SYNOPSIS

OPTIMIZATION STUDIES FOR PHYSICS PROBLEMS IN INDIAN PHWRs

The thesis presents an account of original contribution of the author towards the optimization studies for some physics problems in Indian Pressurised Heavy Water Reactors (PHWRs).

The concept of Nuclear Reactors has its origin in the discovery of nuclear fission in 1939. In a nuclear fission, a neutron is absorbed in a heavy nucleus such as $^{235}$U and two or more fragments are produced. This reaction has two interesting features. First is that a significant amount of energy (about 200 MeV) is produced which is in the form of kinetic energy of fission fragments. Secondly, a few (on the average 2 to 3) neutrons are also produced. These facts immediately suggested the possibility of utilizing the emergent neutrons to cause further fissions in other heavy nuclei and thus to have a self-sustained steady fission chain reaction. Such a system, called a nuclear reactor, could then act as a steady source of energy. Since the first reactor built by Fermi in 1942, the field has continuously evolved leading to the very many complex nuclear reactors of today.

There have been various ways in which reactors are classified. The research reactors operate at low powers, the primary aim being research, isotope production etc. Power Reactors, on the contrary, are designed to act as a source of energy. On the basis of neutron energy, two major types are Thermal and Fast reactors. The Thermal Reactors use a moderator material such as Light Water, Heavy Water or Graphite. Neutrons are slowed down by the moderator to make use of the high value of fission cross-section at low neutron energy. Light Water Reactors (LWR) need enrichment of Uranium to about 3 to 5 percent. They come in two major categories: Pressurised Water Reactors (PWR) and Boiling Water Reactors (BWR). LWRs form a dominant source of Nuclear Energy in the world.

Another popular type of Thermal reactors, which is a subject of the present thesis, is the Pressurised Heavy Water Reactor (PHWR). These reactor designs originated in Canada and are often referred to as CANDU (CANada Deuterium Uranium) Reactor. The use of heavy water moderator is the key to the PHWR system, enabling them to use Natural uranium (NU) as fuel. The PHWR can be operated without expensive uranium enrichment facilities. The relatively lower temperature and high density moderator leads to sufficiently thermalised neutrons and hence a better fuel utilization. There are
fifteen small sized 220 MWe PHWRs and two medium sized 540 MWe PHWRs operating in India. The more advanced 700 MWe PHWR is in design stage and would start operation in near future [1]. The PHWRs constitute the first stage of the “Three-stage Indian Nuclear Power Programme”. These reactors can produce the Plutonium needed in the Second stage for the construction of Fast Breeder Reactors. The Third stage involves utilisation of Thorium to further multiply the power generation capability.

The subject of optimization has permeated the field of nuclear reactors at all stages. Indeed, the choice of most suitable nuclear energy programme depending on the needs and resources of a country is an optimization problem. The design of any specific reactor type would also involve optimization while designing details of fuel lattice, fuel assemblies, full core parameters, thermal hydraulics, shielding, control system and fuel management. In the present thesis, a small subset of these optimization problems, relevant to Indian PHWRs has been studied.

In general, there are two factors which are crucial to optimization problems in Nuclear Power industry: Safety and Economy. For safety, the power distribution in a reactor should be such that the heat is removed safely. Moreover, in case of emergency, it should be always possible to shut down the reactor safely. From the economy point of view, it is always desirable to draw maximum power from the reactor and core parameters (like bulk power, zone power etc.) used in reactor regulation are estimated accurately. Moreover, it is desirable to maximise fuel utilization. Often, all these requirements conflict with each other and what is needed is Optimization.

Specific problems studied in the thesis

The problems studied can be broadly classified in to three types A, B and C.

A. Loading Pattern Optimization

A good part of the studies in the present thesis are concerned with optimization of initial fuel loading in PHWRs, which is a part of the overall Fuel Management over full life time of a reactor. The optimization of initial fuel loading has to be such that maximum economy is achieved without violating any safety constraints. In particular, for economy, it is desirable to operate the reactor at full power right from beginning. This is not possible if the full core is loaded with Natural Uranium (NU) because power peaking would be high. After the reactor has operated for an year or so, power is flattened throughout the life of reactor by having two or three burn-up zones which enables full power operation. In a fresh core, however, the burn-up zones would not be there leading to relatively higher peaking. Hence, it is necessary to load few Thorium or Depleted Uranium (DU) bundles at selected locations in the core. Out
of the total 3672 bundle locations in 220 MWe PHWR, a few tens of locations have to be identified for Thorium/DU bundles. DU bundle contains about 0.3wt% to 0.6wt% $^{235}$U, which is lower as compared to NU (0.7115wt%). Choice of location of Thorium/DU should be such that full power can be drawn, $K$-effective is large as far as possible and safety constraints related to thermal hydraulic and control are satisfied. This is a very large constrained combinatorial optimization problem. The number of possible arrangements is so huge (say $10^{50}$) that it is impossible to try them all. This problem is addressed in the present thesis by the use of evolutionary algorithms. Optimization methods based on the well-known Genetic Algorithm (GA) and a very recent algorithm called Estimation of Distribution Algorithm (EDA) were tried. The EDA was found to work better than GA in the sense it gives better results with lesser computational effort. Hence, further studies like determining fresh core loading pattern for 220 MWe and 700 MWe PHWR with DU bundles were carried out using EDA.

