DEVELOPMENT OF UTILITY SYSTEM SIMULATION MODEL

MIT DSR Project 72107

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DEVELOPMENT OF UTILITY SYSTEM SIMULATION MODEL

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1.0 Summary

The first year of work at M.I.T. under sponsorship of Commonwealth Edison Company on the development of techniques and information for utility systems planning has been varied and productive.

An analytic study of incremental costing for nuclear power plants has been completed with results useful in planning the utilization of nuclear powered capacity in the short-range and mid-range (1). Two shorter projects have been completed through which rapid techniques for in-core nuclear fuel management were improved and evaluated (2, 3). These studies have increased M.I.T.'s capabilities to carry out a variety of sensitivity analyses concerning the effects of nuclear fuel design and operating variables on energy costs.

Two sensitivity studies are nearing completion. One project (4) has been concerned with evaluating the reactivity and economic effects of refueling a nuclear reactor both earlier and later than originally planned but with the planned amount and enrichment of fuel. The effects of resulting changes in both time and fuel burnup were considered. The second project (5) has studied the effects of fuel stretch-out on spent fuel composition and on power distribution in the core and control requirements following refueling. The economic effects of fuel stretch-out, based on the physics and engineering results obtained, will be evaluated in a following study.

In addition to these studies already completed or nearing completion, the major effort has been devoted to the main task of developing methods for planning the economic operation of utility systems which employ a mixture of conventional, hydro, and nuclear power generating equipment. Much time has been spent in acquiring skills believed necessary for power systems modeling. The M.I.T. team has acquired improved background in the following areas: utility systems operation and planning,

optimization and simulation techniques, applied probability theory, and methods of cost and economic analysis. Several of these techniques were applied in the Spring of 1970 to developing a plan for the operation of the Commonwealth system including Dresden Units 1 and 2 during the periods of projected peak demand in the Summer of 1970.

In the area of systems planning, two complementary projects have been initiated. The objective of the first (6) is to develop a model for the optimum scheduling of conventional and nuclear generating units using a marginal cost philosophy. The objectives of this work are similar to the analytic study (1) mentioned above but are aimed at treating the non-steady-state cases that arise in an operating system so that system modeling will therefore be required. System constraints and information concerning maintenance and refueling schedules will be considered as inputs.

The second general project area (7) is aimed at the development of a nuclear simulation model generally useful (a) in supplying information on nuclear fuel composition, energy release, and costs to the systems model referred to above and (b) in carrying out a variety of sensitivity analyses aimed at better defining the bounds of feasible and economic alternatives that should be considered in systems model evaluations.

While these two projects are related and each useful to the other, they are sufficiently decoupled so that the results of either can be used independently of the other.

From an educational viewpoint the project has been very successful in providing challenging thesis topics and in motivating students. One doctoral (1) and two masters (2, 3) have been completed, two more master's theses (4, 5) and two more doctoral theses (6, 7) are underway. Two presentations of the work have been made at a society meeting (8,9). Two of the three graduates have already joined utility companies upon graduation. Of the seven graduate students who have participated so far, it has only been necessary to provide full financial support for one student and to provide limited assistance for report preparation to another.

2.0 Analytic Study of Marginal Costs

2.1 Introduction

This study represented an analytical approach to the optimization of use of nuclear power. A theoretical background was developed which may be useful for solving practical problems of utilization and may provide guidelines for model building. In particular, the derivation of marginal costs of nuclear energy may clarify a fundamental issue.

Concepts were developed for optimization of the utilization of a given nuclear power plant in the generating system of an electric utility company. Two optimization levels were considered: (1) mid-range planning, where the nuclear power plant under consideration is fixed in terms of type, capacity, location, etc. but is not fueled and the fuel enrichment and the duration of any future interval are optimization variables; (2) short-range planning, where also the fuel enrichment and the fueling schedule are given and the utilization of a given fuel load during the current fueling interval is the optimization variable.

2.2 Mid-Range Optimization

On a "mid-range" planning level, the nuclear power plant under consideration is fixed in terms of size, type, location, etc., but is not fixed in terms of fuel or fuel management. Optimal fuel enrichment, plant capacity factor, and duration of fueling interval are closely related variables linked by system optimization, so that the optimal fuel enrichment for future fueling intervals is treated as a planning objective. A method of analysis is developed for steady state operation, i.e., identical capacity factor and duration of each fueling interval. This permits analytical treatment of nuclear fuel-cycle costs.

The time unit of analysis is the fueling interval (time between refuelings) of duration T. The objective is the minimization of total utility system costs C^S incurred in meeting the demand for electricity, E^S. The optimization variable is the enrichment of the nuclear fuel, or interchangeably, the fuel energy potential E^i or the design capacity factor L:

$$E^i = K\ T\ L \qquad (1)$$

(K = rated capacity of nuclear plant - kw).

