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# FINAL REPORT ON IMPROVED URANIUM UTILIZATION IN PWRs

by

M. J. DRISCOLL

DOE Contract No. DE-AC02-79ET34022
Energy Laboratory Report No. MIT-EL-82-032
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and
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NOTICE

## ABSTRACT

This is the final summary progress report on a research program carried out within the MIT Energy Laboratory/Nuclear Engineering Department under the US Department of Energy's program to increase the effectiveness of uranium utilization in light water reactors on the once-through fuel cycle.

Two major themes, methodology and applications, characterize the research. A simple but accurate set of algorithms, designated as "the linear reactivity method" were developed to permit self-consistent evaluations of a broad spectrum of changes in core design and fuel management tactics.

More than a dozen suggested improvements were then evaluated, focusing primarily on retrofitable modifications and pressurized water reactors. In common with the findings of many other investigators, high burnup and routine end-of-cycle coastdown were identified as preferred options.

## ACKNOWLEDGEMENTS

The principal investigator wishes to acknowledge the many contributions of his colleague in these efforts, Professor David D. Lanning. Also greatly appreciated are the technical and programmatic advice and review provided by Dr. Peter M. Lang and William E. Murphie of the U.S. Department of Energy.

This work was carried out by a legion of students - some two dozen, if one includes the precursor NASAP/INFCE-related studies. Their many creative contributions and dedicated attention to detail is evident from the record. The full roster of major contributors is documented in the appendices to this report. The doctoral research by J. A. Sefcik, W. T. Loh and A. Kamal is especially noteworthy, forming, as it does, the backbone of the entire effort.

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# 1. INTRODUCTION

## 1.1 Foreword

This is the Final Report under DOE Contract No. DE-AC02-79ET 34022, summarizing the results of work carried out at MIT under DOE's LWR Technology Program for Improved Uranium Utilization.

The MIT effort under this program focused on pressurized water reactors operating on a once-through fuel cycle, and on retrofitable changes in assembly design and fuel management strategy which would increase the amount of energy extracted from a given amount of natural uranium.

A major element of our work turned out to be the development of a simple, but accurate, methodology for the evaluation of modifications on an all-else-being-equal basis. This was necessitated by the expense of fully-fledged state-of-the-art neutronics computations, and the difficulty, inherent in their execution, of isolating the effect of small perturbations. This formulation,designated the "linear-reactivity model", appears in various guises in all of the major topical reports issued to document the project's efforts.

With this modelling method in hand, it is no exaggeration to say that essentially all ideas advanced by participants in the DOE program were screened for their uranium-saving potential. For this reason, the specific findings are somewhat eclectic in nature - as will become apparent subsequently when the savings are cataloged.

## 1.2 Organization of the report

Because of the central importance of the linear reactivity model, Section 2 presents an abbreviated description of its main features. Much more elaborate versions, as embodied in a series of computer programs - ALARM, DISBURN and SPILBAC (by Sefcik, Loh and Kamal, respectively: see references in Appendix A) - are discussed in topical reports. The resultsoriented reader may wish to skip this section.

In Section 3 a compilation of major findings is presented. Here we try to sort out a vast array of information: the essence of those results which are most pertinent to the decision-maker.

In Section 4 the principal investigator has availed himself of the opportunity to comment on some of the broader issues.

This is followed by three appendices which play an important role in this report:

Appendix A - which lists all publications associated with this project, and reproduces the abstracts of all major reports and theses.

Appendix B - a copy of a major paper on thorium utilization which puts the current project's efforts into perspective with respect to prior MIT research.

Appendix C - the final report on MIT's precursor work under AEC/ERDA/DOE's NASAP/INFCE efforts, which evolved directly into the work reported here. This document was not previously given widespread circulation, and is quite germane to much of the follow-on effort.

# 2. REACTOR MODELLING

## 2.1 Introduction

Although appreciable use was made of the familiar LEOPARD and PDQ-7 programs in the project's efforts, it became clear early on that the tedious procedure of burning cores assembly-wise through several cycles to equilibrium was not a practical way to assess the uranium-saving potential of as many as two dozen alternatives. Even given unlimited computer time and funds the process is fraught with difficulties: it is all too easy to unintentionally obscure the effect of interest by introducing other changes in the evaluatory process - for example, changing the assembly loading map to a configuration which is not comparably optimized versus one's reference case.

Thus an effort was made to develop a simpler approach which retained sufficient accuracy to assess the effects of changes in reload assembly characteristics and fuel management strategy. In particular, the method would have to be sophisticated enough to handle all important options of current interest: out-in/scatter and low leakage loading patterns, burnable poison, coastdown, variable batch size, and the like. This framework, under the rubric of "the linear reactivity model (LRM)" was developed piecemeal at MIT over several years.

The basic features of the LRM are sketched in the section which follows. For more detailed treatises, including its embodiment in a variety of computer programs, see the reports by Sefcik, Loh and Kamal.

## 2.2 Discussion

The analytical basis needed to couple fuel neutronic characteristics to its burnup performance exploits the empirically established observation that assembly reactivity varies linearly with burnup (once the xenon and samarium concentrations have achieved saturation). Table 2.1 summarizes the other key features of this approach: a theoretically sound prescription (power weighting) to compute core-average reactivity, a radial leakage/peripheral assembly power correlation, and finally a prescription for estimating batch power fractions. In references (1) through (3), these relations have been validated against state-of-the-art computations. Analytical applications of this methodology have also been explored in some detail [4].

Several figures, taken from Ref. [2], are included to illustrate the power-sharing and radial leakage correlations used in the linear reactivity model of core behavior.

Discharge burnup for a steady-state reload batch of fuel is readily computed from the intercept and slope of the reactivity vs burnup curve under the approximation of equal power-sharing among in-core batches (equivalent to a uniform cycle-by-cycle power history for a given batch during its in-core residence). For an n-batch core:

$$B_{do} = \left(\frac{2n}{n+1}\right)\left(\frac{\rho_o}{A}\right) \text{ MWD/MT}$$

Under different assumptions (e.g. non-uniform power history) it is convenient to express this result as a small perturbation:

$$B_d = B_{do}/(1+\varepsilon)$$

The quantity $\varepsilon$ can be computed by both analytic and numerical approaches. It is worth noting that this correction is typically small: for a 3-batch core power history as severely skewed as $(\frac{1}{2}, \frac{1}{3}, \frac{1}{6})$, the quantity $(1+\varepsilon)$ is only 6% larger than the equal-power-sharing limit $(\frac{1}{3}, \frac{1}{3}, \frac{1}{3})$. Table 2.2 presents analytic solutions for $\varepsilon$ for several core burnup histories of interest.

Table 2.1

General Features of Model

1. Linear (unpoisoned) fuel batch reactivity vs. burnup

$$\rho = \rho_0 - AB$$

Determine using LEOPARD, for example.

2. Power-weighted reactivity balance

$$\bar{\rho}_{system} = \sum_{i=1}^{n} f_i \rho_i - \rho_L = 0 \text{ at EOC}$$

3. Radial leakage—peripheral power correlation

$$\rho_L = \alpha \sum_{periphery} f_j$$

4. Batch power sharing algorithm

$$f_i = \frac{\bar{f}}{1 - \theta(\rho_i - \alpha)}$$

where $\alpha = 0$ for interior assemblies

$$\theta = \left(1 + \frac{\gamma h^2}{m^2}\right) \approx 1 + \Sigma_R \cdot h = \text{constant}$$

$\bar{f}$ = core-average batch power fraction

can show that setting $\sum_{i=1}^{n} f_i = 1.0$ using Eq. (4) yields Eq. (2)