There are several variations of the above problem that are needed in practice:

1) The required amount of flattening in flux can be achieved by loading either Thorium or DU bundles. The number of DU bundles to be loaded is usually larger (say a few hundreds) as compared to Thorium bundles.

2) It may be desirable to load DU bundles in such a way that they are removed at the first refueling of that channel.

3) For convenience in loading, it may be desirable to load DU bundles in only specific axial locations, say 7th location out of the 12 locations.

4) In case of the large sized 700 MWe PHWR, many new considerations arise. For instance, one has to satisfy two stuck rod criterion instead of the one stuck rod criterion in the 220 MWe PHWR. The 700 MWe core has more symmetry properties providing an opportunity to reduce problem size.

All the above variations of fresh PHWR loading have been analysed in the present thesis.

B. TPMS Optimization

A completely different optimization problem concerned with the Thermal Power Monitoring System (TPMS) in the 700 MWe PHWR is considered. These reactors are equipped with 44 instrumented channels out of the total 392 channels. The instrumented channels are used to estimate reactor bulk power and zone powers. Choosing the 44 channels from amongst 392 channels so as to maximize the power prediction accuracy is a combinatorial optimization problem. There are several constraints to be followed. From the similarity of this problem with the fresh core loading pattern
generation problem discussed in Section A, slightly modified EDA is applied. A more accurate TPMS design was obtained.

C. FMS Optimization

An on-line Flux Mapping System (FMS) is used in 700 MWe PHWRs. There are 102 in-core neutron detectors. The purpose of FMS is to generate a detailed flux map at all points in reactor using the 102 detector readings. It is usually based on use of higher K-modes and certain shape functions as basis functions. There is scope for optimization in terms of choice basis functions and computational method. This optimization problem is completely different in nature from the extremely large sized problems in sections A and B. It is solved by judicious trial and error techniques.

Layout of the thesis

The overall work is presented in nine chapters as follows:

Chapter 1 presents a brief introduction to the topic. The Pressurized Heavy Water Reactors (PHWRs) operating in India and their evolution is described. An outline of the research work presented in the thesis is also given.

Chapter 2 describes the design detail of Indian PHWRs. The 220 MWe PHWR consists of 306 horizontal fuel channels containing coolant and fuel bundles whereas, 540 and 700 MWe PHWR consist of 392 fuel channels. For the purpose of reactor regulation in 220 MWe PHWR, there are 4 Adjuster Rods (ARs), 2 Regulating Rods (RRs) and 2 Shim Rods (2SRs). The prime difference between 220 MWe and 540/700 MWe PHWRs is the neutronic coupling. The 540 MWe reactor, being large sized, is loosely coupled and hence is prone to spatial oscillations during its operation. For the purpose of reactor regulation, the core is logically divided into 14 zones. The zone powers also need to be monitored along with the total reactor power. There are 14 Zone Control Compartments (ZCC), 17 Adjuster Rods (ARs) symmetrically grouped into eight banks and 4 Control Rods (CRs). The zone powers are measured using zone control detectors (i.e. SPNDs). The zone powers measured by these small sized SPNDs need to be corrected by zone powers estimated by some other more accurate means like Thermal Power Measurement System (TPMS) or Flux Mapping System (FMS). The physics optimization problems related to PHWRs are described.

Chapter 3 describes the steady-state neutronic core simulation method for PHWR. The purpose is to find power distribution and effective multiplication factor ($K_{eff}$) for any given configuration of the
reactor core. The solution of optimization problem requires repeated use of this capability. Neutron transport equation is described by considering the rates at which neutrons of different energies moving in different directions enter and leave a small phase space element [3]. The equation consists of nuclear cross-sections (i.e. absorption, scattering and fission) and neutron flux distribution. The cross sections are the quantities which define the particle interaction probabilities and it is measured in barns ($10^{-24}$ cm$^2$). The cross sections of each material are highly dependent on the energy of the incident neutron [2].