The optimization is formulated as a non-linear programming problem. The objective function C^S is broken down into a fuel-cycle cost term for the nuclear plant to be optimized, C^i, and a production cost term of the rest of the (lumped) utility system, C^{S-i}.

$$C^S = C^{S-i} + C^i \qquad (2)$$

The energy production by the nuclear plant, E^i, plus the production of the rest of the system, E^{S-i}, are equal to the demand for electricity, E^S, during the fueling interval under consideration.

$$E^S = E^{S-i} + E^i \qquad (3)$$

The optimality conditions (Kuhn-Tucker)

$$\frac{dC^i}{dE^i} + \lambda = \frac{dC^{S-i}}{dE^{S-i}} \qquad (4)$$

define two mid-range marginal cost terms: dC^i/dE^i is called "mid-range marginal cost of nuclear energy" and is denoted by MMC^i. MMC^i vs. capacity factor L of the nuclear plant defines a nuclear supply curve. dC^{S-i}/dE^{S-i} is called "system opportunity cost" and is denoted by MMC^{S-i}. MMC^{S-i} vs. L defines an internal demand curve for electricity from the nuclear plant, which indicates the capacity factor L at which the dispatcher would load the nuclear plant during the fueling interval for any marginal cost of nuclear energy and can be evaluated by simulation of the dispatching during the future fueling interval. λ is a "shadow" price. If $\lambda = 0$, the optimality condition may be interpreted as the intersection of the internal demand curve, MMC^{S-i}, and the nuclear supply curve, MC^i. For $\lambda \neq 0$, the constraint is active that the nuclear energy supply cannot exceed the production potential given by the product of rated capacity K and duration of irradiation period T' ($T'=T(1-R)$, where R = fueling down-time fraction).

In order to be able to treat analytically mid-range marginal costs of nuclear energy, the fuel-cycle cost C^i incurred by n fuel batches during one fueling interval is replaced by the cost C incurred by one fuel batch during n fueling intervals (this is true for steady state operation). Identically, E^i is replaced by E. MMC^i is given by the partial derivative of C with respect to E holding fixed the fueling interval duration T, which is an external parameter in the analysis:

$$MMC^i = \left.\frac{\partial C}{\partial E}\right|_{T} \tag{5}$$

For optimal duration of the fueling interval, $T=T^*$, i.e., for the T that minimizes average fuel-cycle costs for a capacity factor L, MMC^i is equal to interest free average fuel-cycle costs e_o (free of interest during the in-core period of the fuel batch; pre- and post-irradiation interest is included in e_o). For $T \neq T^*$, MMC^i has a slightly different value.

For present day light-water reactors, e_o is on the order of 1.2...1.5 mills/kwh for capacity factors $L > 0.5$, and is therefore very low in comparison with system opportunity costs (of the order of 2...2.5 mills/kwh and higher). Therefore at present the production potential constraint is, in general, active, and strong incentives exist to provide for large fuel energy potentials, E^i.

However, when designing the fuel for such energy potentials, a risk may be incurred in that, due to future forced outages, depletion of E^i during T' may not be possible, and at the end of the irradiation period the fuel must either be discharged undepleted or the fueling date must be deferred. In both cases, a cost penalty is incurred. The analysis presented here applies for a fixed fueling schedule, and the risk is quantified by a risk premium, RP. RP is added to mid-range marginal costs and defines an expected nuclear supply curve, $EMMC^i$:

$$EMMC^i = MMC^i + RP \qquad (6)$$

This risk premium is found to be

$$RP = \frac{1-p(E^i)}{p(E^i)} (MMC^i - MC^i) \qquad (7)$$

where $p(E^i)$ is the left tail integral of the forced outage distribution from 0 to $(1-L/L_{max})$ (L_{max} = maximum capacity factor if no outages occur, L = actual design capacity factor). MC^i is the discounted marginal cost of isotope consumption (of the order of 0.5-mills/kwh for present day LWR's) and is not a "risked" cost but is incurred like costs of oil or coal. As L approaches L_{max}, the RP increases rapidly.

The intersection of internal demand curve MMC^{S-i} and expected nuclear supply curve, $EMMC^i$, determine the optimal design capacity factor L* and the optimal fuel energy potential E^i_*. In general, at present where nuclear plants represent a small fraction of system generating capacity, E^i_* is larger than the amount of energy that can be expected to be generated with certainty, and may, in general, even be larger (by 1 to 3%) than the amount of energy that can be generated on average (with the expected forced outages). This is due to the strong present-day economic incentives to make utmost use of nuclear fuel. The optimal amount of excess energy can be determined by the risk premium method. In the future, when the fraction of nuclear generating equipment will have become large, system opportunity costs will therefore be lower, and the intersection of the internal demand and nuclear supply curves should take place at lower values of L, where the risk premium practically vanishes.

2.3 Short-Range Optimization

On a "short-range planning level", a nuclear power plant with fixed fueling schedule is specified in terms of size, type, location, etc., and its fuel load and next fueling data are also fixed parameters. In the optimization of the utilization of such a plant in the generating system of a utility company, the time unit of analysis is the remainder of the current fueling interval from "now" (time t) to shutdown for fueling at time T'. A method is presented for evaluating the marginal cost of energy which when used for dispatching of the nuclear plant leads to optimal utilization of the nuclear fuel.

Short-range (re-)optimization may be necessary because mid-range planning quantities (fuel enrichment and duration of fueling interval) are based on assumed future states for the nuclear reactor, the system, and the demand for electricity which, by virtue of their stochastic nature, may not be realized in future actual operation.

At any time, t, during the current fueling interval, the optimization of plant utilization can be formulated as a nonlinear programming problem with the objective of minimizing the total system production costs $C^S(t)$ incurred during the remainder of the current fueling interval in meeting the demand for electricity, $E^S(t)$.