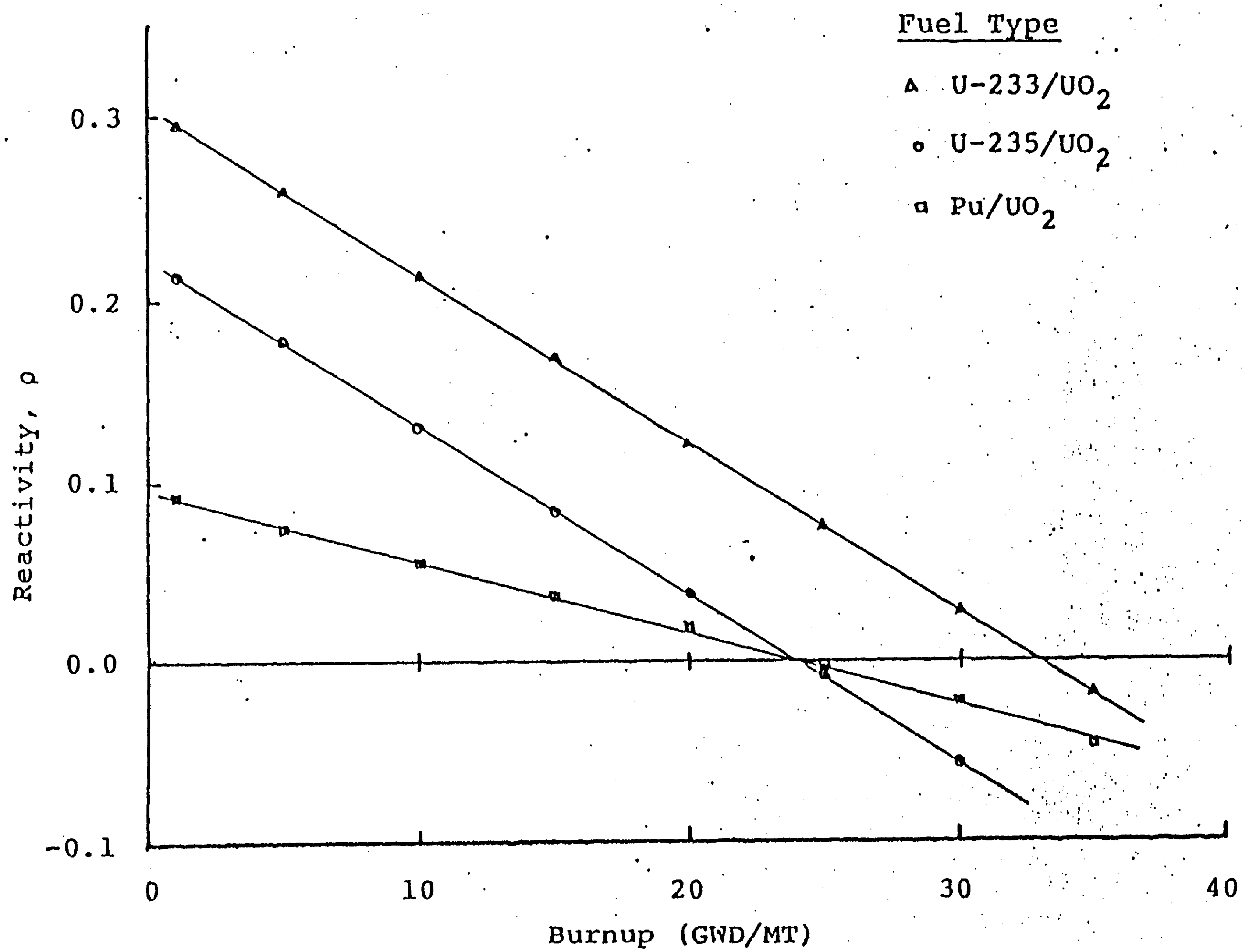



Fig. 2.1 Reactivity versus Burnup for Various Fissile Materials in $UO_2$

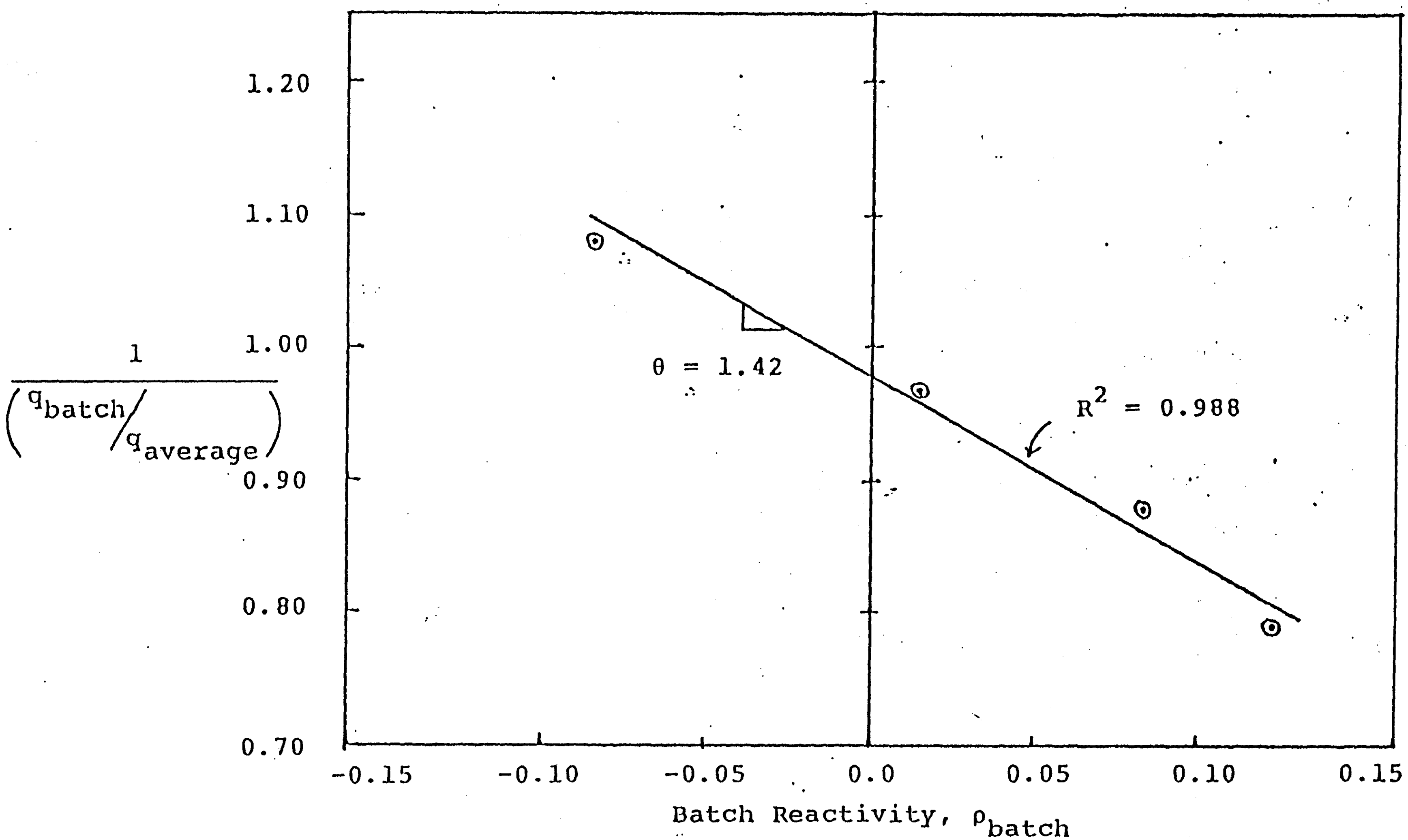



Fig. 2.2 Regression Line of Power vs. Reactivity for Main Yankee Fuel Batches

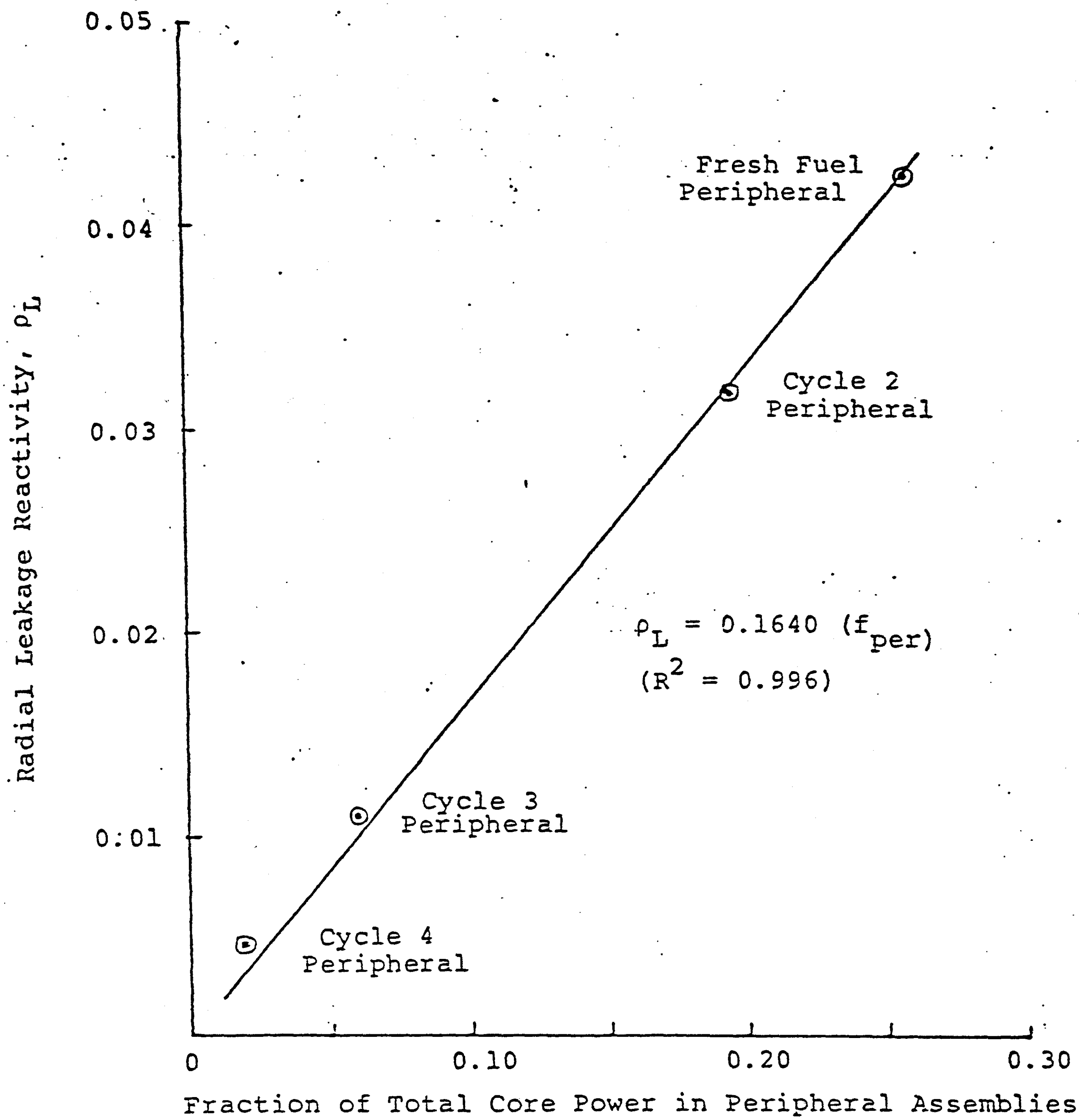
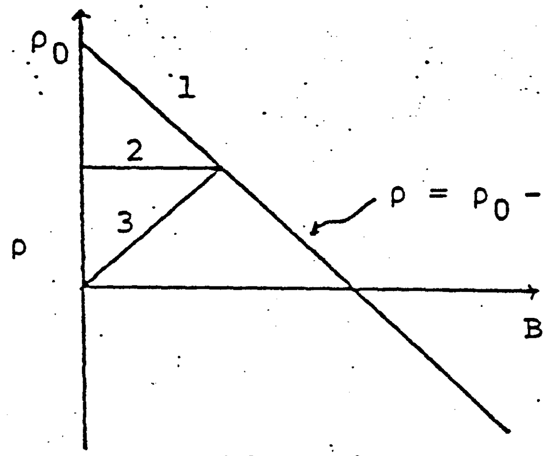


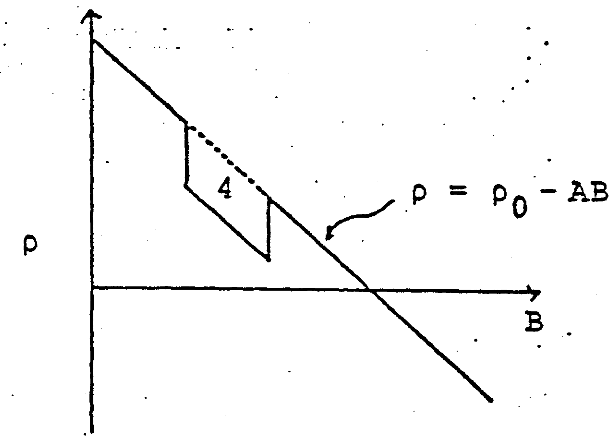

Fig. 2.3 Radial Leakage Reactivity for M.Y. Reactor vs. Fraction of Core Power Generated in Peripheral Assemblies

Table 2.2

ANALYTICAL RESULTS FOR COMMON REACTIVITY HISTORIES




Burnable Poison in Reload Fuel




Shim or Peripheral Leakage in Cycle j

| Case* | Prescription for $\varepsilon$ |
|---|---|
| 1. (no burnable poison)** | $2(\theta\rho_0)^2 \dfrac{(n-1)}{3(n+1)^2}$ |
| 2. (constant $\rho$ in 1st Cycle) | $-(\theta\rho_0) \dfrac{(n-1)}{n(n+1)^2} + (\theta\rho_0)^2 \dfrac{(n-1)^2}{3n(n+1)^2}$ |
| 3. (zero $\rho$ at BOL) | $-(\theta\rho_0) \dfrac{(n-1)}{2n(n+1)} + (\theta\rho_0)^2 \dfrac{(n-1)^2}{6n(n+1)^2}$ |
| 4. (shim or leakage in Cycle j) | $\dfrac{(15.6-n)(6.8+n)R(\theta\rho_0)}{[(38n-25)+(2n-7)j]} + 2(\theta\rho_0)^2 \dfrac{(n-1)}{3(n+1)^2}$ |

$$\text{where } R = \frac{(\text{shim or leakage reactivity penalty, } \Delta\rho)}{\rho_0}$$

*Case number corresponds to those in the figure.

**Factor of 2 in prescription for $\varepsilon$ is an empirical normalization.

With the discharge burnup in hand, a front-end material balance readily yields the natural uranium requirement, and hence the uranium utilization, commonly expressed as megawatt days thermal (or electric) per short ton of $U_3O_8$ (or its inverse). The only complication to be aware of, as spelled out in reports by Kamal, is the choice of ground rules for comparing alternatives: equal burnup, efph, natural uranium, or U-235 commitment - each of which gives slightly different estimates of the savings.

The LRM, as outlined above, is used to evaluate alternatives according to the following generic procedure. First, the lattice design in question is burned over the range of interest using the LEOPARD program. Then the LEOPARD $\rho(\bar{B})$ output is least-squares curve fit to obtain its slope, A, and intercept $\rho_o$. These values in turn are used in the LRM algorithms to compute the steady state discharge burnup, $B_d$ MWD/MT - which relates directly to natural uranium usage.

As noted in the appropriate topical reports, the LRM gives results in good agreement with the more elaborate state-of-the-art results, in the relatively few instances where useful output (full-burn to steady state) have been published.

## 2.3 Comments and Conclusions

The linear reactivity model has proven equal to the task for which it was devised. It has been applied to all options suggested to us for evaluation. Variations on the basic model have been implemented to handle special cases - for example, where reactivity is not a sufficiently linear function of burnup.

Work is continuing in this area. A fuel-batch-coupling program is being developed to permit ready assessment of performance under non-steady-state conditions such as startup and transition cycles, and a second program is being developed to optimize assembly reload patterns.

## 2.4 References

1. J. A. Sefcik, M. J. Driscoll and D. D. Lanning, "Analysis of Strategies for Improving Uranium Utilization in Pressurized Water Reactors", MIT-EL-80-132 (January 1981).

2. W. T. Loh, M. J. Driscoll and D. D. Lanning, "The Use of Burnable Poison to Improve Uranium Utilization in PWRs", MIT-EL-82-014 (May 1982).

3. A. Kamal, "The Selective Use of Thorium and Heterogeneity in Uranium-Efficient Pressurized Water Reactors", Ph.D. Thesis, MIT Nuclear Engineering Department (August 1982).

4. M. J. Driscoll, Class Notes, MIT Subject 22.35 - Nuclear Fuel Management, Spring Term 1982.

# 3. APPLICATIONS

## 3.1 Introduction

In this section the major findings resulting from the project's efforts are recapitulated. Appendix A contains a complete list of all publications prepared by workers supported by, or associated with the project. In this latter category are several self-supported students who contributed their efforts to elucidate important issues bearing on the main-line subtasks. Appendix A also contains the abstracts of all reports/theses. The major findings reflect primarily the research reported in the topical reports/ doctoral theses by Sefcik [1], Loh [2], and Kamal [3]; other important specific contributions of central interest are in the theses by Malik (on axial power shaping) [4], and Lobo (on coastdown) [5].

## 3.2 Discussion

Table 3.1 summarizes the primary options which have been evaluated, and their projected yellowcake savings. Table 3.2 supplements this information with results applicable to the selective use of thorium and heterogeneity, excerpted from the recently published topical report by Kamal.

Not surprisingly, extended burnup is the most effective means to improve uranium utilization, although its efficacy is diminished somewhat by the concurrent move toward 18-month cycles, which (until ultra-high burnup is a reality) precludes increasing the number of staggered reload batches (from a current 3 to a potential 5). This illustrates an important point: while $U_3O_8$ consumption is a useful surrogate variable for fuel cycle or, better yet, system energy cost, it is really the explicit value of the latter which serves as the basis for decision-making. Even so, some demonstrably favorable options, such as routine pre-planned coastdown, have not been universally adopted, although there appears to be a trend in this direction. As we found, a very creditable $U_3O_8$ savings (on the order of 8%) can be

TABLE 3.1 POTENTIAL IMPROVEMENTS IN URANIUM UTILIZATION FOR PWRs ON A ONCE-THROUGH FUEL CYCLE

| | OPTION | $NAT_U$ SAVINGS (%) | COMMENTS |
|---|---|---|---|
| 1. | Extended Burnup and Increased Number of Batches | ∿15 | 5-batch core with discharge burnup of ∿55,000 MWD/MT; risk of premature fuel failure must be considered |
| 2. | Low-Leakage Fuel Management | ∿ 3 | Must cope with power peaking problem; if burnable poison is used, residual poison may negate savings |
| 3. | Axial Blankets | ∿ 2 | Aggravates axial power peaking; may require poison or enrichment zoning |
| 4. | Re-Optimizing Lattice Fuel-to-Moderator Ratio | 2-3 | For high burnup cores; depends on specifics of current core design |
| 5.(a) | Continuous Mechanical Spectral Shift | 10-15 | May not be practical from an engineering standpoint |
| 5.(b) | $D_2O$ Spectral Shift | 10-15 | $D_2O$ is expensive |
| 6. | Mid-Cycle Pin Pulling and Bundle Reconstruction | ∿10 | Potential thermal-hydraulic problems; reduces plant capacity factor |
| 7. | Routine Preplanned Coastdown | ∿ 7 | If coastdown to economic breakeven is considered instead of to the optimum, the uranium savings can be approximately doubled (as is the duration of coastdown) |

TABLE 3.2

Potential Uranium Savings for Selected PWR Fuel Management Strategies Emphasizing the Use of Thorium

| Strategy | Uranium Savings[a] | Comments |
|---|---|---|
| 1. Thorium Internal Blankets | (CB)[b] <0.5%<br>(HB)[c] ∿3% | The use of blanket assemblies having a different $V_F/V_M$ from driver assemblies may be problematic from a thermal-hydraulic standpoint. |
| 2. Spent Fuel Internal Blanket | (HB) 3% | In the steady-state this corresponds to adding one more reload batch in the core, in which case cycle length is shorter, other things being equal. |
| 3. Thorium Radial Blanket[d] | (CB) ∿ 1%<br>(HB) ∿ 6% | Power-peaking in the core interior may force less than optimal deployment. (See also comments on strategy #1.) |
| 4. Natural Uranium Radial Blanket[d] | (CB) ∿ 2%<br>(HB) ∿ 4% | Should be possible to get somewhat higher savings through blanket lattice optimization. |
| 5. Low-Leakage Fuel Management (using oldest fuel batch)[d] | (CB) ∿ 4%<br>(HB) ∿ 5% | Potential power peaking problems in core interior; burnable poison required. |
| 6. Spent Fuel Radial Blankets[d] | (CB) ∿ 9%<br>(HB) ∿ 9% | Best radial blanket material. Corresponds to adding one more reload batch to the core and using oldest batch on core periphery. |
| 7. Thorium Pins Uniformly Dispersed within Uranium Fuel Assemblies | (CB) negative<br>(HB) ∿ 2% | Selective use of thorium pins for power shaping within uranium assemblies should be considered, as has been proposed for BWRs. |

TABLE 3.2 (Cont'd.)

| Strategy | Uranium Savings[a] | Comments |
|---|---|---|
| 8. Reconstitution/ Reinsertion of Thorium Assemblies as Radial Blankets | (CB) ∿ 5%<br>(HB) ∿ 7% | Assumes exogenous source of reconstituted assemblies. Uranium utilization is fairly insensitive to pre-reconstitution burnup. |
| 9. Spectral Shift Control for 3-Batch All-Uranium Core[f] | (CB) ∿ 9% | Savings increase as number of reload batches is increased (at fixed fuel enrichment); mechanical or $H_2O/D_2O$ spectral shift is difficult to implement. |
| 10. Spectral Shift Control for Cores Containing Thorium[f] | (CB) ∿ 14% | Spectral shift control is difficult to implement in practice. Quoted savings are an upper limit. |
| 11. Small PWR Fuel Assemblies in Low-Leakage Cores (with oldest batch on periphery)[e] | (CB) ∿ 4%<br>(HB) ∿ 5% | Savings comparable to those in strategy #5. Reduced power peaking. |
| 12. Small Fuel Assemblies with Thorium Radial Blanket[e] | (CB) ∿ 5%<br>(HB) ∿ 8% | Reduced power peaking. |

[a] All savings for steady state once-through operation (no recycle).

