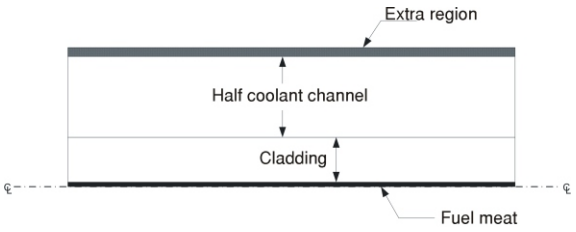

Figure 2. Cross-sectional view of the super cell used for generation of the multigroup constant for standard fuel element (SFE)

EIGENVALUE CALCULATIONS

For the determination of effective multiplication factor and excess reactivity available in the system, the multigroup diffusion theory based CITATION computer code has been used as a subroutine in this work [12]. For detailed spatial calculations, a 2-D fine-mesh model of the PARR-1 core and reflector has been used. Five energy groups have been employed for the k_{eff} calculations. The group constants for each fuel and control element have been generated by using the fitted polynomials along with the current accumulated burnup of each element in the BFMTR code. The fuel manager may choose the desired loading configuration and carry out the excess reactivity optimization search, keeping the peak-to-average flux as well as power ratios within the prescribed limits.

SIMULATION RESULTS

The diffusion theory based MTR fuel management methodology employing burnup dependent group constants developed in this work and implemented in the computer code BFMTR has been tested for the equilibrium core configuration of the PARR-1 system as a benchmark. For this purpose, a five group of 25-independent standard fuel elements (SFE), 5-in-

dependent control fuel elements (CFE) along with graphite elements, water boxes based model of PARR-1 core has been used in the BFMTR code. The initial grid of BFMTR has 5 6 loading matrix where a user can specify the position of various core elements. The BFMTR code uses a history data for burnup of various standard fuel as well as control fuel elements. The user can choose among some predefined initial configurations, including the fresh clean startup core of the upgraded LEU PARR-1 core as well as the equilibrium core configurations. Also, any desired core loading pattern may be specified along with prescribed values of burnup for SFE and CFE. The PARR-1 equilibrium core configuration is taken as the reference for the simulations and is shown in fig. 1.

For the generation of burnup dependant group constants, the super-cell used in the WIMSD calculations for SFE is shown in fig. 2. The resulting data have been processed and polynomial fittings have been saved in a separate library. The BFMTR code uses the fitted polynomials for computing the necessary group constants for SFE and CFE, while the data for water boxes, GRE, *etc*., are taken directly from the library file maintained for this purpose. The resulting data set along with core configuration and geometry details have been used for eigenvalue calculations in the CITATION subroutine.

The assembly-wise power profile computed by the BFMTR code has been compared with the corresponding experimental values in fig. 3. It is clear from these data that the predicted values are generally in good agreement with the corresponding experimental values. Deviations are seen in places where steep gradients exist and these can be attributed to the inadequacy of diffusion theory in such regions. It is expected that detailed transport theory results would be better for such regions. A similar trend is seen in the case of core-wide thermal flux profile as shown in fig. 4. Again, in regions with moderate flux gradients, theoretical values agree well with the corresponding experimental values, whereas in regions with steep flux gradients discrepancies are seen and are expected to be resolved by detailed transport theory analysis. The core-wide power profile predicted by the BFMTR code is shown in fig. 5. The regions near the central flux trap are the main contributors towards total power generated by the core.

CONCLUSIONS

A burnup-dependant group constant based in-core fuel management methodology for the MTR type systems has been developed and implemented in the BFMTR computer code. It incorporates burnup history data for each fuel element in the core, and as a result, fuel shuffling and reload pattern optimization can be carried out at various stages in a standard cycle.

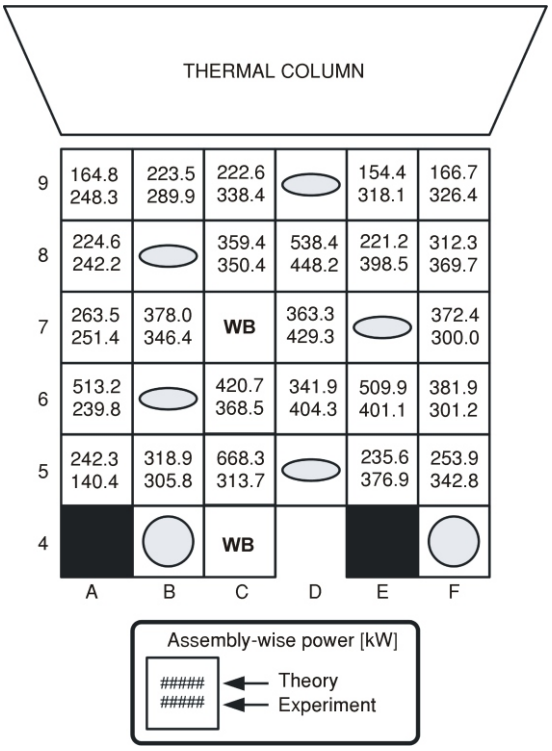

Figure 3. Comparison of the calculated and experimentally measured assembly power profile [10] for the PARR-I equilibrium core