Equations (1) and (2) above apply to the remainder of the current refueling interval. The optimality conditions (Kuhn-Tucker) are again given by (see Equation (4))

$$\frac{dC^{i}(t)}{dE^{i}(t)} + \lambda = \frac{dC^{S-i}(t)}{dE^{S-i}(t)} \qquad (8)$$

where λ is a shadow price and arises from the constraint that $E^{i}(t)$ cannot exceed the fuel energy potential, $E^{i}_{*}(t)$, available for depletion during the remainder of the fueling interval.

$dC^{S-i}(t)/dE^{S-i}(t)$, the "system opportunity cost" denoted by $MC^{S-i}(t)$ defines an "internal demand curve" for nuclear energy. This demand curve can be evaluated by simulation of the system dispatching process during the remainder of the fueling interval.

$dC^{i}(t)/dE^{i}(t)$, the "short-range marginal cost of nuclear energy" denoted by $MC^{i}(t)$ defines a "nuclear supply curve". For optimal operation either $MC^{i}(t)$ and $MC^{S-i}(t)$ intersect, or the constraint that the nuclear energy production cannot exceed the fuel energy potential, $E^{i}(t) \leq E^{i}_{*}(t)$, is active (so that, $\lambda = 0$).

Since during any fixed fueling interval, out of the total fuel-cycle cost C^{i} only the discounted fuel salvage value SV(t), is variable, $MC^{i}(t)$ is given by

$$MC^{i}(t) = PV\left(\frac{dSV(t)}{dE^{i}(t)}\right) \qquad (9)$$

(PV(...) = present value of (...)). This cost may be interpreted as the discounted marginal cost of isotope consumption. $MC^{i}(t)$ is on the order of 0.4 to 0.7 mills/kwh for present day light water reactors.

Since present day system opportunity costs, $MC^{S-i}(t)$, are higher than this by a factor of three or four, the incentives to deplete a given fuel energy potential, $E^{i}_{*}(t)$, during the remainder of the fueling interval from time t to shut-down for fueling at time T'. However in the same time, $E^{i}_{*}(t)$ represents a scarce resource. As the optimality condition indicates, for optimal dispatching the marginal value of the nuclear energy potential to the system, $MC^i(t)$ plus the shadow price λ, must be used. This is equal to the system opportunity costs at the constraint that the nuclear energy production cannot exceed the energy potential.

Thus, since $E_*^i(t)$ will be depleted, if possible, from time t to the refueling data T', the average capacity factor L' during the time the plant is available should be

$$L'(t) = E_*^i(t)/K(T'-t)(1-F) \leq 1$$

K = rated capacity

F = mean forced outage fraction of nuclear plant.

The factor (1-F) takes into account possible forced outages from time t to T' and represents, in engineering terms, a safety factor. L'(t) cannot exceed unity.

System opportunity costs at $E_*^i(t)$ are the highest marginal cost for dispatching of the nuclear plant with which $E_*^i(t)$ can just be depleted within (T'-t). This implies, that the fuel energy potential will be used to displace the highest possible marginal generating equipment of the generating system and would carry peak load if $E_*^i(t)$ was small in comparison with the plant energy production potential given by the product of plant capacity K and irradiation time (T'-t). However the plant would carry base-load if the fuel energy potential $E_*^i(t)$ was relatively large. In this respect, nuclear power behaves like storage hydro power.

In the future as the fraction of nuclear capacity becomes significant, system opportunity and the short-range marginal costs of nuclear energy will become equal, so that the economic availability-based capacity factor will not necessarily be unity and the incentives during reoptimization toward shifting of scheduled refueling dates should increase.

3.0 System Modeling and Optimization

3.1 Introduction

The analytic treatment of the steady state case for mid-range optimization gives valuable insights into the general characteristics of nuclear marginal costs and indicates how a probabilistic approach provides valuable information when planning for the future. However in practice, steady state nuclear behavior from one refueling to another probably will not be normal and the quantitative results of the analytic solution are not precisely applicable. Considering the large sums of money involved in buying and financing the various materials and services required in the nuclear fuel cycle, approximations based on analytic solutions are not tolerable, so that accurate modeling and optimization of system performance are required.

As seen by the Commonwealth - M.I.T. joint effort, for utility systems containing nuclear power plants there are four time scales which have different characteristics and constraints and hence probably will require somewhat different simulation models. These are:

(1) Daily Model: This time scale deals with the hour-by-hour dispatching of the various generating units. Only a small fraction of the energy potential in the nuclear fuel is released and the sole design and operating parameter available for optimization is the power output of each plant.

(2) Annual Model: This time scale deals with operation of the nuclear plants between refuelings. The fuel in a reactor cannot be changed but the power level of the reactor, date of the next refueling, and energy potential of the discharge fuel are decision variables for each station. In the study by Widmer (1, 8) this time scale was referred to as short-range.

(3) Multi-year Model: This time scale spans the time required for the complete fuel cycle of about 5 to 10 years. In addition to the variables mentioned for the annual model, this one includes fuel management variables of reload fuel enrichment and batch size. This time scale plays the determining role in planning for the purchase of fuel and its required processing and fabrication operations as well as financing all these costs. In the study by Widmer (1,9) this time scale was referred to as mid-range.

(4) Expansion Model: This time scale covers a period of many years -- of the order of the expected lifetime of generating stations -- and is employed in planning for the addition and retirement of generating equipment. For the first three models, certain plants are assumed to exist or to have been ordered so that the type and characteristics of each unit are specified, while in the expansion model a variety of new energy production equipment is under consideration.