[b] CB = Current Burnup PWR (3-batch core, discharge burnup ≃ 36 GWD/MT).

[c] HB = High Burnup PWR (5-batch core, discharge burnup ≃ 61 GWD/MT).

[d] Savings relative to Out-In/Scatter all-uranium reference cores.

[e] Compared to reference cores having regular PWR assemblies and Out-In/Scatter fuel management.

[f] Compared to all-uranium PWR at fixed $V_F/V_M$.

achieved by coastdown to the economic optimum duration - a value which can be doubled if coastdown to economic breakeven were carried out! Here again the necessity to focus on system energy cost (which includes the cost of replacement energy during reactor outages or shutdown) rather than merely fuel cycle cost is essential.

Most of the other easily-retrofit changes which were evaluated give much smaller savings - although they can accumulate if all are implemented. Here one must be careful not to assume that savings can be linearly combined: specific composite case histories must be analyzed.

There are other strategies, such as the use of annular fuel, which we found to be neutronically neutral in its effect (in that equivalent non-annular lattices can always be devised), but which may facilitate other desirable changes, such as increased burnup and an increased moderator-to-fuel ratio, which do augment uranium utilization.

Axial [4] and radial [3] blankets were given special attention. The former yielded modest but worthwhile improvements, while the re-use of spent fuel generally gave superior results compared to specially fabricated blanket assemblies of depleted uranium (or of thorium). Amelioration of the central power peaking increase which accompanies the replacement of end-of-assembly fuel pellets by depleted uranium blanket pellets was investigated. An optimum axial power profile was devised, and a practical 3-enrichment zone approximation developed. The use of annular pellets in the zone of higher enrichment between the large central zone and the blanket was found to give a power profile which held its shape well over the assembly's burnup lifetime. Since this configuration is no more elaborate than comparable zoning already implemented in BWRs, it does not appear that the central peaking problem is an inherent limitation on the use of axial blankets in PWRs.

The use of thorium in the once-through fuel cycle was also a major topic of inquiry [3] - one which also followed naturally from our earlier work on recycle-mode applications of this fertile material. Appendix B to this report summarizes the past decade of thorium-related work at MIT, prepared as a review paper at the behest of the US National Science Foundation. Here we need only note that thorium appears not to offer worthwhile advantages in PWRs without recycle. At most we found opportunities for but a few percent uranium savings, and then only under special circumstances, such as ultra-high burnup. In general, re-use of "spent" fuel is preferable.

A final major topic was the use of burnable poison, because of its role in facilitating the implementation of uranium-saving innovations, such as high burnup and low-leakage fuel management. We studied this option from a generic point of view, but with an obvious view to the use of gadolinium. It was found that the flatter power history which BP permits is sufficient to offset the penalty of residual poison reactivity at and beyond the end of a fuel batch's first cycle in core. Thus, while alternatives are still of interest, the use of burnable poison can be contemplated with equanimity.

## 3.3 Conclusions and Recommendations

All of the options examined in the MIT research effort were not pursued to the full extent which their promise would appear to justify. The most important bit of unfinished business lies in the area of beginning-of-life axial power shaping using enrichment zoning. BOL power peaking near the assembly midplane is the limiting condition in many instances - for example, it limits the benefits achieved through the use of axial blankets of natural or depleted uranium. It would be of considerable interest to see to what extent axial enrichment zoning could substitute for the use of burnable poison in high burnup, low-leakage core loadings. While full substitution is an unlikely prospect, because burnable poison also benefits the moderator temperature coefficient, a combined strategy in which both the B.P. and axial enrichment are zoned might pay dividends.

## 3.4 References


1. J. A. Sefcik, M. J. Driscoll and D. D. Lanning, "Analysis of Strategies for Improving Uranium Utilization in Pressurized Water Reactors", DOE/ET/34022-1, MIT-EL-80-032, January 1981.

2. W. T. Loh, M. J. Driscoll and D. D. Lanning, "The Use of Burnable Poison to Improve Uranium Utilization in PWRs", DOE/ET/34022-3, MIT-EL-82-014, May 1982.

3. A. Kamal, M. J. Driscoll and D. D. Lanning, "The Selective Use of Thorium and Heterogeneity in Uranium-Efficient Pressurized Water Reactors", DOE/ET/34022-4, MIT-EL-82-033, August 1982.

4. M. A. Malik, A. Kamal, M. J. Driscoll and D. D. Lanning, "Optimization of the Axial Power Shape in Pressurized Water Reactors", DOE/ET/34022-2, MIT-EL-81-037, November 1981.

5. L. Lobo, "Coastdown in Light Water Reactors as a Fuel Management Strategy", SM Thesis, MIT Nuclear Engineering Department, December 1980.

## 4. COMMENTARY

The work under the current contract was limited to the once-through fuel cycle. It now appears that policy, both worldwide and in the US, is moving back toward acceptance of recycle into thermal reactors. This being the case, it is worth asking whether the results of the subject effort are destined to be ephemeral. The following considerations suggest that they are not:

1. Improvements which impinge directly on the neutron economy, such as axial blankets and low-leakage fuel management, are advantageous whether or not recycle is contemplated.
2. So long as system energy cost is the objective function, high burnup is also optimum in the recycle mode; although uranium consumption is optimized (minimized) by low burnup in the recycle mode (at ~ 15,000 MWD/MT, or even lower if reprocessing losses are reduced).
3. High burnup and low leakage fuel management motivate the use of burnable poison in-core, regardless of the out-of-reactor fate of the fuel.
4. Coastdown is similarly advantageous, its economic advantage depending more on the plant capacity factor and the replacement energy cost than it does on fuel cycle cost considerations.
5. While the benefits of thorium use are greatly improved when recycle is permitted, so are its costs, and thus its economic prospects still do not appear to be bright enough to alter the conclusion that incentives for its development are not compelling for the foreseeable future.
6. In any event, the improvements in the once-through cycle, once

they are fully implemented, and the large uncertainties associated with back-end fuel cycle costs may well insure that it will be a long time before recycle in light water reactors is justified on purely economic grounds.

All things considered, the program to increase the burnup and neutronic efficiency of current LWR cores pays high dividends without regard to future developments in either the front or back ends of the fuel cycle.

There is one area, however, where the contemplation of LWR recycle does motivate change: tight-pitch plutonium-uranium lattices, which are not interesting if attention is restricted to the once-through mode, now merit investigation if uranium utilization remains a high priority objective. Otherwise their large fissile inventory, difficult retrofitability, and indifferent economic advantages work against their appeal.

# APPENDIX A

## BIBLIOGRAPHY AND ABSTRACTS

The compilation below lists publications of work associated in whole, or in part, with project efforts. Following the list, abstracts are appended for all of the major topical reports and theses.

### Topical Reports

1. J. A. Sefcik, M. J. Driscoll and D. D. Lanning, "Analysis of Strategies for Improving Uranium Utilization in Pressurized Water Reactors", DOE/ET/34022-1, MIT-EL-80-032, January 1981.

2. M. A. Malik, A. Kamal, M. J. Driscoll and D. D. Lanning, "Optimization of the Axial Power Shape in Pressurized Water Reactors", DOE/ET/34022-2, MIT-EL-81-037, November 1981.

3. W. T. Loh, M. J. Driscoll and D. D. Lanning, "The Use of Burnable Poison to Improve Uranium Utilization in PWRs", DOE/ET/34022-3, MIT-EL-82-014, May 1982.

4. A. Kamal, M. J. Driscoll and D. D. Lanning, "The Selective Use of Thorium and Heterogeneity in Uranium-Efficient Pressurized Water Reactors", DOE/ET/34022-4, MIT-EL-82-033, August 1982.

### Theses

1. Same as Topical Report No. 1: Ph.D. Thesis by J. A. Sefcik.

2. Same as Topical Report No. 2: S.M. Thesis by M. A. Malik.

3. Same as Topical Report No. 3: Ph.D. Thesis by W. T. Loh.

4. Same as Topical Report No. 4: Ph.D. Thesis by A. Kamal.

5. P. E. Cavoulacos, "Probabilistic Analysis of Nuclear Fuel Cycle Economics", SB and SM Theses, MIT Nuclear Engineering Department, May 1980.

6. D. C. Cambra, "Development of an Interactive Fuel Shuffling Program", SB Thesis, MIT Nuclear Engineering Department, June 1980.

7. M. A. de Fraiteur, "Correlating Economic Optimum Burnup for Light Water Reactors", SM Thesis, MIT Nuclear Engineering Department, December 1980.

8. L. Lobo, "Coastdown in Light Water Reactors as a Fuel Management Strategy", SM Thesis, MIT Nuclear Engineering Department, December 1980.

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1. J. A. Sefcik, M. J. Driscoll and D. D. Lanning, "Evaluation of Core Designs and Fuel Management Strategies for Improved Uranium Utilization", Trans. Am. Nucl. Soc., Vol. 35, November 1980.

2. M. J. Driscoll, J. A. Sefcik, A. Kamal and D. D. Lanning, "Using Annular Fuel to Save Uranium in PWRs", Trans. Am. Nucl. Soc., Vol. 38, June 1981.

3. A. Kamal, M. J. Driscoll and D. D. Lanning, "The Selective Use of Thorium in PWRs on the Once-Through Fuel Cycle", Trans. Am. Nucl. Soc., Vol. 39, November 1981.

4. M. J. Driscoll, "Methods for the Evaluation of Improved PWR Core Design", Proc. ANS Topical Meeting on Technical Bases for Nuclear Fuel Cycle Policy, Newport, R.I., September 1981.

5. P. E. Cavoulacos and M. J. Driscoll, "Probabilistic Analysis of Nuclear Fuel Cycle Economics", Proc. ANS Topical Meeting on Technical Bases for Nuclear Fuel Cycle Policy, Newport, R.I., September 1981.

# ANALYSIS OF STRATEGIES FOR IMPROVING URANIUM UTILIZATION IN PRESSURIZED WATER REACTORS

Joseph A. Sefcik, Michael J. Driscoll and David D. Lanning

ABSTRACT

Systematic procedures have been devised and applied to evaluate core design and fuel management strategies for improving uranium utilization in Pressurized Water Reactors operated on a once-through fuel cycle. A principal objective has been the evaluation of suggested improvements on a self-consistent basis, allowing for concurrent changes in dependent variables such as core leakage and batch power histories, which might otherwise obscure the sometimes subtle effects of interest. Two levels of evaluation have been devised: a simple but accurate analytic model based on the observed linear variations in assembly reactivity as a function of burnup; and a numerical approach, embodied in a computer program, which relaxes this assumption and combines it with empirical prescriptions for assembly (or batch) power as a function of reactivity, and core leakage as a function of peripheral assembly power. State-of-the-art physics methods, such as PDQ-7, were used to verify and supplement these techniques.

These methods have been applied to evaluate several suggested improvements: (1) axial blankets of low-enriched or depleted uranium, and of beryllium metal, (2) radial natural uranium blankets, (3) low-leakage radial fuel management, (4) high burnup fuels, (5) optimized H/U atom ratio, (6) annular fuel, and (7) mechanical spectral shift (i.e. variable fuel-to-moderator ratio) concepts such as those involving pin pulling and bundle reconstitution.

The potential savings in uranium requirements compared to current practice were found to be as follows: (1) $\sim$0-3%, (2) negative, (3) 2-3%; possibly 5%, (4) $\sim$15%, (5) 0-2.5%, (6) no inherent advantage, (7) $\sim$10%. Total savings should not be assumed to be additive; and thermal/hydraulic or mechanical design restrictions may preclude full realization of some of the potential improvements.

DOE/ET/34022-2

# OPTIMIZATION OF THE AXIAL POWER SHAPE IN PRESSURIZED WATER REACTORS

M. A. Malik, A. Kamal, M. J. Driscoll, D. D. Lanning

## ABSTRACT

Analytical and numerical methods have been applied to find the optimum axial power profile in a PWR with respect to uranium utilization. The preferred shape was found to have a large central region of uniform power density, with a roughly cosinusoidal profile near the ends of the assembly. Reactivity and fissile enrichment distributions which yield the optimum profile were determined, and a 3-region design was developed which gives essentially the same power profile as the continuously varying optimum composition.