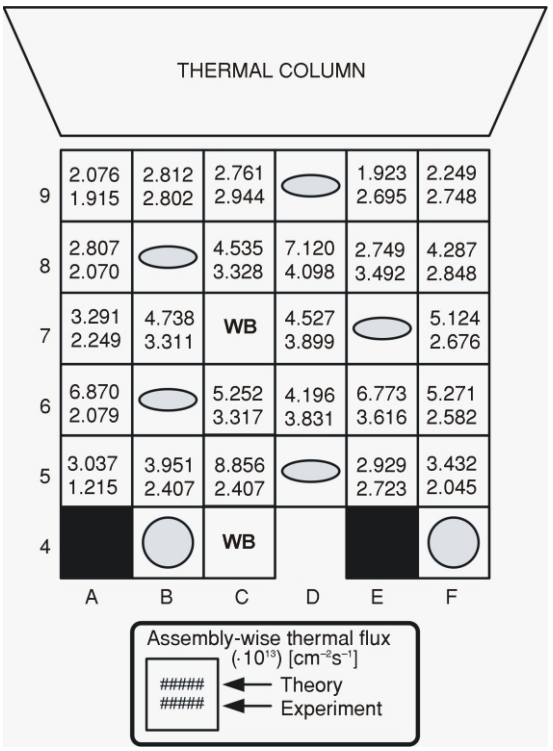

Figure 4. Comparison of the calculated and experimentally measured thermal neutron flux profile [10] for the PARR-1 equilibrium core

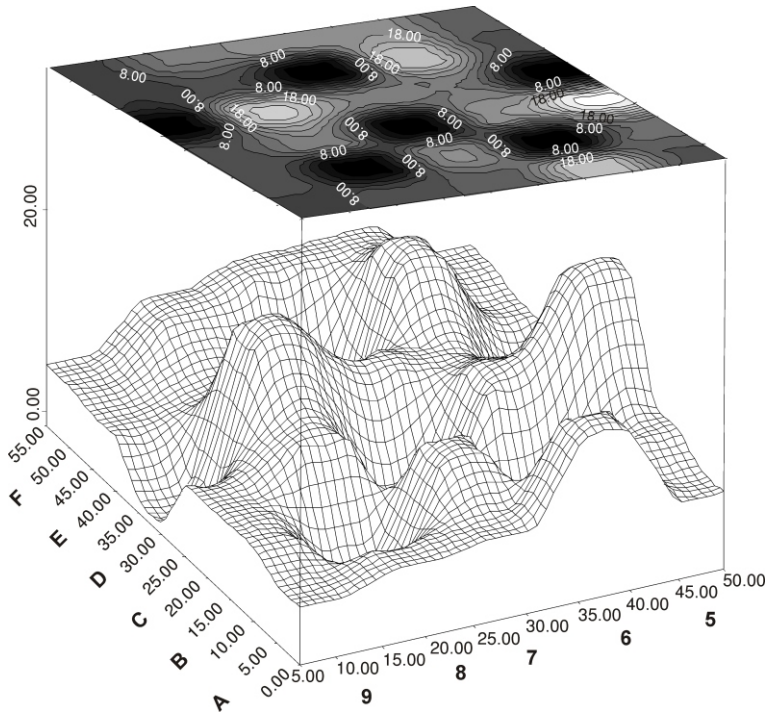

Figure 5. Graphical view of the calculated power profile as the surface mesh and as contours for the PARR-1 equilibrium core

For the burnup-dependant group constant generation, the WIMS-D/4 computer code has been employed. The CITATION code is used as a subroutine in the BFMTR package for calculating eigenvalue, flux and power profile throughout the core. The target objective functions include the maximization of excess reactivity, maximization of element burnup, limiting power peaking factor, *etc*. The BFMTR code is flexible and allows a user to specify a given starting re-load pattern along with the optimization criterion. The predicted values of flux and power profiles have compared well with the corresponding experimental values in the regions with moderate gradients.

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ОПТИМИЗОВАЊЕ ПУЊЕЊА ЈЕЗГРА МТР РЕАКТОРА УПОТРЕБОМ ГРУПНИХ КОНСТАНТИ ЗАВИСНИХ ОД ИЗГАРАЊА

Развијена је методологија управљања горивом у МТР реакторима заснована на дифузионој теорији која утврђује боље начине пуњења језгра у ма ком стању МТР система, пратећи изгарање појединачних горивних склопова током њихове историје. Заснива се на употреби групних константи зависних од изгарања, добијених WIMS-D/4 рачунарским програмом за стандардне горивне и контролне елементе. Ова методологија примењена је у рачунарском програму BFMTR, којим се обавља детаљни петогрупни дифузиони прорачун уз коришћење CITATION кода као потпрограма. Тако добијене просторне расподеле флукса и снаге у језгру коришћене су за израчунавање односа максималне и средње снаге и односа флуксева, као и вишка реактивности система. Оператор горивом може користити BFMTR код да оптимизује начин пуњења језгра ради постизања максималног вишка реактивности, задржавајући односе максималне и средње снаге као и односе флуксева унутар прописаних граница. Показало се да су резултати добијени BFMTR кодом у доброј сагласности са одговарајућим експерименталним вредностима равнотежног језгра Првог пакистанског истраживачког реактора.

Кључне речи: МТР, изгарање горива, пуњење горива, управљање језгром, оптимизација, PARR-1