Multi-year considerations vitally affect decisions regarding long term fuel financial commitments. Furthermore, a multi-year model is expected to have elements useful in the development of the other three models required. Consequently, the two simulation and optimization projects which have been initiated at M.I.T. under the Commonwealth Edison program are directed primarily at developing a multi-year model. A schematic diagram of an overall multi-year model for nuclear power management is shown in Figure 1, which is a slightly modified version of a diagram presented by W. M. Kiefer and E. F. Koncel in a letter to M.I.T. (10). The project to be outlined in Section 3.2 applies to the area labeled I in Figure 1 while the project discussed in Section 3.3 applies to the area labeled II.

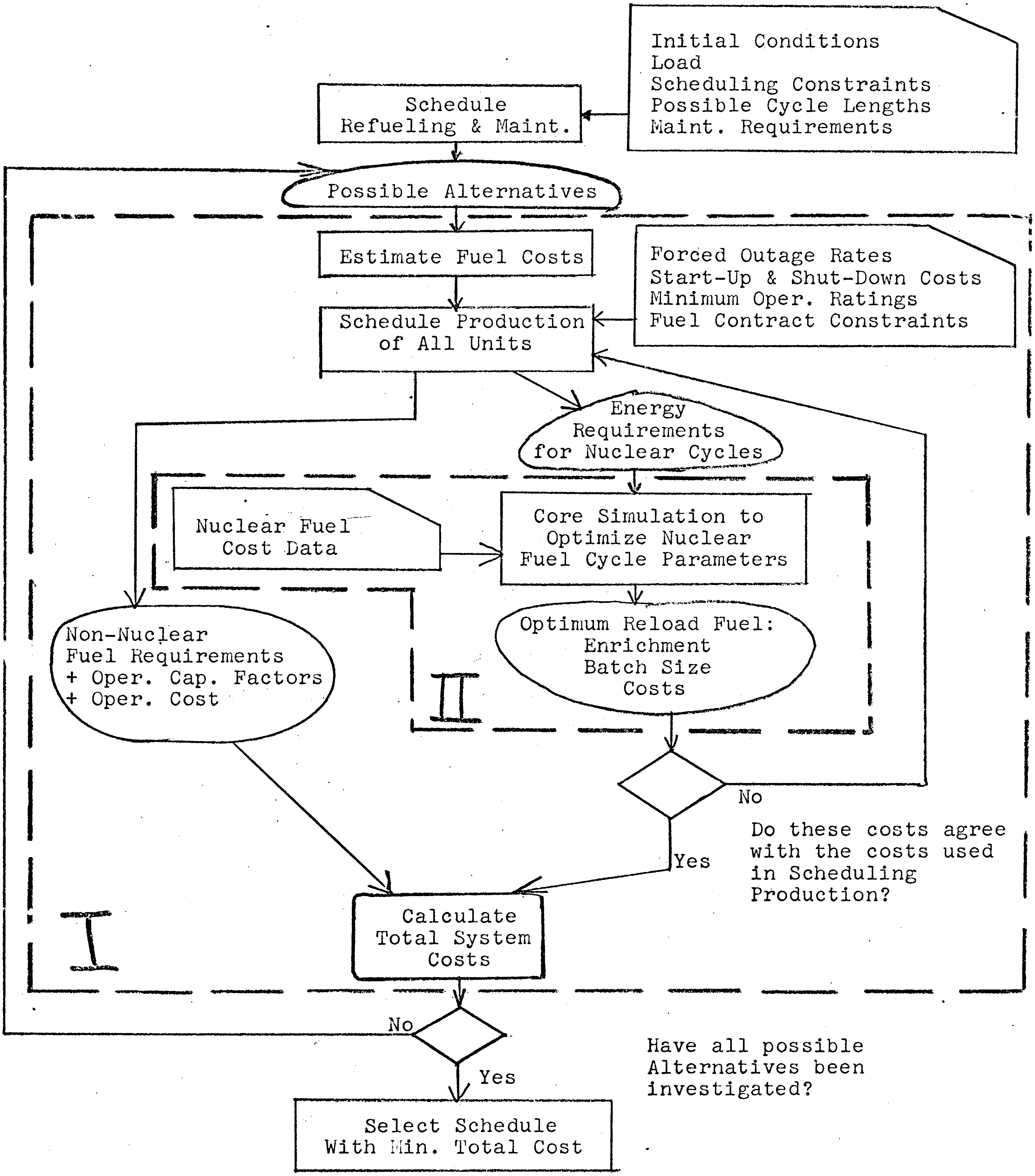

Figure 1 Nuclear Power Management Multi-Year Model

3.2 System Integration and Optimization

Following a period of several months in which E.A. Mason and P.F. Deaton became familiar with the Commonwealth Edison Company's present methods of system planning and dispatching and studied methods of systems simulation and optimization, Commonwealth Edison Company proposed that the M.I.T. group make recommendations concerning the 1970 summer operation of Dresden Units 1 and 2. The analysis reached the following conclusions: (a) the only time full power operation of these nuclear units would not be possible was during early morning low-load periods during the weekdays of a few low-load weeks and (b) many hours before the nuclear plants would be forced to load-follow, it would be economically justifiable on a daily basis to shutdown plants previously kept on the line. Recommendations which followed from the analysis were:

(a) re-evaluation of physical constraints presently justifying continuous operations of larger plants,

(b) re-evaluation of startup and shutdown costs for all plants considered flexible enough to permit overnight or weekend shutdown, and

(c) development of an n-plant dynamic programming package similar to Joy's work (11) which would allow (a) and (b) to be incorporated into a new overnight operating policy.