State of the art computational methods, LEOPARD and PDQ-7, were used to evaluate the beginning-of-life and burnup history behavior of a series of three-zone assembly designs, all of which had a large central zone followed by a shorter region of higher enrichment, and with a still thinner blanket of depleted uranium fuel pellets at the outer periphery. It was found that if annular fuel pellets were used in the higher enrichment zone, a design was created which not only had the best uranium savings (2.8% more energy from the same amount of natural uranium, compared to a conventional, uniform, unblanketed design), but also had a power shape with a lower peak-to-average power ratio (by 16.5%) than the reference case, and which held its power shape very nearly constant over life. This contrasted with the designs without part length annular fuel, which tended to burn into an end-peaked power distribution, and with blanket-only designs, which had a poorer peak-to-average power ratio than the reference unblanketed case.

DOE/ET/34022-3

# THE USE OF BURNABLE POISON TO IMPROVE URANIUM UTILIZATION IN PWRs

by

W. T. Loh, M. J. Driscoll, D. D. Lanning

## ABSTRACT

A methodology based on the linear reactivity model of core behavior has been developed and employed to evaluate fuel management tactics for improving uranium utilization in Pressurized Water Reactors in a once-through fuel cycle mode on a consistent basis. A major focus has been on the benefit of using burnable poison in conjunction with low-leakage fuel management schemes. Key features in the methodology, such as power weighting of batch reactivity values and correlation of neutron leakage effects with peripheral assembly power, were verified against results generated using detailed state-of-the-art computer analyses. A relation between batch power fraction and batch reactivity was derived from a 1½-group diffusion theory model, and similarly validated. These prescriptions have been used in two ways: to develop analytical models which allow quick scoping calculations; and, programmed into a code, to facilitate more rigorous applications.

The methodology has been applied to evaluate fuel management schemes of contemporary interest, such as the use of burnable poison to shape the power history profile, the use of low-leakage fuel loading patterns, and extended cycle length/burnup, and combinations of these individual schemes.

It was found that shaping of the power history profile in a low-leakage assembly pattern by means of burnable poison, even after accounting for the anticipated residual poison reactivity penalty, has the potential of <u>increasing</u> PWR discharge burnup, and hence uranium utilization by roughly 1%. The overall improvement in uranium utilization for a low-leakage loading over that for the current out-in/scatter scheme, was about 3.6% for current cycle lengths (3-batch, discharge burnup ∿ 30,000 MWD/MT), and approximately 11.1% for extended cycle operation (3-batch, discharge burnup ∿ 50,000 MWD/MT).

# "THE SELECTIVE USE OF THORIUM AND HETEROGENEITY IN URANIUM-EFFICIENT PRESSURIZED WATER REACTORS"

by

Altamash Kamal

## ABSTRACT

Systematic procedures have been developed and applied to assess the uranium utilization potential of a broad range of options involving the selective use of thorium in Pressurized Water Reactors (PWRs) operating on the once-through cycle. The methods used rely on state-of-the-art physics methods coupled with batch-wise core depletion models based on the "group-and-one-half" theory.

The possible roles for thorium that were investigated are: as internal and radial blanket material, as thorium pins dispersed within uranium fuel assemblies, its use in PWRs operating on spectral shift control, and its reconstitution and reinsertion as radial blanket assemblies. The use of smaller assemblies in PWRs (for cores with and without thorium) was also investigated, as well as options which can be regarded as reasonable substitutes for employing thorium. The analyses were performed for both current (3-batch, discharge burnup $\sim$ 30 GWD/MT) and high-burnup (5-batch, discharge burnup $\sim$ 50 GWD/MT) PWR cores in their steady-state.

It was found that except for special circumstances (dry lattices and/or high burnup), the use of thorium does not save uranium compared to the conventional all-uranium PWRs. When savings are achieved (typically 1-3%, but as high as 9% in special circumstances), they can be, for the most part, equalled or exceeded by easier means: in particular, by the re-use of spent fuel. On the other hand, up to 15 or 20% thorium could be added into PWRs without significant losses in uranium utilization, if policies called for the build up of a U-233 inventory for later use in the recycle mode.

It was also found that, regardless of the deployment of thorium, the use of smaller fuel assemblies with the concurrent deployment of radial blankets is an effective uranium conservation strategy, with accompanying power-shaping advantages.

# COASTDOWN IN LIGHT WATER REACTORS
# AS A FUEL MANAGEMENT STRATEGY

by

Leancy Giovanni Lobo

Submitted to the Department of Nuclear Engineering on December 23, 1980 in partial fulfillment of the requirements for the degree of Master of Science in Nuclear Engineering

ABSTRACT

Improved uranium utilization by means of extended burnup via routine end-of-cycle coastdown has been analyzed, with a specific focus on pressurized water reactors. Both computer and simple analytic models have been developed to determine the optimal coastdown length. Coastdown has been compared with the use of higher fuel-enrichment to achieve comparable burnup values. Temperature and Power coastdown modes were analyzed and changes in the plant thermodynamic efficiency determined. Effects on fuel integrity due to coastdown were examined using a fuel reliability code (SPEAR). Finally the effects on coastdown duration of major parameters involved in characterizing reactor operation and the economic environment were examined.

It was found that natural uranium savings up to 7% could be achieved in a typical application by the use of routine pre-planned coastdown up to the economic optimum. If coastdown is carried out all the way up to the economic breakeven point yellowcake savings sum up to 16%. Coastdown is substantially more effective than increasing enrichment to extend cycle length without coastdown. Thermodynamic efficiency does not change appreciably during coastdown, a circumstance which greatly simplifies modeling. Coastdown

was found to have no statistically significant effect on predicted fuel failure rates. Finally, simple back-of-the envelope analytic models were found to give an excellent estimate of coastdown duration to both the optimum and breakeven points, and to correctly track the functional behavior induced by all major variables.

Thesis Supervisor: Michael J. Driscoll

Title: Professor of Nuclear Engineering

# PROBABILISTIC ANALYSIS OF NUCLEAR FUEL CYCLE ECONOMICS

by

PANAYOTIS ELIAS CAVOULACOS

Submitted to the Department of Nuclear Enginnering
on May 23, 1980 in partial fulfillment of the
requirements for the Degree of Bachelor of Science and
Master of Science in Nuclear Engineering

## ABSTRACT

The objective of the present work was to evaluate the fuel cycle cost component of nuclear-generated electricity under conditions of uncertainty, in which cost components are specified by a range and a probability distribution. In particular, the nuclear fuel cycle costs for the once-through and the recycle modes of a pressurized water reactor (PWR) were analyzed.

Simple nuclear core and nuclear fuel cycle economics models were used, modified to account for uncertainty in the input data. The uncertainty in each input quantity was represented by either a beta or a normal probability distribution function (pdf). For the beta pdf, it was assumed that the range and the mode were given and that the standard deviation was a sixth of the range. The same mode and standard deviation were used in the case of the normal pdf. A comprehensive data base was established after an extensive literature survey for all transaction unit costs in the fuel cycle, in 1980 dollars. A value for fissile plutonium was calculated based on the indifference value principle. Linear statistical uncertainty propagation was used to derive the mean and the variance of the nuclear electricity fuel cycle cost, assuming that unit costs and the capacity factor were random, independent variables. In addition, a Monte Carlo simulation was performed, assuming that the discount rate was a random variable as well, using the uniform probability transformation and standard computer subroutines to invert the beta and normal pdfs. A computer code, ENUF, was written in Multics Fortran in order to implement the

uncertainty analysis. The first four central moments were estimated from the empirical pdfs generated by ENUF using 500 trials.

It was found that the mean nuclear fuel cycle cost of the recycle mode PWR was slightly smaller, about 2 %, than that of the once-through PWR : an insignificant margin, in view of the $1\sigma$ value assigned to the nuclear fuel cycle cost difference, which is about 12 % of the means. Nuclear fuel cycle costs were found to be about 11 mills/kWhe in 1980 dollars. Hence, the purely economic advantages of plutonium recycle into thermal light water reactors of current design are marginal in the short term. It was found that linear uncertainty propagation and Monte Carlo simulation, with beta or normal input pdfs, both gave substantially the same results, within 3 %, while standard deviations were about 10 % of their respective means.

| | |
|---|---|
| Thesis Supervisor: | Michael J. Driscoll |
| Title: | Professor of Nuclear Engineering |
| Thesis Reader: | Elias P. Gyftopoulos |
| Title: | Ford Professor of Engineering |

# CORRELATING ECONOMIC OPTIMUM BURNUP FOR LIGHT WATER REACTORS

by MARC A. DE FRAITEUR

Submitted to the Department of Nuclear Engineering, December, 1980, in partial fulfillment of the requirements for the degree of Master of Science in Nuclear Engineering.

## Abstract

Large changes in the relative costs of Uranium, separative work, fabrication and reprocessing have taken place since the basic features of light water reactor fuel designs were established many years ago. Accordingly, continuing interest exists in the evaluation and optimization of fuel management schemes.

Furthermore, it is well known that the nuclear fuel cycle cost as a function of the irradiation time presents a rather flat minimum. It is the goal of the designer and the fuel manager to attain this optimum.

The purpose of this work has been to find a relation between the value of the optimum irradiation time (or the discharge burnup) and the various parameters which define the economic environment, such as the cost of ore, separative work and fabrication, the discount factor, etc.

This task is undertaken for both the once-through and the recycle fuel cycle modes. Optimization is considered with respect to fuel cycle cost, busbar cost or system energy cost.

Results are obtained which show the linear dependance of the optimum irradiation upon a composite economic index that accounts for the various parameters governing the financial environment and for the nuclear parameters. This linear relationship holds for both the once-through and the recycle fuel cycle modes. Two formulas are then derived that permit the computation of the optimum irradiation time for the busbar cost and the system energy cost, given the optimum for the fuel cycle cost.

Thesis Supervisor: Professor Michael J. Driscoll

Thesis Reader: Professor Richard K. Lester

# APPENDIX B

## COPY OF PAPER BY M. J. DRISCOLL

for the

## US-JAPAN JOINT SEMINAR ON THE THORIUM FUEL CYCLE NARA, JAPAN, OCTOBER 1982

The paper which follows was prepared for the NSF, which is sponsoring the US participants in this seminar. Inasmuch as a portion of the MIT work discussed in this paper was done under the subject contract, and the paper provides value perspective on this contract-related work, it has been reproduced here, the concern being that it would otherwise not be readily available to the reader.

# A REVIEW OF THORIUM FUEL CYCLE WORK AT MIT*

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Recent results are reviewed showing that in PWRs 10-20% thorium does not significantly penalize uranium utilization in the absence of recycle, and small uranium savings (1-2%) accrue in selected high-burnup applications. With recycle, tight-pitch $PuO_2$/$UO_2$ fueling competes favorably in many respects with $^{233}UO_2/ThO_2$. In the LMFBR, thorium can be used in the internal and external blankets of heterogeneous cores without significant penalties to overall performance, providing a substantial source of $^{233}U$ for thermal reactor fueling.

(thorium in PWRs; thorium in LMFBRs; using thorium to save uranium)

## I Introduction

Interest at MIT in thorium as a nuclear fuel dates back to the first days of the Nuclear Engineering Department in the mid-1950s and subsequent efforts [1-5] paralleled those of major thorium-related programs on the US national level. Recent work [6-22], which is the subject of this review paper, has focused on the use of thorium in light water reactors in both the recycle and once-through modes of fueling, and on its use in the internal and external blankets of fast reactors.

## II Thorium in PWRs in the Recycle Mode

As part of the NASAP/INFCE effort, work sponsored by the U.S. ERDA was carried out to evaluate the use of thorium in uniform pressurized water reactor lattices [6, 7]. Fuel-to-moderator ratio was a key variable in these studies, and the uranium utilization of thorium fueled cores was compared to that of lattices in which uranium-238 was the dominant fertile species.

An important result of these studies was the confirmation of earlier suggestions by Edlund [23] that tight-pitch plutonium-uranium fueled cores exhibited the potential for operation in a near-breeding regime, and the demonstration that such cores could compete quite favorably with thorium fueled light water reactors. Work by others has subsequently strengthened the technical foundation upon which the design of cores of this genre can be based [24]. Hence it is now less clear that thorium based fuel cycles are the preferred option for pressurized water reactors.

In this work, most of the calculations were carried out using the EPRI version of LEOPARD and its ENDF/B-IV-derived cross section library. The code was benchmarked against data reported on some 245 critical and subcritical assemblies culled from the literature - all we could find involving the fissile isotopes U-235, U-233 and Pu-239, mixed with U-238 or Th-232, and in both metal and oxide forms. The results (average absolute error in multiplication factor of ~ 0.012) were considered adequate for present purposes, but uncertainties arising from the lack of significant data on tight-pitch cores should be noted. For this work, the resonance integral representation for thorium developed by Steen was incorporated into LEOPARD to provide the normalization it requires for the realization of satisfactory accuracy [7]. Although LEOPARD is one of the older tools of the reactor physicist, a recent evaluation rates it highly in comparison with newer methods [25]. For whole-core calculations on the assembly level, PDQ-7 was employed.

The most recent round of thorium-related work began with exploratory studies [6, 8] comparing the uranium utilization of a number of fuel cycles, focusing on key variables such as the fuel-to-moderator ratio (volume ratio of heavy metal oxide to $H_2O$) and the rate of growth of the demand for nuclear energy, hence nuclear fuel. It was found that:

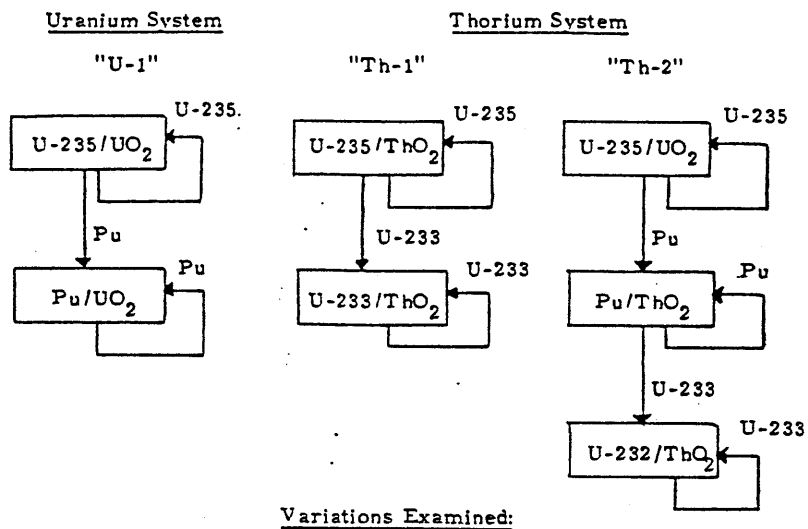
1. Under most circumstances, current lattice designs are close to optimum with respect to the amount of natural uranium (and/or separative work) needed to generate a given amount of energy; furthermore, as the rate of demand increases, the optimum moves closer to today's design point, and, if fuel cycle cost is the objective function, current designs are even more strongly favored. This observation is particularly true of "producer" (i.e. U-235/U-238 fueled) cores.