At reasonably low energies of incident neutrons, cross sections are quite smooth in energy. However, as the energy increase, the cross sections are dominated by resonance peaks that result from unstable state of compound nucleus formed by the collision. In the off-resonance region, the variation in cross section shows $1/E^{1/2}$ or $1/v$ dependence. The relative geometric arrangement of fuel bundles, pressure tubes, coolant materials, calandria tubes, moderating material follows regular patterns inside the core. These patterns are referred to as lattices. Homogenous lattice cross-sections are obtained by solving neutron transport equation using the 69-energy group nuclear cross section data library in WIMSD conventions. For this purpose 2-D code ‘CLUB’ [9] that is based on collision probability method is used. The incremental cross-sections corresponding to various reactivity devices present in the core are obtained by solving 3-D neutron transport code ‘BOXER’ [10]. Full reactor core simulation is carried out by solving neutron diffusion equation applied to the full core. Diffusion theory is sufficiently accurate [4] to provide a quantitative understanding of many physics features of nuclear reactors and is, in fact, the workhorse computational method of nuclear reactor physics. A computer code ‘DOLP’ has been developed for full core simulation. It solves neutron diffusion equation by finite difference method. The finite differenced multi-group neutron diffusion equation can be written [5] as

$$M\phi = \frac{1}{K}F\phi$$

The matrix $M$ represents absorption, leakage and group to group transfer. The matrix $F$ represents fission neutron generation. There are many eigen solutions corresponding to different values of $K$. The $K$ values are real and positive. The largest one is called effective multiplication factor ($K_{\text{eff}}$). It corresponds to fundamental mode. This mode is found by power iterations (or outer iterations). Inner iterations are used to solve within group source problem. The outer iterations can be accelerated using Chebyshev method [7]. There exists another quite efficient approach based on Orthomin(1) algorithm [8] to solve the eigenvalue problem $M\phi = \lambda F\phi$. The residual vector $r = M\phi - \lambda F\phi$ is determined. The function $f(\phi) = (r, r)$ is minimized in Orthomin(1) algorithm. All the three options are provided in
DOLP. A comparison of CPU time is made between power iteration, Chebyshev method and Orthomin(1) algorithm.

Chapter 4 outlines the basic ideas in optimization methods. Objective function, decision variables and constraints are three main components of an optimization problem. A typical optimization problem can be stated as follows:

To find \( x = \begin{bmatrix} x_1 \\ x_2 \\ \vdots \\ x_n \end{bmatrix} \), which minimizes or maximizes \( f(x) \);

Subject to the constraints

\[
g_i(x) \leq 0; \quad i = 1,2,3 \ldots m
\]

\[
h_j(x) = 0; \quad j = 1,2,3 \ldots p
\]

where \( x \) is an \( n \)-dimensional vector called design variable, \( f(x) \) is called the objective function, and \( g_i(x) \) and \( h_j(x) \) are known as inequality and equality constraints respectively. This type of problem is called constrained optimization problem. Optimization problems can be classified based on the type of constraints, nature of design variables, nature of the equations involved and type & number of objective functions. As per the classification suitable solution method is applied to solve the optimization problem. The classical optimization techniques are useful in finding the optimum solution or unconstrained maxima or minima of continuous and differentiable functions. These methods assume that the function is differentiable twice with respect to the design variables and that the derivatives are continuous. For problems with equality constraints the Lagrange multiplier method can be used. Graphical optimization and method of steepest descent can also be used to solve the problem.

Most of the real world optimization problems involve complexities like discrete, continuous or mixed variables, multiple conflicting objectives, non-linearity, discontinuity etc. The search space may be so large that the global optimum can not be found in reasonable time. The classical methods may not be efficient to solve such problems. Various stochastic methods like hill climbing [26], simulated annealing [13] or evolutionary algorithm [ 12, 15, 18, 25] can be used in such situations. Some methods start with a single guess solution and improve it gradually. Evolutionary Algorithm (EA) is a population based search procedure that incorporates random (stochastic) variation and selection. It is developed to arrive
Chapter 5 describes the optimization of Thorium/DU loading in fresh core of 220 MWe PHWRs. A suitable fresh core loading pattern to extract full power from the reactor core is required for a new reactor as well as for a reactor which has undergone EMCCR (EnMass Coolant Channel Replacement). Few Thorium or Depleted Uranium (DU) bundles and rest Natural Uranium (NU) bundles are used in the fresh core to obtain required amount of flattening in the flux. The problem of choosing locations of Thorium/DU bundles in the initial core to achieve nearly full power is considered. This is a fairly complex combinatorial problem with many conflicting requirements. One has to obtain more than 95% full power, maximum possible reactivity, permitted bundle and channel powers and sufficient shutdown system worth. Two evolutionary algorithms Genetic Algorithm and Estimation of Distribution Algorithm [24] are used. In our approach, it is necessary to perform a very large number (~10^5) of neutron diffusion calculations. This has been possible due to the use of parallel super computer system ANUPAM at BARC and CRL at Pune. The two models using X-symmetry (half core) and X&Z-symmetry (one-fourth core) are elaborated. The random correction approach and penalty approach were applied [29] to the cheaper XZ-model. It was observed that the random correction approach applied on XZ-model is more economic. Several patterns have been generated that contain a range (from 12 to 44) of Thorium bundles and rest NU bundles. A loading pattern consisting of 92 DU (0.3wt% U235) and remaining NU fuel bundles generated using the algorithm has been loaded on December 2010 at KAPS#1, Kakrapar, Gujarat after EMCCR (EnMass Coolant Channel Replacement).