Having demonstrated an understanding of utility problems and practices, work was begun to develop a nuclear power management system model along lines remarkably similar to those of of the Kiefer-Koncel model and is shown in Figure 1. The multi-year system model will perform those tasks outlined in area I of Figure 1. Input data includes all fossil operating and maintenance data, some in a precalculated form to eliminate repetitious calculations. Joy's initial work on dynamic programming (11) may be incorporated at this point. Nuclear input is via the refueling dates of the "Possible Alternative" being considered and various nuclear operating constraints (e.g., xenon override must be available all summer). After estimating (or using revised estimates of) marginal nuclear costs, production is optimally scheduled from all fossil and nuclear units. The crux of the model, the optimization method, is under investigation but tentatively, a network algorithm looks promising. Ordinary linear programming fails to exploit the more easily calculated network structure while dynamic programming appears to be at a disadvantage as the number of reactors increases.

In addition to the model's output indicated in Figure 1, both "opportunity cost" or "value" as well as the "production cost" of nuclear power will be indicated. These nuclear opportunity costs will be useful to the dispatcher in achieving the production goals during each period.

The advantages of such a model are numerous. As mentioned above, the non-steady-state cases one encounters in real utility operation can be simulated easily. It allows one to now answer <u>system</u> questions rather than just <u>reactor</u> questions. Additionally, it will itself <u>ask</u> questions: What attributes of the system caused this seemingly inefficient reactor strategy to result in large system savings? As pointed out above, this will be particularly useful in pinpointing important research areas and further model improvement. After running many problems, one would hope to also benefit by deducing "rules-of-thumb" which, though not perfect for all circumstances, often result in near-optimal decisions.

3.3 Nuclear Simulation Modeling

As is evident in the block diagram of Figure 1, the Core Simulation model forms an integral part of the overall system optimization task. One or more in-core simulation codes must be selected or developed and an optimization methods developed for calculating and selecting optimum nuclear parameters - enrichment, number of zones and batch sizes. Having been given the energy requirements and refueling times, a set of E's and T's, for each nuclear plant over the multi-year period this code in conjunction with an optimization technique will find the nuclear parameters that minimize the nuclear fuel costs.

These costs are used in two capacities:

1) to check the estimated costs which were used to establish the scheduled operation of the nuclear plants and
2) to be added to the non-nuclear costs to find the total system costs.

At present the code CELL MOVE is being tested for its viability as the nuclear parameter calculator for PWR's. If the optimization technique used requires varying the enrichment and batch size a very great deal for each fueling interval, then CELL MOVE probably would be inadequate. The code's ability to consider burnup histories and core characteristic s for more than a few refuel batches containing different enrichments and/or different batch sizes for each reactor is severly limited. However in the case where many options are to be considered at any decision point, computer interpolation among precalculated values of reactor parameters may be required in any case.

The optimization procedure is schematically represented by the feedback loop shown in Figure 2.