2. State-of-the-art computations confirmed the theoretical expectation [10] that neutronically equivalent oxide and metal-fueled lattices can be devised, and that their inherently different linear power capabilities (kw/ft) are likewise irrelevant to ore usage. Hence the use of metallic thorium fuel in LWRs should be decided on other grounds, such as LOCA behavior.

3. While the thorium fuel cycle (U-235/Th-232 recycling U-233 in Th-232) is superior with regard to $U_3O_8$ utilization, it exhibits higher fuel cycle costs, and becomes less attractive in both respects as the growth rate increases (as it must, particularly for a new fuel cycle during its market penetration phase). The bias in favor of thorium use inherent in comparing alternatives under zero growth/steady-state scenarios (or, equivalently, on a single-reactor basis) is often left unstated.

4. It is important for many reasons to decide whether U-233 is to be integrated with, or segregated from, the U-235 used to "fire-up" the fuel cycle; in later work we chose to recycle plutonium into thorium to produce U-233 for subsequent recycle into thorium. This obviates the need for commercial traffic in highly enriched uranium (which now appears to be gaining acceptance as a safeguards convention), postpones the need for reprocessing thorium-containing fuel, and leaves discharged U-235 free of U-233, and hence acceptable as feed to a separation plant for re-enrichment. Segregated recycle also permits separate optimization of the various lattice designs involved. It is also important to decide whether single or multiple pass recycle is intended - the former is hardly worthwhile for thorium systems.

*Work supported by the U.S. AEC/ERDA/DOE.




Variations Examined:

(a) once-through mode (no recycle).

(b) uranium recycle only and recycle of both U and Pu.

(c) 0-10% system growth rates (0% ≡ single reactor basis).

(d) both one-pass recycle and recycle to extinction.

(e) both segregated and integrated recycle of U-233/U-235.

Fig. 1 Coupled PWR systems evaluated.

5. This work also demonstrated the importance of applying practical ground rules for concept evaluation. In particular, using the overall system cost of electricity mills/kwhre (busbar plus replacement energy costs), as the primary objective function, was found to favor burnup as high as 60,000 MWD/MT, whereas optimization to minimize nuclear fuel cycle cost alone, or to minimize uranium consumption typically favor much lower burnup. For example, both uranium and thorium fueled reactor systems in the recycle mode show minima in their natural uranium requirements around 15,000 MWD/MT, and even lower values would result if reprocessing losses could be reduced.

6. Under representative economic conditions, the premium value of U-233 as a thermal reactor fuel was confirmed. Indifference values of U-233, fissile plutonium and hightly (93%) enriched U-235 were established in terms of the price of ore, \$/lb $U_3O_8$, and the cost of separative work, \$/Kg SWU (8):

$$C_{U\text{-}233} = 0.678\,C_{U_3O_8} + 0.318\,C_{SWU} - 13.72 \text{ \$/g}$$

$$C_{Pu_f} = 0.578\,C_{U_3O_8} + 0.178\,C_{SWU} - 13.90 \text{ \$/g}$$

$$C_{U\text{-}235} = 0.400\,C_{U_3O_8} + 0.236\,C_{SWU} \quad \text{\$/g}$$

These values include cost penalties for fabrication and reprocessing - aspects in which U-233 ranks unfavorably [9].

Figure 1 shows the reactor systems evaluated, Table 1 presents selected results from Ref. [6], and Table 2 compares the final reactors in the U-1 and Th-2 systems, as their fuel-to-moderator ratio is varied. As can be seen, tight pitch $Pu/UO_2$ cores can surpass (in large part due to U-238 and Pu-240 fast fissions) the neutronic performance of their $^{233}U/ThO_2$ counterparts for sufficiently large fuel-to-moderator ratios. While a high plutonium inventory is incurred (because the absorption resonance integral for Pu-239 is a factor of two smaller than that of U-233), this may not be prohibitive, considering that fast breeder reactors will apparently not be competing for this fuel in large numbers for many decades.

Some caveats are to be noted with regard to the results reported in this and the following section. Our focus was, for the most part, on fuel assembly designs and fuel management strategies which would be retrofittable into current PWR units. In particular, we did not re-examine in any detail the seed and blanket concept embodied in the Light Water Breeder Reactor, as employed in the recent core configuration tested in the Shippingport PWR. Furthermore, in Table 2 the inferior performance of the thorium system is attributable to the mediocre neutronics of the $Pu/ThO_2$ lattice, and not the $U\text{-}233/ThO_2$ fuel cycle per se.

## III Once-Through Mode of Operation

As the NASAP/INFCE programs progressed, it became clearer that commercial reprocessing might well be deferred longer than originally anticipated, and attention turned more to extended burnup on the once-through fuel cycle. The projected uranium savings from high burnup of on the order of 15% in the near term, and perhaps double that in the far term, clearly reduce the incentive for introducing thorium and recycling spent fuel - which, as already noted, achieve maximum $U_3O_8$ conservation at quite short burnups (~ 15,000 MWD/MT). The fact that overall system energy costs typically decrease monotonically with burnup also means that, in some respects, thorium fueled systems are being forced

Table 1

Important Results for 0 and 10%/Year

Growth Rates - Full Recycle

| Fuel Cycle | Fuel-to-Coolant Volume Ratio[1] | System Ore Usage ST $U_3O_8$/GW(e)yr[2] | % Savings Over Today's Once-Through PWR |
|---|---|---|---|
| **A. Zero Percent Per Year Growth Rate (i.e., Single Reactor Case)** | | | |
| 1. Uranium (Once-Through) | Present-day lattices. | 182.0 | - |
| 2. Fully enriched Uranium/Thorium (full recycle) | Present-day lattices. | 91.9 | 49.5 |
| 3. Uranium Cycle (full recycle) | Present-day lattices. | 111.2 | 38.9 |
| 4. Fully enriched Uranium/Thorium (full recycle) | Tight lattice (producer). Very tight lattice (consumer). | 77.5 | 57.4 |
| 5. Uranium Cycle (full recycle) | Tight pitch (producer). Present-day lattices (consumer). | 100.8 | 44.6 |
| **B. Ten Percent Per Year Growth Rate** | | | |
| 1. Uranium (Once-Through) | Present-day lattices. | 222.2 | - |
| 2. Fully enriched Uranium/Thorium (full recycle) | Present-day lattices. | 154.3 | 30.6 |
| 3. Uranium Cycle (full recycle) | Present-day lattices. | 152.8 | 31.2 |
| 4. Fully enriched Uranium/Thorium (full recycle) | Present-day lattices (producer). Tight lattices (consumer). | 148.6 | 33.1 |
| 5. Uranium Cycle (full recycle) | Present-day lattices. | 152.8 | 31.2 |

(1) Cycles 4 and 5 are optimized mixed $V_F/V_M$ systems - tight lattice refers to $V_F/V_M$ = 0.9161. Very tight lattice refers to $V_F/V_M$ = 1.497.

(2) Per GW(e)yr (rated) at 75% capacity factor, 0.2% tails.

Table 2

Core Characteristics as a Function of Fuel-to-Moderator Ratio

| F/M | Reload Enrichment w/o | | Conversion Ratio Cycle-Average | | System Ore Consumption ST $U_3O_8$/Gwe · yr | |
|---|---|---|---|---|---|---|
| | U-233/$ThO_2$ | Pu/$UO_2$ | U-233/$ThO_2$ | Pu/$UO_2$ | U-233/$ThO_2$ | Pu/$UO_2$ |
| 0.5 | 2.8 | 2.7 | 0.76 | 0.72 | 103 | 106 |
| 1.0 | 3.0 | 6.2 | 0.82 | 0.85 | 100 | 90 |
| 2.0 | 4.2 | 8.4 | 0.87 | 0.94 | 99 | 71 |
| 3.0 | 5.4 | 8.8 | 0.91 | 0.99 | 96 | 44 |

BASIS:

(a) 75% capacity factor, 0.2 w/o tails, 1% losses in reprocessing and in fabrication; successive recycle to extinction with worth-weighting for isotopic composition. On the same basis the once-through PWR would require 167 ST $U_3O_8$/Gwe · yr.

(b) Initial isotopic compositions:

"U-233": 91 w/o U-233, 8 w/o U-234, 1 w/o U-235.

"Pu": 54 w/o Pu-239, 26 w/o Pu-240, 14 w/o Pu-241, 6 w/o Pu-242

(c) System uranium consumption pertains to use of the subject reactors in complete systems, namely the thorium system U-235/$UO_2$: Pu/$ThO_2$: U-233-$ThO_2$ and the uranium system, U-235/$UO_2$: Pu/$UO_2$. All cores use 3-batch fuel management, discharge fuel at 33,000 MWD/MT and (except for the final core in each sequence) have F/M = 0.5. The system growth rate is zero.

to compete under increasingly unfavorable ground rules.

Thus an evaluation was undertaken to ascertain whether the selective use of thorium on the once-through fuel cycle might be attractive in PWRs [12]. Table 3 summarizes the potential improvements in uranium utilization available from the various fuel management schemes analyzed in this work. Note that the savings from a composite core, employing some combination of strategies, would, in general, be less than the algebraic sum of the savings from each individual innovation.

On the basis of these results, attention is called to the following points:

1. The introduction of thorium in PWRs on the once-through fuel cycle offers, for the most part, uranium savings which can be equaled (and frequently exceeded) by the deployment of options that are simpler to implement: re-use of "spent" fuel, in particular.

2. This conclusion must be tempered by the observation that in a recycle mode of operation (which is recommended and anticipated by most fuel cycle engineers), the premium U-233 fuel bred in the thorium blankets would be a valuable asset. While uranium savings from using thorium are small, it is also true that up to 15 or 20 percent thorium can be introduced into PWR cores without incurring a large penalty in uranium utilization. Thus if a policy decision were made to build up an inventory of U-233 as a prelude to future deployment of the thorium cycle in the recycle mode, this could be done if the resulting core designs met all licensing margins (an issue not addressed in the subject work, and an obvious priority area requiring attention).

3. Thorium pins strategically placed in uranium fuel assemblies (e.g. at assembly corners and next to water holes) deserve investigation in more detail. Such a scheme has the potential to locally improve power peaking, and may be particularly useful in low-leakage schemes. GE researchers have already shown this strategy to be advantageous in BWRs.

4. As the ultimate burnup capability of LWR fuel and the fuel management practices of utilities become better defined, the practicality of introducing thorium should be re-evaluated. In general, the attractiveness of thorium increases as the burnup and cycle-length are increased. Thus, if LWR burnups as high as 70 or 80 GWD/MT could ever be contemplated, and if cycle-lengths as long as 18 to 24 months gain favor, thorium might find a place in the LWR once-through cycle.

5. The smaller fuel assembly option, with the concurrent deployment of thorium radial blankets should be evaluated in more detail and other aspects related to its eventual deployment should be investigated: economics, thermal-hydraulics, effect on burnable poison requirements, effect on refueling down-time, etc.

All-in-all, the use of thorium in PWRs prior to recycle appears to hinge more on policy decisions than technological incentives or impediments.

## IV Fast Reactors and Thorium

In the area of fast reactor physics, work involving thorium was carried out in both the analytical and experimental areas. On the computational level, internal and external blankets of thorium were evaluated for use with conventional plutonium-uranium driver fuel, and to a lesser extent, with U-233/Th-232 cores [14]. It was found that thorium could be substituted for depleted uranium without substantially degrading overall performance. Table 4 summarizes some of the

Table 3

Potential Uranium Savings for Selected PWR Fuel Management Strategies Emphasizing the Use of Thorium in a Once-Through Fuel Cycle

| | Strategy | Uranium Savings[a] | Comments |
|---|---|---|---|
| 1. | Thorium Internal Blanket | (CB)[b] < 0.5%<br>(HB)[c] ~ 3% | The use of blanket assemblies having a different $V_F/V_M$ from driver assemblies may be problematic from a thermal-hydraulic standpoint. |
| 2. | Spent Fuel Internal Blanket | (HB) 3% | In the steady-state this corresponds to adding one more reload batch in the core, in which case cycle length is shorter, other things being equal. |
| 3. | Thorium Radial Blanket[d] | (CB) ~ 1%<br>(HB) ~ 6% | Power-peaking in the core interior may force less than optimal deployment. (See also comments on strategy #1.) |
| 4. | Natural Uranium Radial Blanket[d] | (CB) ~ 2%<br>(HB) ~ 4% | Should be possible to get somewhat higher savings through blanket lattice optimization. |
| 5. | Low-Leakage Fuel Management (using oldest fuel batch)[d] | (CB) ~ 4%<br>(HB) ~ 5% | Potential power-peaking problems in core interior; burnable poison required. |
| 6. | Spent Fuel Radial Blanket[d] | (CB) ~ 9%<br>(HB) ~ 9% | Best radial blanket material. Corresponds to adding one more reload batch to the core and using oldest batch on core periphery. |
| 7. | Thorium Pins Uniformly Dispersed Within Uranium Fuel Assemblies | (CB) negative<br>(HB) ~ 2% | Selective use of thorium pins for power shaping within uranium assemblies should be considered, as has been proposed for BWRs. |
| 8. | Reconstitution/Reinsertion of Thorium Assemblies as Radial Blanket | (CB) ~ 5%<br>(HB) ~ 7% | Assumes exogenous source of reconstituted assemblies. Uranium utilization is fairly insensitive to pre-reconstitution burnup. |
| 9. | Spectral Shift Control for 3-Batch All-Uranium Core[f] | (CB) ~ 9% | Savings increase as number of reload batches is increased (at fixed fuel enrichment); mechanical or $H_2O/D_2O$ spectral shift is difficult to implement. |
| 10. | Spectral Shift Control for Cores Containing Thorium[f] | (CB) ~ 14% | Spectral shift control is difficult to implement in practice. Quoted savings are an upper limit. |
| 11. | Small PWR Fuel Assemblies in Low-Leakage Cores (with oldest batch on periphery)[e] | (CB) ~ 4%<br>(HB) ~ 5% | Savings comparable to those in strategy #5. Reduced power peaking. |
| 12. | Small fuel Assemblies with Thorium Radial Blanket[e] | (CB) ~ 5%<br>(HB) ~ 8% | Reduced power peaking. |

[a]All savings for steady state once-through operation (no recycle).