Chapter 6 describes the optimization of DU loading in fresh core 700 MWe PHWRs. Estimation of Distribution Algorithm is used to find optimum DU loading. The core lay-out, control devices and safety parameters of 700 MWe PHWR differ widely from those of 220 MWe. A requirement imposed in the present study was that the DU bundles should lie only at certain axial location to make loading
operation easy. Four types of optimization studies (case A, case B, case C and case D) are carried out in which DU bundles are placed at only one axial location and that is 7th, 8th, 9th or 10th location respectively. The optimum fresh core configuration was obtained [30] in each of the 4 cases. It was found that three cases satisfy all the safety constraints and give 100 % FP. Later on, burn-up of the optimum configurations was simulated for about 115 FPD during which no refueling is needed. On the basis of behavior of maximum CPPF in this period and better operating margin for regional over power (ROP) system, optimum configuration of DU in case A (i.e. axially at 7th location) is found to be most suitable. Finally, to satisfy the stuck rod criterion, additional DU bundles were loaded at axial ends. It may be mentioned that the stuck rod criterion cannot be introduced in the form of penalty function because the maximum worth rods can vary with the various DU configurations tried by the EDA algorithm.

Chapter 7 describes optimization of the design of Thermal Power Monitoring System (TPMS) for the forthcoming 700 MWe PHWR. The problem of choosing 44 fuel channels for instrumentation (out of 392 fuel channels) which can predict reactor bulk power and zone powers (on per unit basis) accurately is a combinatorial constraint optimization problem. The problem has been solved using Estimation of Distribution Algorithm and the constraints were handled in a different way. The individuals are generated (during initial and subsequent generations) in such a way that the constraints are satisfied. 44 fuel channels are selected for instrumentation. The selected fuel channels are satisfying all the imposed constraints. The error by TPMS in estimating reactor bulk power and zone powers are determined for reactor core having various reactivity device configurations to ensure that the movement of reactivity devices does not affect the TPMS results. The per unit basis bulk power estimated by instrumented channels lies between 0.995 and 1.005. Thus error in bulk power is less than ±0.5%. The %error in estimating reactor zone power lies between -2% and 2%.

Chapter 8 describes the optimization of the Flux Mapping method in Online Flux Mapping System (FMS) for 700 MWe PHWR. The choice of computational method for FMS is a sensitive issue since it is expected to be versatile, accurate and fast. Three computational methods namely Flux Synthesis (FS) method [33, 34], Internal Boundary Condition (IBC) method [41] and Combined Least Square (CLSQ) method [40, 42] were numerically tested to carry out flux mapping for some representative cases that can occur frequently in a 700 MWe PHWR. The FS method needs very little computation but has poor accuracy. The main reason for the inaccuracy is that the precise knowledge of prevailing snap-shot configuration such as burn-up distribution, AR positions and ZCCs water levels is
not used in the calculations. The IBC method also has errors comparable with FS method. On the other hand, the CLSQ method, which is based on sound principles, gives good accuracy in the proposed application but needs too much computational effort. In particular, the CGNR (Conjugate Gradient Normal Residual) [43] calculations need maximum effort. In view of this, a new hybrid method called modified FS (MFS) method [44] based on combination of FS and CLSQ methods was tried. It is found to give good accuracy without too much computational effort. It makes use of precise knowledge of prevailing state for fundamental K-mode calculation. In addition, it makes use of pre-computed higher K-modes of reference state for minimizing error at detector locations; a procedure which requires solving only a few tens of linear equations. The most difficult CGNR calculations are avoided altogether. The MFS method was further studied for cases with detector failures. It seems better to avoid faulty detectors in the calculation. It may be possible to identify faulty detectors by comparison of detector fluxes with flux predicted by standard mesh K-calculation for the snap-shot configuration.

Chapter 9 summarizes overall work and discusses possible extensions of the work.