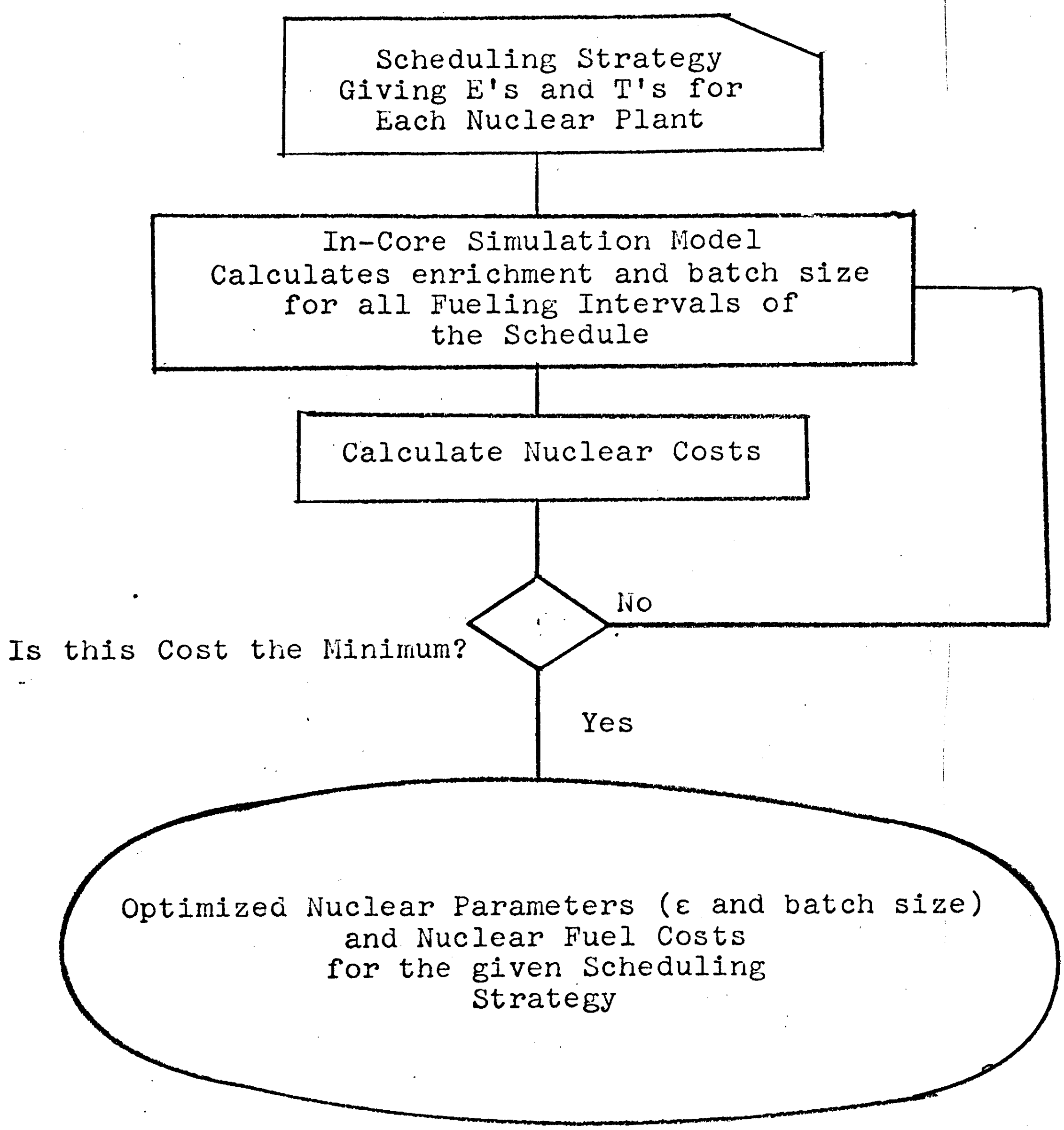

Figure 2   Nuclear Simulation and Optimization

Once this core simulation and optimization capacity is developed it will be used for scoping studies if the entire system model is not yet completed. These scoping studies will establish economically feasible operating regions for the parameters, T, enrichment, batch size and number of zones.

Knowledge of these feasible regions will make not only the system optimization of the multi-year model but also other nuclear system optimizations converge to solutions much faster.

3.4 Nuclear Core Code Development

Two master thesis projects relating to nuclear core code development were completed early in the Fall of 1970. Both of these projects were aimed at increasing M.I.T.'s capabilities for rapid nuclear core simulation calculations for use in the overall model development described in Section 3.3 and for use in specific sensitivity studies (see Section 4.0).

3.4.1 CELL-MOVE and 2DB

The first thesis made improvements in two existing nuclear fuel depletion codes, CELL-MOVE and 2DB, and compared the results produced.

The first objective in this project was to improve the depletion results predicted by the modified two group diffusion depletion code CELL-MOVE. Specifically, to decrease the calculated ratio of maximum to central mid-plane flux after fresh fuel is added to the reactor.

The second objective was to modify the fast reactor multigroup multiregion diffusion depletion code 2DB in order to yield satisfactory results for thermal reactors. Areas of change include burnup model, treatment of Xe and Sm, calculation of diffusion coefficient, energy per fission, and fission yields.

The final objective was to compare results predicted by the improved computer code CELL-MOVE and the results predicted by the modified computer code 2DB. Comparisons were made with regard to thermal flux shapes, power density distributions, keff values and fuel burnup during an equilibrium cycle of the Donald C. Cook PWR.

The present version of the computer code CELL-MOVE was written as two separate M.I.T.-developed codes, CELL and MOVESC II. The CELL code is a pseudo two-group point depletion code which calculates unit cell properties as a function of thermal flux-time. The code MOVESC II uses this information plus geometric input data to calculate spatial flux shapes and power densities during fuel irradiation.

The code CELL assumes that a unit cell can be represented by a volume and flux weighted homogenized unit cell and that nuclide concentrations and unit cell properties can be effectively calculated as simple functions of thermal flux-time. The fast non-leakage probability is an input parameter and is assumed to remain constant during irradiation. Neutron energy spectra are assumed to be independent of control poison effects.

The code MOVESC II has evolved from MOVE, the original version, and was written especially to simulate modified scatter refueling in PWR's with a variable number of batches. The MOVESC II code uses CELL output and geometric data to calculate reactivity and flux distributions changes during irradiation. A pseudo-two group diffusion theory calculation is used to arrive at converged flux distributions in R-Z geometry. The code can use a maximum of fifteen axial and ten radial mesh points in calculating reactor characteristics.

The pseudo two group treatment uses two groups of neutron leakage but only iterates on the thermal flux. As irradiation proceeds, unit cell properties are determined by the flux-time experienced at each mesh point by using the functions generated by CELL. When end of life is reached, fresh fuel with zero flux-time replaces the irradiated fuel in the outer region which replaces the most depleted batch of fuel in the scatter region.

At present, CELL-MOVE must be used for thermal reactors employing a modified scatter scheme with the number of batches less than or equal to ten. The code begins its first cycle with fuel of a given enrichment placed throughout the core. It then steps out in time until keff = 1.000. At this point, fuel from the outer region replaces the most depleted fuel in the scatter region and fresh fuel is added to the vacated outer region.

In CELL-MOVE, the fission product yields and values of ν for the fissile isotopes were updated and the number of radial mesh points permitted was increased from 10 to 40. As a result of depletion runs made with the D.C. Cook PWR parameters a number of suggestions were made for further improvement; some of these have already been introduced (4).