[b]CB = Current Burnup PWR (3-batch core, discharge burnup ~30 GWD/MT).

[c]HB = High Burnup PWR (5-batch core, discharge burnup ~50 GWD/MT).

[d]Savings relative to Out-In/Scatter all-uranium reference cores.

[e]Compared to reference cores having regular PWR assemblies and Out-In/Scatter fuel management.

[f]Compared to all-uranium PWR at fixed $V_F/V_M$.

differences identified in this work. The results in this table apply to external blankets only; in Ref. [15] internal thorium blankets are shown to reduce the burnup reactivity swing at the expense of a slightly higher fissile inventory. The beneficial synergism of LWR and LMFBR fuel cycles trading plutonium and U-233 was also quantified; Fig. 2 shows the savings in terms of a reduction in FBR fuel cycle costs [14].

In the overall scoping studies [15], seven combinations of reactor design and fuel cycle were examined: homogeneous cores having $PuO_2/UO_2$ driver and blanket assemblies, the same core with thorium external (radial and axial) blankets, and the latter case but with $PuO_2/ThO_2$ driver assemblies in the core; and heterogeneous cores, including $PuO_2/UO_2$ driver fuel, first with depleted uranium internal and external blankets, then with three variations - thorium external blankets, thorium internal and external blankets, and finally a case in which BeO was added to the thorium internal blanket. All of the internal blankets were of the axial parfait type, but most of the system characteristics are also representative of the more common heterogeneous cores using radial internal blankets.

The use of internal thorium blankets was found to increase U-233 production by as much as 50% - a considerable benefit if crossed-progeny fuel cycles coupling fast and thermal reactors are contemplated.

Fast breeder cores using Th-232 in both the core and blankets were found to have such poor performance that one can scarcely recommend them over thorium cycle light water reactors optimized for breeding performance.

The experimental work carried out in the blanket test facility at the MIT Research Reactor, was aimed primarily at quantification of self-shielding phenomena at material interfaces. It was found that at a thorium/uranium interface, fertile capture rates are nearly doubled over those in the immediate interior because of the scattering source of unshielded neutrons incident on each of the dissimilar zones [16, 17, 18]. This may pose specific design problems because of the consequential more rapid fissile accumulation in interface pins.

Our most recent work has involved a core design concept and fuel management strategy designated "breed/burn", in which thorium internal blanket assemblies, after U-233 is bred in over several cycles, are shuffled into a zirconium hydride moderated radial blanket and/or central island [19]. Cores of this genre can reduce core plutonium inventory by as much as 30%, fuel cycle cost by 20-40%, total reprocessing requirements by 50%, and the transportation/reprocessing of plutonium-bearing assemblies by 60%. Thus the breed/burn system is a useful addition to the FBR designer's repertoire of variations which can be accommodated in the same core grid configuration.

## V Thorium Use in HTGRs and Fusion-Fission Hybrids

Much less work has been done at MIT on the use of thorium in high temperature gas-cooled reactors, except to confirm the well-known particular suitability of these systems for the U-233/Th-232 fuel cycle [20, 21]. However, work is now in the planning phase for a project having as a major subtask the evaluation of small HTGRs as a candidate system for meeting future energy needs. It is the current perception of the author that fueling with low enrichment uranium may be selected for the reference design, purely on pragmatic grounds.

Some interesting work has been done on the use of thorium as a blanket material for a fusion reactor used as a source of neutrons to breed fuel for fission reactors [22]. It was found that a molten salt concept offered the best overall combination of characteristics for this class of applications. While not explicitly investigated, many of the same considerations should apply to the blanket zones required by accelerator-driven spallation source fuel breeding devices.

## VI Comments and Conclusions

Technical disincentives and institutional barriers to the near term use of thorium in the light water reactor fuel cycle appear to be more imposing than perceived just a few years ago. In particular, when viewed on the basis of overall discounted system costs, especially during the initial growth phase, the thorium fuel cycle is more expensive than the uranium cycle until $U_3O_8$ prices substantially exceed 100 $/lb in today's dollars. In addition, the safeguards prejudice against commercial traffic in fully enriched uranium apparently rules out the most efficacious route to the introduction of thorium (via U-235/Th-232 fueling). (In the opinion of the author there appears to be a logical inconsistency in this prohibition, if plutonium and/or undenatured U-233 recycle is to be permitted.) In any event, the recycle of plutonium/uranium in tight pitch lattices offers breeding performance characteristics which are comparable. Given these circumstances, the current surplus of enrichment capacity, and the prospects of less expensive future technology (proponents project costs as low as 40 $/kgSWU for laser-isotope-separation) also foster perceptions that the all-uranium fuel cycle will remain competitive in the long run, as does the emphasis on ever higher burnup and longer cycles for LWRs.

Of the variations examined in the present work, the most attractive use of thorium appears to be in the internal and external radial blankets of heterogeneous LMFBR core designs. Thorium can be introduced in this manner without penalizing the breeding (or other) characteristics of the LMFBR to any significant extent, while creating a beneficial synergism between the LWR and LMFBR fuel cycles. Unfortunately, this scenario postpones the extensive use of thorium for many decades. Given a commitment to the eventual use of thorium it would appear that the introduction of thorium into the once-through LWR to the extent practicable, and Pu/Th-232 recycle in LWRs, should be encouraged as the quickest and least onerous approach to acquisition of a technological base upon which to build a more widespread thorium fuel cycle.

Table 4
Summary of Differences in System Physics Characteristics Between Uranium and Thorium Blanketed (Radial and Axial) LMFBRs

| System Characteristic | Beginning-of-Life Ratio Thorium System/Uranium System |
|---|---|
| Core fissile loading | 1.040 |
| Control requirements | 1.093 |
| Central core sodium void coefficient | 1.028 |
| Isothermal Doppler coefficient | 0.910 |
| Doppler power coefficient | 0.938 |
| Adiabatic power coefficient | 0.958 |
| Delayed neutron fraction, $\beta$ | 0.981 |
| Prompt neutron lifetime, $\Lambda$ | 0.889 |

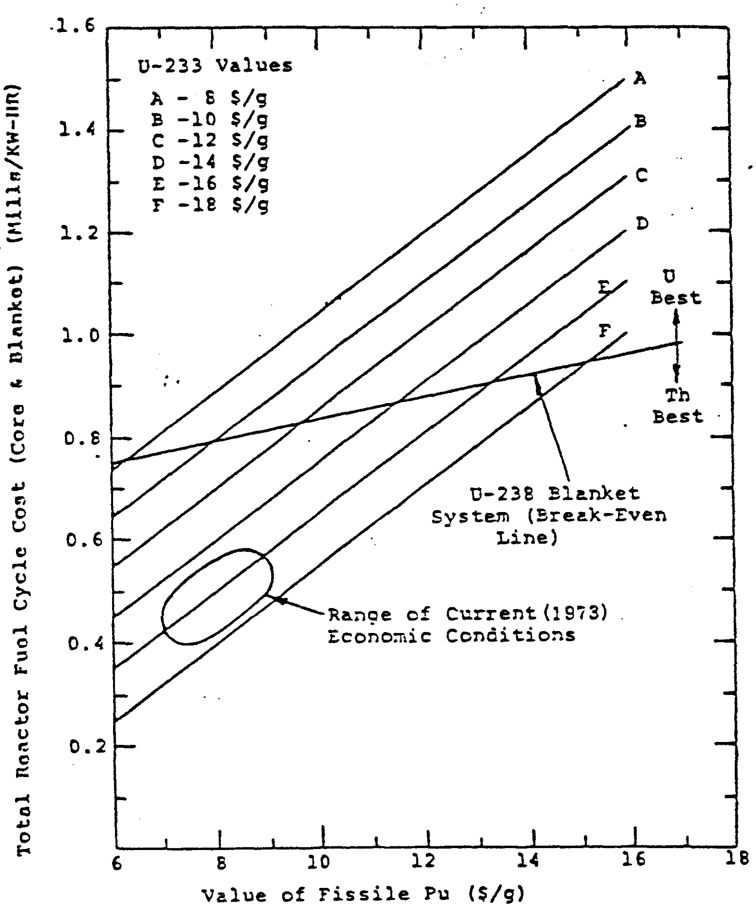



Fig. 2 Economic Comparison of Thorium and Uranium Blankets for a 1000 MWE LMFBR.

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## Acknowledgements

The author gratefully acknowledges the many essential contributions of his students and colleagues, whose work is referenced above. Particular thanks are due Prof. D. D. Lanning; and, for their thesis research on the thorium fuel cycle, Drs. Corrêa, Kamal, Wood, Garel and Abtahi. The past leadership and continuing encouragement of Institute Professor Emeritus Manson Benedict in MIT's fuel cycle efforts is especially appreciated.

# APPENDIX C

## FINAL PROGRESS REPORT

## ON A

## PRECURSOR PROJECT

The report which follows was not widely distributed, and then only under an internal document number. Hence it is reproduced here inasmuch as later work draws heavily on this background material.

DOCUMENT ID. NO:

LWRCD - 20

# EVALUATION OF IMPROVED LIGHT WATER REACTOR CORE DESIGNS

FINAL PROGRESS REPORT

September, 1979

Prepared by: Project Staff

MIT Nuclear Engineering Department/MIT Energy Laboratory
Massachusetts Institute of Technology
Cambridge, Massachusetts 02139

Date Published: October 31, 1979

Prepared for the United States Department of Energy

## 1. Introduction

This is the final, and final monthly, report under the subject contract. In addition to summarizing work done since the last monthly report (1), a brief synopsis of all tasks investigated during the course of the project will be presented.

Primary documentation for all results resides in the topical reports issued by the project, copies of which are available through NTIS. Supporting efforts in the form of otherwise unpublished MIT theses are also available; a specimen order form is appended.

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(1) LWRCD-19, Evaluation of Improved Light Water Reactor Core Designs, Monthly Progress Report, August 1979.

## 2. Synopsis of Overall Project Effort

The precursor contract which evolved into the present effort began in May 1976 as one component under a block grant to the MIT Energy Laboratory by DOE (then ERDA). The initial focus was on re-evaluation of the use of thorium in light water reactors (obviously in the recycle mode). From the beginning the emphasis was on uniform lattices, as opposed to the seed-and-blanket configuration of the LWBR under development by Naval Reactors. Attention was also concentrated on performance improvements achievable through variation of the fuel-to-moderator ratio. Another feature of this early effort was the use of standard and tight-pitch PWR core designs on the uranium/plutonium cycle as benchmarks against which the thorium designs were to be evaluated. Finally, the prospective use of tight pitch cores also required an examination of thermal-hydraulic constraints on core performance to define an allowable envelope bounding the physics and fuel management investigations.

As the contract progressed, the evolution of national and international policy, as reflected in the NASAP and INFCE efforts, led to a parallel re-orientation in the relative emphasis placed on the various subtasks within the MIT project. Accordingly, over the last 18 or 20 months, attention has been re-focused on the once-through uranium cycle, on a wide spectrum of measures to reduce ore consumption, and on concepts having retro-fit capability. Since it was established early-on that today's fuel-to-moderator ratios were near optimum for this purpose, there was also a corresponding diminution in the attention given to thermal-hydraulic concerns.

This historical perspective must be appreciated if one is to extract a sense of purpose from the otherwise diverse series of subtasks reflected in the project's published record. Appendix A contains reproductions of the abstracts from major topical reports and theses completed under project sponsorship, and Appendix B is a more complete bibliography of all publications associated with the project's efforts. Some of the listed students were supported in whole or in part by project funds, while others were self (or foreign-government) supported.

Although both general and specific conclusions are presented in the reports, theses and publications listed in Appendices A and B, it is of some use to review some of the major findings here, if for no other reason than to provide an overview. In retrospect, the following points now appear to have the greatest significance:

(1) The effort to improve upon ore utilization divides naturally into two almost diametrically opposite strategies depending on whether the recycle or once-through modes are to be employed:

in the former case tight-pitch cores and only moderate discharge burnups are to be preferred (Correa)*; while in the latter case, conventional lattice designs and long burnup are optimal (Fujita).

(2) Thus, thermal-hydraulic considerations (and the necessary plant redesign) are only of concern if advanced recycle cores are contemplated. For programmatic reasons alone this area would only be of long term interest--following extensive experience with recycle into more conventional cores. We can therefore separate discussion of these aspects from the more immediately interesting topic of improving ore utilization of the once-through fuel cycle.

(3) In the once-through mode the most productive approach to increasing ore utilization lies in increasing fuel burnup and the number of staggered batches in the core; additional improvements can be achieved by the careful accretion of savings from a large number of changes in core design, material composition and operating strategy.

(4) In particular, the use of routine coastdown (Driscoll et al.) and axial blankets (Kamal) give modest, but easily-realized improvements. (The verdict is still out on radial blankets for PWRs.)

(5) If and when recycle becomes of interest, tight-pitch plutonium-uranium cores appear to be able to compete successfully with U-233-thorium cores in terms of conversion ratio (Correa) and fuel cycle economics (Abbaspour). A weakly negative temperature/void coefficient may pose difficulties, however.

(6) As regards economics (Abbaspour), the objectives of ore conservation and lowest fuel cycle cost are generally compatible, particularly for the once-through cycle. Tight pitch cores show little overall short-term economic incentives (or disincentive) (Correa). In the long term the lower cost of ore should be taken into account: a start on characterization of this relationship has been made (Ghahramani).

(7) Although most of the effort was focused on individual reactor performance, some overall system's considerations were investigated (Garel). The main finding was that reactor inventory could play a dominant role in a rapidly growing nuclear economy. Recent experience, however, suggests that growth will be slow enough to mitigate this concern. Another

*Names in parentheses refer to principal authors of topical reports and theses dealing with the points at issue.

important finding (Abbaspour) was that system costs should be considered (instead of merely the fuel component), since refueling downtime and replacement costs shift the optimum burnup toward higher values for both the once-through and recycle modes. A related negative consideration is that the same driving force may motivate utilities to adopt 18 month refueling intervals instead of increasing the number of staggered fuel batches in the core. This would reduce the savings achieved by increasing fuel discharge burnup.