As a result of these new modifications to the code CELL-MOVE, its usefulness for fuel management calculations is improved. Because of its modified two group flux calculation and reflector treatment, it is not a highly accurate code. Its low computer running time should strongly influence the adoption of the code whenever the accuracy sought is not too great. The code can be extremely beneficial in survey-type work in which many different cases must be considered and an overall effect is looked for. The computer code CELL-MOVE is based on relatively simple, yet sound principles and even though the degree of approximation is high, the low running time of the code is a very influential factor.

The two dimensional multigroup multiregion diffusion depletion code 2DB was written by W. W. Little, Jr., and

R. W. Hardie for use in fast reactor criticality and burnup analysis (12). It was believed, that with minor changes, it could be converted for thermal reactor use.

The code can explicitly treat many regions, including reflectors, and also calculate group fluxes using a sound multigroup treatment. The previous concepts were the basic cause of approximation in the code CELL-MOVE. It was believed that this more sophistocated code would be ideal to use for a comparison. Even though the code 2DB has a considerably longer running time, if converted, it could prove to be an extremely valuable tool when used in conjunction with CELL-MOVE.

The code 2DB requires approximately four minutes of computer time for each burn-up time step. It is suggested that time steps do not exceed seventy-five days if rapid variations in isotopic concentrations or flux shapes are anticipated.

Since the code 2DB was written explicitly for fast reactors, its input parameters must be used wisely so as to simulate a thermal reactor. Because of fast reactor fuel loading simplicity, no treatment of modified scatter fuel loading schemes were incorporated in the code. Also, because of the peaked flux shape and differently enriched fuel elements a more detailed picture of the reactor is needed.

Major areas requiring modification include adding a provision for calculating xenon and samarium equilibrium concentrations, investigating the accuracy of the approximations used in depletion calculations and providing an accurate treatment for slowing down materials for neutron moderation. The inadequacies of the code 2DB for treatment of thermal reactors are caused by the following: xenon and samarium build-up are not important in fast reactors because of their small microscopic cross sections at high energies; in fast reactors, material number densities are not rapidly varying; and in fast reactors moderating materials, such as hydrogen and light water, are not present.

Considering its demonstrated speed and usefulness for fast reactor depletion calculations, several modifications were made in 2DB in an attempt to adopt it for use in thermal reactor fuel depletion calculations. However evaluation of the results obtained using the D.C. Cook PWR as an example has indicated that the code 2DB, in its presently modified form, is not reliable for the treatment of thermal reactors due to the inadequacy of the burn-up model used. The method of solution of the nuclide rate equations are too approximate for thermal reactors. This inaccuracy arises because of rapidly varying concentrations isotopes (other than xonon and samarium) which are present in thermal reactors.

Highly accurate solutions are needed in thermal reactors for number densities because of the large effect small number density changes have on reactor properties. This arises because thermal cross sections are much higher than fast group cross sections.

Nuclide concentrations, for most isotopes predicted by the code 2DB after irradiation has proceeded, are too low. In a chain of nuclides, some nuclides are "lost" at the end of a burnup step.

A number of specific recommendations were developed concerning further modifications in the code 2DB for fuel management in thermal reactors.

The code CELL-MOVE makes two major assumptions. One is the approximate treatment of the reflector region by using a reflector savings value. The other is the treatment of only the thermal group of neutrons in flux and burnup calculations. These approximations yield substantial savings in the running time of the code.

The code 2DB removes the major assumptions of the code CELL-MOVE by explicitly treating a reflector region and using as many as fifty groups of neutrons. Even through the code is characteristic of a substantially longer running time, its degree of accuracy will definitely be higher.

Since the code 2DB failed to accurately treat fuel depletion, comparisons could not be made concerning discharged burnup and time at full power. However, initial and equilibrium flux and power density distributions can be compared.

3.4.2 FLARE Code Evaluation and Modification

Although considerable work has been carried out at M.I.T. on the development and use of two-dimensional nuclear depletion codes (which are quite useful for PWR's) no significant work had been performed with rapid three-dimensional depletion codes, which are required to handle BWR's. A master's thesis project(3) was directed at adopting an existing simple 3-D program, FLARE, to M.I.T.'s computer and to make certain desirable improvements.

In 1964, the General Electric Company developed a 3-D boiling water simulator code, FLARE. The FLARE code has been modified in various proprietary forms, such as the TRILUX-ISOLUX codes of United Nuclear Corporation, but these versions have not been released to the general public.

With the desire to make improvements in the original FLARE code, a study of the logic structure of the program was undertaken. The code was changed to reduce running time, correct logic mistakes found in the version available, provide additional user options for programming convenience, and modify the physics calculations for improved accuracy. The modified version has been labeled FLARE-G.

The FLARE-G code has been changed in several areas from the original program distributed by the General Electric Company.

Numerous logic errors were found in the version of FLARE obtained and they were corrected. Discrepancies in the methods of volume averaging and applications of the boundary conditions at lines of symmetry were the most significant of these.

A new quarter core symmetry option has been included in FLARE-G -- quarter core -- no center nodal line -- mirror symmetry -- as need arose by users of the original code for the added flexibility which this option gives.

The original FLARE code depends on two direct access devices or scratch tapes for operation. FLARE-G provides for all data to be stored in the computer core, thus reducing execution time of the code.

In the original FLARE, the code had no provision for positioning the control elements between time steps. A new case had to be started each time a different pattern was required. Besides being time consuming, this process did not always give the desired reactor eigenvalue (usually 1.00) since the rod pattern was pre-calculated. FLARE-G contains a control element positioning subroutine which iterates rod positions in a programmed sequence until a desired reactor eigenvalue is reached at each time step.

Two changes were made in the physics calculations. The migration area at a position is now calculated as a function of both control fraction and relative moderator density, instead of relative moderator density only, as is the case in FLARE. The second transport kernel option of FLARE has been altered to take into account transport from either six adjacent or eighteen surrounding nodes with arbitrary mixing constants determining the magnitude of the kernels and the volume of their effectiveness. This simple calculational approach is used in an effort to account for passage of fast neutrons beyond nodes adjacent to the source node, a source of error in FLARE, without appreciably affecting the running time of the code.

In order to prove the applicability of the new program, FLARE-G, to operational and planned BWR and PWR plants, calculated results were compared to TRILUX and measured data on the 700 MWt Dresden 1 plant and to CELL-MOVE and Westinghouse predicted for the 1346 MWt San Onofre 1.

The FLARE-G code was found to work well with the 700 MWt BWR system considered, but the results obtained show that in working with a boiling core, the constant values used as input for the code, especially in the area of fluid flow and void-quality relations, should be checked with thermal-hydraulic calculations of a more sophisticated nature for consistently accurate results.

When applied to a 1346 MWt PWR system, the FLARE-G code provided excellent results in both power and burnup distribution predictions, indicating that the code can be applied to non-boiling cores as well as the boiling reactor without loss of accuracy.

4.0 Sensitivity Studies

Two masters thesis projects are nearing completion which have dealt with

(a) the effect of refueling a nuclear reactor either earlier or later than had been scheduled, and

(b) the effect of fuel stretch out.

Details of the results will be transmitted in about a month, when the reports on these projects have been completed and submitted.

The objective of the first study is to evaluate the economic consequences of removing a batch of fuel from the reactor before its design reactivity-limited burn-up is achieved. This situation might be experienced if the plant had undergone a forced outage of one or more months duration and the originally scheduled refueling date was adhered to. It is desirable to know the economic consequences of off-design refueling in order to compare them with the costs of alternate strategies, such as plant derating or rescheduling of refueling.

Steady state fueling was achieved and then it was assumed that only the refueling date (or discharge batch exposure) was altered and the discharge batch replaced with a normal batch of fuel (same amount and enrichment as a steady state batch). The effect of both variable refueling time at constant fuel burnup and of variable fuel burnup were considered. Most of the cases considered were for refueling a batch before the design reactivity-limited exposure was achieved. The economic analysis is not yet complete, but it was found that the reactor returns to its original steady-state behavior with respect to fuel burnup in about four or five refueling intervals following the abnormal refueling date.

The second study concentrated more on the physics and engineering consequences of fuel stretch-out through plant derating. The energy attainable was calculated for various deratings. The effect of the stretch out on core power distribution and control requirements following refueling were also studied.

The economic evaluation of these results will be the subject of a subsequent study. Other masters thesis projects planned include evaluation of variation of refueling batch size and enrichment as means of accomodating changes in expected systems performance so as to provide flexibility in systems planning.

5.0 References

(1) H. Widmer, "The Optimal Utilization of a Nuclear Power Plant With Fixed Fueling Schedule," Ph.D. Thesis, Department of Nuclear Engineering, M.I.T., September 1970.

(2) W. R. D'Angelo, "Comparison of Nuclear Fuel Depletion Codes: Cell-Move and 2 DB," S.M. Thesis, Department of Nuclear Engineering, M.I.T., September 1970.

(3) P. N. Ontko, "FLARE G, A Three-Dimensional Reactor Simulator," S.M. Thesis, Department of Nuclear Engineering, M.I.T., February 1971.

(4) D. F. Martin, "The Effect of Refueling Date Changes on Fuel Cycle Costs," S.M. Thesis, Department of Nuclear Engineering, M.I.T., in preparation, February 1971.

(5) R. M. Banister, "Considerations Pertaining to Power Reactor Core Stretchout in Pressurized Water Reactors," S.M. Thesis, Department of Nuclear Engineering, M.I.T., in preparation, February 1971.

(6) Paul F. Deaton, Ph.D. Thesis, Department of Nuclear Engineering, M.I.T., in process.

(7) Joseph P. Kearney, Ph.D. Thesis, Department of Nuclear Engineering, M.I.T., in process.

(8) Hans Widmer, Edward A. Mason, Paul W. MacAvoy, "Mid-Range Optimization of a Nuclear Power Plant," Trans. Am. Nuc. Soc. 13, No. 2, 782-85 (November 1970).

(9) Hans Widmer, Edward A. Mason, Paul W. MacAvoy, "Short-Range Optimization of Use of a Nuclear Power Plant," Trans. Am. Nuc. Soc. 13, No. 2, 785-6 (November 1970).

(10) W. M. Kiefer and E. F. Koncel, Personal Communication to E. Mason, M.I.T., dated November 30, 1970.

(11) David S. Joy, "A Dynamic Programming Solution of the Electric Generating System Unit Commitment Problem," ORNL-TM-2881 (April 1970).

(12) W. W. Little, Jr., and R. W. Hardie, "2DB User's Manual-Revision 1," BNWL-831 (1969).