Concurrent with core physics and fuel management investigations of tight pitch cores in the recycle mode, an investigation of the thermal-hydraulics has also been conducted. Preliminary parametric studies indicated that decreasing the pitch affected the core pressure drop more than either MDNBR or fuel centerline temperature. WABCORE, a single channel thermal-hydraulics code, was developed for the purpose of investigating tight pitch lattices (Boyd). Various schemes for reducing P/D were analyzed, resulting in a better understanding of the interrelationships between geometric and thermal-hydraulic parameters. A design study on the thermal-hydraulics of ultra-tight pitch cores was performed, resulting in an optmized core design which could be substituted into an existing PWR with some modification (Griggs). A transient analysis of the optimized ultra-tight core was performed for a loss of plant power with failure to scram (Sigg). This analysis was performed with the loop code RELAP3B and gave insight into the impact of the new core design on the accident behavior of a tight-pitch plant. A LOCA analysis of the ultra-tight pitch core was considered for inclusion in the design study, but proved to be cost prohibitive. Work was done to find a simple relationship among basic core design parameters such as MDNBR, pressure drop, temperature rise, fuel rod diameter and core length which could then be used in assessing new core designs. Data was generated for this purpose by the WABCORE code, and curve fits to the simple expressions were performed.

In conclusion, the work conducted under this research project has developed information which supports in all respects the U.S. position evolved under the NASAP/INFCE programs with respect to the near and intermediate term potential for ore conservation in LWRs on the once-through fuel cycle. Moreover, in the even longer term, we have confirmed that contention by Edlund and others that tight-pitch $Pu/UO_2$ PWR cores can achieve conversion ratios which may allow these reactors to provide a competitive energy source far into the ore-scarce post-2000 era.

Some work necessarily remains undone. Refinement of various leads uncovered in the current studies are recommended, particularly with respect to the use of blankets on PWRs; and, in addition, a continued effort to effect cycle reactivity control without the use of control poison should be made, since therein lies one of the greatest potentials forfurther ore savings.

3.0

The section which appears here in the original draft has been deleted, since it was an incremental update on bimonthly progress. The subject in question, axial power shaping, has been dealt with in two publications which summarize the entirety of the MIT effort from a broader perspective:

A. Kamal, "The Effect of Axial Power Shaping on Ore Utilization in Pressurized Water Reactors", S.M. Thesis, MIT Nuclear Engineering Department, January 1980.

M. A. Malik, A. Kamal, M. J. Driscoll and D. D. Lanning, "Optimization of the Axial Power Shape in Pressurized Water Reactors", DOE/ET/34022-2, November 1981.

Similarly, the following two sections - a list of then-current project staff, and monthly budget - are now dated, and have also been omitted.

## Appendix C/A

## Abstracts of Major Topical Reports and Theses

The following compilation contains abstracts of the topical reports and major theses issued under project auspices. All topical reports are also available as theses, submitted by the principal author in each instance. Topical reports are available from NTIS; theses are available from MIT under arrangements as specified in the attached order form.

Not abstracted here are two theses still in the rough-draft stage:

A. Kamal
"The Effects of Axial Power Shaping on Ore Utilization in PWRs" (tent.)

SM Thesis, MIT Nuclear Engineering Dept., Nov. 1979 (est.)

D. Griggs
"Steady State and Transient Thermal-Hydraulic Design of an Ultra-Tight Pitch Pressurized Water Reactor Core" (tent.)

Nucl. Eng. Thesis, MIT Nuclear Eng. Dept., Nov. 1979 (est.)

These theses will be available shortly after submission under the same conditions as the other unpublished MIT theses listed here.

Postscript: Abstracts of these two documents have now been added.

COO-4570-9
MITNE-224
MIT-EL-79-001

Ali T. Abbaspour

THE FUEL CYCLE ECONOMICS OF IMPROVED URANIUM UTILIZATION IN LIGHT WATER REACTORS

ABSTRACT

A simple fuel cycle cost model has been formulated, tested satisfactorily (within better than 3% for a wide range of cases) using a more elaborate computer program, and applied to evaluate a variety of PWR fuel cycles and fuel management options, with an emphasis on issues pertinent to the NASAP/INFCE efforts. The uranium and thorium cycles were examined, lattice fuel-to-moderator and burnup were varied, and once-through and recycle modes were examined.

It was found that increasing core burnup was economically advantageous, particularly if busbar or total system cost is considered in lieu of fuel cycle cost only, for both once-through and recycle modes, so long as the number of staggered core batches is increased concurrently. When optimized under comparable ground rules, the once-through fuel cycle is competitive with the recycle option; differences are well within the rather large (± 20%) one sigma uncertainty estimated for the overall fuel cycle costs by propagating uncertainties in input data. Optimization on mills/kwhre and ore usage, tones/GWe,yr, are generally, but not universally, compatible criteria.

To the extent evaluated, the thorium fuel cycle was not found to be economically competitive. Cost-optimum thorium lattices were found to be drier than for current PWRs, while cost-optimum uranium lattices are essentially those in use today. The cost margin of zircaloy over stainless steel decreases as lattice pitch is decreased, to the point where steel clad could be useful in very dry cores where its superior properties might be advantageous.

Increasing the scarcity-related escalation rate of ore price, or the absolute cost of ore, does not alter any of the major conclusions although the prospects for thorium and recycle cores improve somewhat.

ABSTRACT

OUT-OF-REACTOR ASPECTS OF THORIUM UTILIZATION
IN LIGHT WATER REACTORS

by

Fereydoon Abtahi

Submitted to the Department of Nuclear Engineering
on July 12, 1977 in partial fulfillment
of the requirements for the degree of
Doctor of Philosophy

Out-of-reactor aspects of the Th/$^{233}$U fuel cycle in light water reactors are compared to the present uranium-plutonium system. The study shows that:

1. Although the extent of thorium reserves is not well-known, future demands should easily be satisfied.
2. Radiation due to daughters of U-232 in irradiated fuel is a major problem in the fabrication stage of the thorium fuel cycle.
3. Fluoride-induced corrosion during the storage of the nuclear waste and difficulties in waste glassification due to aluminum loading of the waste are potentially major problem areas in the Thorex Process.
4. The Th/$^{233}$U cycle has lower long-term (i.e., actinide) thermal decay heating, and lower airborne hazards than uranium-based fuels.
5. The Th/$^{233}$U cycle has a lower risk of diversion than the uranium fuel cycle because of penetrating radiation due to daughters of U-232 in irradiated fuel.
6. While the similarity of the Thorex and Purex processes as regards equipment requirements, and the apparent requirement that Pu-bearing fuel may also have to be fabricated by remote methods suggest that the same facilities can, in principle, be used for the two fuel cycles, one must consider the consequences of contaminating uranium fuels with U-232. The present $\leq$0.110 ppm U-232 (U-235-basis) limit set for diffusion plant feed will be violated if relatively small amounts of thorium-based fuels are mixed with uranium-based fuels. This would prevent recycle of spent uranium through enrichment facilities.

The assessment concludes by recommending that:

1. Better cross section data are needed to determine more accurately the U-232 buildup in irradiated fuel. Test irradiation of several fuel pins in a LWR would also be helpful.
2. The Thorex Process should be re-examined with respect to fluoride-induced corrosion during storage of the wastes

and difficulties in waste glassification due to aluminum loading.

3. Lattice redesign should be investigated to fully exploit thorium's advantages from an ore utilization aspect.

4. The 0.110 ppm U-232 (U-235 basis) criterion for diffusion plant feed needs to be re-evaluated; if dual purpose reprocessing and fabrication facilities cannot be used then the case for thorium utilization may be weakened, or, at the least, the introduction of thorium into the fuel cycle may prove more expensive than envisioned.

5. And finally, methods for quantification of the presumed safeguards and safety advantages of using thorium/U-233 must be developed to permit rational assessment of, the need for, and pace of its introduction into the fuel cycle.

Thesis Supervisor: David J. Rose
Professor of Nuclear Engineering

Thesis Supervisor: Michael J. Driscoll
Associate Professor of Nuclear Engineering

Thesis Reader: Manson Benedict
Institute Professor

# GENERATION OF FEW GROUP CROSS SECTIONS FOR THERMAL REACTORS USING FAST REACTOR METHODS

by

Habib Aminfar

Submitted to the Department of Nuclear Engineering on May 11, 1978 in partial fulfillment of the requirements for the degrees of Master of Science in Nuclear Engineering and Nuclear Engineer.

## ABSTRACT

Epithermal (>0.6 ev) cross sections prepared using thermal reactor (LEOPARD) and fast reactor (ANISN) preparation codes are compared for PWR lattices as a function of fuel-to-moderator ratio. The fast reactor approach is based on the shielding factor (f factor) method and a new equivalence theorem relating the background cross section per shielded nucleus, $\sigma_b$, in heterogeneous and homogeneous unit cells.

Systematic differences of 5% to 10% in fissile and fertile absorption cross sections are found above 5 Kev, and discrepancies as large as 30% in fissile cross sections are evident between 0.6 ev and 5 Kev. Although sensitivity studies of the effect on the overall multiplication factor indicate that the differences are self compensating to a considerable degree, reasons are developed for preferring the fast reactor methodology.

It is concluded that the fast reactor method can be adapted to serve both the thermal and fast applications given a modest amount of additional work.

Thesis Supervisor: Michael J. Driscoll
Title: Associate Professor of Nuclear Engineering

Thesis Reader: David D. Lanning
Title: Professor of Nuclear Engineering

# SPECIFIC INVENTORY AND ORE USAGE CORRELATIONS FOR PRESSURIZED WATER BY REACTORS

by

BAHMAN ATEFI

Submitted to the Departmentof Nuclear Engineering on May 16, 1977 in partial fulfillment of the requirements for the Degree of Nuclear Engineer

## ABSTRACT

Motivated by recent interest in the use of thorium, recycled and perhaps denatured U-233, and better ways of recycling plutonium in light water reactors, the present work develops methods having a strong foundation in experimental data for estimation of the fissile enrichment required to fuel lattices composed of combinations of fissile (U-235, U-233, plutonium) and fertile (U-238, Th-232) species.

Simple models are developed for the variation of epithermal-to-thermal reaction rate ratios with the fuel-to-moderator ratio of LWR lattices, confirming the observed linear relationship observed in experimental data for fertile capture ratios (e.g., $\rho_{28}$) and fissile fission ratios (e.g., $\delta_{25}$). These models are then used to adjust fertile resonance integrals and moderator downscattering cross sections in a two-group model to fit the observed data. The models also permit generation of results for fissile/fertile mixtures for which experimental data is lacking. Two group theory can then be used to generate clean critical lattice enrichments; the results also permit collapsing cross sections to one group for subsequent calculations.

Relations for the amount of beginning-of-life over-enrichment necessary to sustain a given amount of burnup are then developed and used to correct the clean critical results. These data in turn are used to generate fissile mass requirements for six fissile/fertile combinations as a function of fuel-to-moderator volume ratio for reactor systems undergoing steady exponential growth. The results are in qualitative agreement with state-of-the-art computer calculations for fuel-to-moderator volume ratios covering the range for which experimental data on $\rho_{28}$ and $\delta_{25}$ are available, but agreement grows progressively worse as the present model is extrapolated outside this range to higher fuel-to-moderator volume ratios. The discrepancy is attributed to oversimplification of the burnup model and suggestions are made for its improvement.

Thesis Supervisor: Michael J. Driscoll
Associate Professor of Nuclear Engineering

1

# THERMAL-HYDRAULIC ANALYSIS OF TIGHT LATTICE LIGHT WATER REACTORS

by

William Artis Boyd

## ABSTRACT

This thesis investigates the thermal-hydraulic sensitivity of the Main Yankee core with respect to changes in rod diameter, rod spacing, linear heat generation rate and axial heat flux shape using a specially developed steady-state, single channel code (WABCORE) for this purpose.

A review of the information available in the open literature on the effects of small rod spacings is presented to bring into perspective, the type of thermal-hydraulic changes that can be expected to occur with core geometry changes. This review will also give insight into the necessary thermal-hydraulic effects that must be considered and modeled by WABCORE.

WABCORE has been designed such that the large amount of computer runs needed for a sensitivity study of this kind is performed in a fast and orderly fashion with a minimum of computer processor (CPU) time. Results of a particular set of calculations can be automatically plotted. Verification of the results obtained by WABCORE were made by comparison with those obtained by COBRA IIIC/MIT for examples representative of PWR and BWR core designs. The agreement in most of the parameters is surprisingly good.

The physical models used in the code to represent the phenomena in the fuel pin, and its associated coolant channel are discussed in detail. An analytical approach is used in WABCORE to represent the axial flux shape. This representation allows a very elegant solution of the axial DNBR shape correction factor. Finally, the limitations of the correlations used in the code are summarized to indicate where further improvement should be made.

Results of the sensitivity study indicate that the linear heat generation rate of the Main Yankee core could be increased from 5.7 KW/ft to 10.0 KW/ft using a flat heat flux profile while maintaining the initial total mass flow, core cross-sectional area, and total power. This increase is achieved by decreasing the rod length and increasing the

rod diameter over the initial values. The minimum allowable DNBR for steady-state operation is reached before centerline temperature becomes unsafely high for the suggested linear heat generation rate. Other results of the sensitivity study indicate that

- the core pressure loss is independent of the axial heat flux profile
- the axial distribution of the rod centerline temperature and DNBR closely follow the axial heat flux profile.

Finally, optimized regions for steady-state operation at linear heat generation rates greater than the initial of 5.7 KW/ft are outlined graphically.

Thesis Supervisor: Professor Lothar Wolf
Title: Associate Professor of Nuclear Engineering

MITNE-227
MIT-EL-79-022

# AN EVALUATION OF TIGHT-PITCH PWR CORES

Francisco Corrêa

## ABSTRACT

The impact of tight pitch cores on the consumption of natural uranium ore has been evaluated for two systems of coupled PWR's namely one particular type of thorium system--U-235/$UO_2$: Pu/$ThO_2$: U-233/$ThO_2$--and the conventional recycle-mode uranium system-- U-235/$UO_2$: Pu/$UO_2$. The basic parameter varied was the fuel-to-moderator volume ratio (F/M) of the (uniform) lattice for the last core in each sequence.

Although methods and data verification in the range of present interest, 0.5 (current lattices) < F/M < 4.0 are limited by the scarcity of experiments with F/M > 1.0, the EPRI-LEOPARD and LASER programs used for the thorium and uranium calculations, respectively, were successfully benchmarked against several of the more pertinent experiments.

It was found that by increasing F/M to ∿3 the uranium ore usage for the uranium system can be decreased by as much as 60% compared to the same system with conventional recycle (at F/M = 0.5). Equivalent savings for the thorium system of the type examined here are much smaller (∿10%) because of the poor performance of the intermediate Pu/$ThO_2$ core--which is not substantially improved by increasing F/M. Although fuel cycle costs (calculated at the indifference value of bred fissile species) are rather insensitive to the characteristics of the tight pitch cores, system energy production costs do not favor the low discharge burnups which might otherwise allow even greater ore savings (∿80%).

Temperature and void coefficients of reactivity for the tight pitch cores were calculated to be negative. Means for implementing tight lattice use were investigated, such as the use of stainless steel clad in place of zircaloy; and alternatives achieving the same objective were briefly examined, such as the use of $D_2O/H_2O$ mixtures as coolant. Major items identified requiring further work are system redesign to accommodate higher core pressure drop, and transient and accident thermal-hydraulics.

COO-4570-4
MITNE=215
MIT-EL-78-017

# DESIGN AND FUEL MANAGEMENT OF PWR CORES TO OPTIMIZE THE ONCE-THROUGH FUEL CYCLE

by

Edward K. Fujita

## ABSTRACT

The once-through fuel cycle has been analyzed to see if there are substantial prospects for improved uranium ore utilization in current light water reactors, with a specific focus on pressurized water reactors. The types of changes which have been examined are: (1) re-optimization of fuel pin diameter and lattice pitch, (2) Axial power shaping by enrichment gradation in fresh fuel, (3) Use of 6-batch cores with semi-annual refueling, (4) Use of 6-batch cores with annual refueling, hence greater extended (~doubled) burnup, (5) Use of radial reflector assemblies, (6) Use of internally heterogeneous cores (simple seed/blanket configurations), (7) Use of power/temperature coastdown at the end of life to extend burnup, (8) Use of metal or diluted oxide fuel, (9) Use of thorium, and (10) Use of isotopically separated low $\sigma_a$ cladding material.

State-of-the-art LWR computational methods, LEOPARD/PDQ-7/FLARE-G, were used to investigate these modifications. The most effective way found to improve uranium ore utilization is to increase the discharge burnup. Ore savings on the order of 20% can be realized if greatly extended burnup (~ double that of current practice) is combined with an increase in the number of batches in the core from 3 to 6. The major conclusion of this study is that cumulative reductions in ore usage of on the order of 30% are foreseeable relative to a current PWR operating on the once-through fuel cycle, which is comparable to that expected for the same cores operated in the recycle mode.

# AN ANALYSIS OF PROSPECTIVE NUCLEAR FUEL CYCLE ECONOMICS

by

Kamran Ghahramani

Submitted to the Department of Nuclear Engineering on September 7, 1978, in partial fulfillment of the requirements for the degrees of Nuclear Engineer, and Master of Science in Mechanical Engineering.

## ABSTRACT

Simple process design models have been developed for unit costs in the more important steps in the nuclear fuel cycle: ore supply, enrichment, fabrication, MOX fabrication, reprocessing and waste disposal. These models were then combined into a simple computer code (CONFUSE) which is capable of performing nuclear fuel cycle cost calculations as a function of system size and economic factors for self-contained systems on a self-consistent basis.

A model of the resource/recovery/cost relationship for $U_3O_8$ ore has been developed using cost-of-processing and resource vs. grade submodels. It was found that the scarcity-related escalation in the future purchase price of $U_3O_8$ should increase at a rate approximately two-thirds of the average ore demand growth rate. The model also shows that the relative $U_3O_8$ cost varies as the 2/3 power of the yearly reactor system ore usage, Tons $U_3O_8$/MWe yr. The results support a uranium cost doubling time of about 8 years in then-current dollars for a 6% /yr rate of growth in demand and a 6%/yr inflation rate, or 14 years in constant dollars.

The computer code CONFUSE was used to perform demonstration applications, including sensitivity analyses and examation of the question of economy of scale in a self-contained nuclear economy (excluding ore production) supported by a fixed number of manufacturing facilities which vary in size (rather than in number). Enrichment was found to be the most important step, and benefits most from the economy of scale in both recycling and throwaway mode fuel cycles. A modest economy of scale was found: fuel cycle cost, mills/KWher, was found to vary as system MWe to the -0.14 power. Thus for a small system comprised of 10 or so reactors a fuel cycle saving of as much as 20% is achievable if access to facilities of a larger system, of say 50 reactors, were available. Attendant added costs, such as transporation and supply assurance (stockpiling, for example) would have to be debited against that potential saving.

Thesis Supervisor: Michael J. Driscoll
Title: Associate Professor of Nuclear Engineering

ANALYSIS OF ATWS IN A

TIGHT-PITCHED THORIUM FUELED PWR

by

DANIEL ROBERT SIGG

Submitted to the Department of Nuclear Engineering on 15 September 1978 in partial fulfillment of the requirements for the Degree of Master of Science

ABSTRACT

After defining the design of a thorium-fueled tight-pitched core which is inserted into a conventional nuclear steam supply system (NSSS), a transient thermal-hydraulic analysis consisting of an investigation into the effects of an ATWS on the new core is conducted. The particular ATWS studied is a loss of station power since it establishes the minimum DNBR reached for the entire class of ATWS incidents. The objective of the study is to determine, using the NRC established criteria listed in NUREG-0460, whether the new designed core can pass licensibility requirements by successfully mitigating the consequences of the incident. The tool used to conduct the investigation is the reactor system transient code RELAP3-B.

During the investigation of the transient, numerical instabilities in the code developed as it analyzed the ATWS transient which necessitated the use of much smaller time steps. This unforeseen change resulted in the exhaustion of support funds before the entire necessary transient period had been observed. As a result, the data compiled is inadequate to effectively meet all objectives of the study.

The following results are drawn from the analysis: (1) a minimum DNBR for the new core cannot be established due to the limited period of investigation, (2) the U-233/$ThO_2$ core power drops slower than the conventional core, (3) fuel temperatures drop as the incident progresses, (4) the behavior of the system parameters for the new core closely match those of the conventional core, and (5) the addition of boron into the core via the CVCS charging pumps would be necessary to bring the system to a hot shutdown condition. The conclusions

projected from the limited analysis are that the new design would meet NRC criteria for fuel temperature, clad temperature, and peak RCS pressure. However the minimum DNBR must be established and due to the slower power drop of the new core, the ATWS incident responsible for the peak reactor coolant system pressure should be investigated.

Thesis Supervisor: Professor Lothar Wolf

Title: Associate Professor of Nuclear Engineering

# THE EFFECT OF AXIAL POWER SHAPING ON ORE UTILIZATION IN PRESSURIZED WATER REACTORS

by

ALTAMASH KAMAL

Submitted to the Department of Nuclear Engineering on January 18, 1980, in partial fulfillment of the requirements for the degree of Master of Science.

## ABSTRACT

Axial power shaping in Pressurized Water Reactors has been analyzed to determine prospects for improved Uranium utilization. Emphasis has been placed on fuel assembly design modifications which would be relatively easy to retrofit. The modifications examined are: (1) Use of short axial blankets of natural uranium, (2) Use of annular fuel, and (3) Use of improved structural materials in the assembly and core end-zones.

State-of-the-art Light Water Reactor computational methods (LEOPARD and PDQ-7) have been used to investigate these modifications. A linear model for reactivity as a function of burnup has been developed and used to determine relative ore usage under a variety of constraints (constant ore requirement per batch, or U-235 loading per batch, or burnup, or efph). The main conclusions of this study are: (1) The use of improved structural materials in core and assembly end-zones can result in ore savings of about 5%, with no adverse effects, (2) The use of short axial blankets of natural uranium can result in ore savings of about 4%, if the slightly higher power peaking factors can be tolerated, (3) The use of improved structural materials negates any further advantages from the additional use of axial blankets, and (4) The use of annular fuel over at least part of the core length has potential advantages and deserves a more detailed analysis.

Thesis Supervisor: Michael J. Driscoll
Title: Associate Professor of Nuclear Engineering

# STEADY-STATE AND TRANSIENT THERMAL-HYDRAULIC DESIGN OF AN ULTRA-TIGHT PITCH PRESSURIZED WATER REACTOR CORE

by

DAN P. GRIGGS

Submitted to the Department of Nuclear Engineering on Feburary 1, 1980 in partial fulfillment of the requirements for the Degree of Nuclear Engineer and the Degree of Master of Science in Nuclear Engineering.

## ABSTRACT

Thermal-hydraulic design studies of an ultra-tight pitch (closely packed array of fuel rods) core for a Pressurized Water Reactor (PWR) were made. Appropriate design criteria were established such that an ultra-tight pitch core having U-233/$ThO_2$ fuel could be substituted into an existing reference $UO_2$-fueled PWR with no change in power level and minimal changes outside of the core. Hexagonal fuel assemblies with 217 wire-wrapped rods were selected as the basis for the core design. A methodology was devised to select optimized values of the design parameters. The COBRA IIIC/MIT code was validated for tight pitch lattices and steady-state subchannel analyses were performed. Approximate blowdown calculations were also made.

Ultra-tight pitch cores appear to be practical from a thermal-hydraulics point of view, if carefully designed. The selected optimum core required a higher average mass flux and pressure drop than the reference core. Peak fuel and clad temperatures during steady-state and blowdown were comparable for reference and ultra-tight pitch cores. Stainless steel was recommended as a cladding material.

Thesis Supervisor: Dr. David D. Lanning
Title: Professor of Nuclear Engineering

## Appendix C/B

## Bibliography of Publications

The compilation below lists, under several categories, publications of work associated in whole, or in part, with project efforts.

A total of some 22 progress reports, initially on a quarterly, and later on a monthly basis were also issued. However, no formal compilation of these documents has been published subsequent to their issue-by-issue distribution. Moreover, it is recommended that topical reports be relied upon as the final word on the status of all findings.

### 1. Major Topical Reports

(1) Salehi, A. A., M.J. Driscoll, and O.L. Deutsch, "Resonance Region Neutronics of Unit Cells in Fast and Thermal Reactors," COO-2250-26, MITNE-200, May 1977.

(2) Garel, K.C. and M.J. Driscoll, "Fuel Cycle Optimization of Thorium and Uranium Fueled PWR Systems," MIT-EL-77-108, MITNE-204, October 1977.

(3) Fujita, E.K., M.J. Driscoll and D.D. Lanning, "Design and Fuel Management of PWR Cores to Optimize the Once-Through Fuel Cycle," MIT-EL-78-017, COO-4570-4, MITNE-214, August 1978.

(4) Abbaspour, A.T. and M.J. Driscoll, "The Fuel Cycle Economics of Improved Uranium Utilization in Light Water Reactors," MIT-EL-79-001, COO-4570-9, MITNE-224, January 1979.

(5) Correa, F., M.J. Driscoll and D.D. Lanning, "An Evaluation of Tight-Pitch PWR Cores," MIT-EL-79-022, COO-4570-10, MITNE-227, August 1979.

### 2. Doctoral Theses

(1) Salehi, A.A., "Resonance Region Neutronics of Unit Cells in Fast and Thermal Reactors," May 1977.

(2) Abtahi, F., "Out-of-Reactor Aspects of Thorium Utilization in Light Water Reactors," July 1977.

(3) Garel, K.C., "Fuel Cycle Optimization of Thorium- and Uranium-Fueled PWR Systems," October 1977.

(4) Fujita, E., "Design and Fuel Management of PWR Cores to Optimize the Once-Through Fuel Cycle," August 1978.

(5) Correa, F., "An Evaluation of Tight-Pitch PWR Cores," August 1979.

3. SM and Nucl. Eng. Theses

(1) Atefi, B. "Specific Inventory and Ore Usage Correlations for Pressurized Water Reactors," May 1977.

(2) Boyd, W.A. "Thermal-Hydraulic Analysis of Tight-Lattice Light Water Reactors," May 1977.

(3) Chen, R., "A Simple Model for Analysis of LWR Fuel Cycle Economics," May 1977.

(4) Beard, C.L., "An Improved Long-Range Fuel Management Program," May 1978.

(5) Ghahramani, K., "An Analysis of Prospective Nuclear Fuel Cycle Economics," August 1978.

(6) Sigg, D.R. "Analysis of ATWS in a Tight-Pitched Thorium Fueled PWR," September 1978.

(7) Abbaspour, A.T.m "The Fuel Cycle Economics of Improved Uranium Utilization in Light Water Reactors," January 1979.

(8) Kamal, A., "The Effect of Axial Power Shaping on Ore Utilization in PWRs" (tent.), expected November 1979.

(9) Griggs, D., "Steady State and Transient Thermal-Hydraulic Design of an Ultra-Tight Pitch Pressurized Water Reactor," (tent.), est. November 1979.

4. Other Publications

(1) Aminfar, H., M.J. Driscoll and A.A. Salehi, "Use of Fast Reactor Methods to Generate Few Group Cross Sections for Thermal Reactors," Proceedings ANS Topical Meeting, Advances in Reactor Physics, Gatlinburg, Tennessee, April 1978.

(2) Ghahramani, K. and M.J. Driscoll, "Ore Price Escalation in Fuel Cycle Economic Analyses," Trans. Am. Nuc. Soc. 28, 383, June 1978.

(3) Driscoll, M.J., E.K. Fujita, and D.D. Lanning, "Improvement of PWR's on the Once-Through Fuel Cycle" (invited paper), Trans. Am. Nuc. Soc. 30, Nov. 1978.

(4) Driscoll, M.J., et al., "Routine Coastdown in LWRs as an Ore Conservation Measure," Trans. Am. Nuc. Soc. (in press), November, 1979.

(5) Driscoll, M.J. and F. Correa, "The Reactor Physics of Tight-Pitch PWR Cores," (invited paper) Trans. Am. Nuc. Soc. (in press), November 1979.