AN ABSTRACT OF THE THESIS OF

Charles P. Kliewer for the degree of Master of Science in Nuclear Engineering presented on December 19, 1986.

TITLE : A Feasibilty Analysis of High Conversion Ratio Pressurized Water Reactor Designs

Abstract approved: Redacted for Privacy
Samim Anghaie

A significant amount of interest has been aroused recently concerning the advancement of the current pressurized water reactor core designs with a special emphasis towards improving the conversion characteristics of these reactors. Most reports have been divided into two camps; those that deal with the neutronic aspects and those that deal with the thermohydraulic concerns. Seldom do these two areas get combined for purposes of evaluating a new design.

In this effort, the author takes a pragmatic approach to this area in so far as looking into ways of incorporating this advanced technology into

current operational power plants. In so doing the Trojan Nuclear Power Plant was selected to serve as the reference design plant. This was done since it is a Westinghouse designed reactor, as are a large portion of the PWR's in the United States, and because it has one of the largest thermal power ratings in the nation as well.

Both neutronic and thermohydraulic aspects are examined as well as an alternative fuel concept. In order to carry out the analysis computer codes COBRA-IV and LEOPARD were employed on a CYBER 170/710 mainframe computer. COBRA-IV was used to obtain results relating to the associated pressure loss and core temperature characteristics while LEOPARD was used for the neutronic aspects. A parametric study was initiated using fuel enrichment and pitch as the variables that would be systematically changed. As an additional factor to assure cross compatibility, the fuel rod diameter was held to a constant value throughout this analysis.

Results of this research strongly indicate that current operational power plants can be effectively altered to become converters or low

level breeders with only configurational changes in the core itself and no major equipment changes. Hence the author concludes that this concept is both feasible and readily attainable with the current level of technology.

An additional benefit that would be realized under the adoption of this design would be the marked improvement in the utilization of uranium ore which ultimately becomes fuel. This would directly result in the extension of the power generation capability associated with nuclear power well into the next century.

A FEASIBILTY ANALYSIS OF HIGH CONVERSION RATIO PRESSURIZED WATER REACTOR DESIGNS

by

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A THESIS
submitted to
Oregon State University

in partial fulfillment of
the requirements for the
degree of

Master of Science

Completed December 19, 1986

Commencement June 1987

APPROVED:

Redacted for Privacy

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Date thesis is presented on December 19, 1986

Typed for Charles P. Kliewer by Charles P. Kliewer

ACKNOWLEDGEMENT

I wish to take this moment to offer my sincere appreciation to a number of people that have played a significant role in this endeavor.

First I must give my warmest thanks to my parents and immediate family for suppling endless quantities of encouragement, support, and the occasional inspirational "kick in the pants" to see this work to completion.

Second I offer my gratitude to the U.S. NAVY for allowing me the time necessary to pursue this project.

Last but certainly not least, I would like to thank my friends and professors for all of their collective support and guidance.

Table of Contents

LIST OF FIGURES

LIST OF TABLES

A FEASIBILTY ANALYSIS OF HIGH CONVERSION RATIO PRESSURIZED WATER REACTOR DESIGNS

1. INTRODUCTION

The future of nuclear power seems to be solid and well assured, or so the average person would think given the content of the pro-nuclear publicity of current circulation. This viewpoint may be accurate in some regards, but in some areas it is dangerously naive. This false sense of security contains enough power to severely cripple the future of the nuclear industry if it is not exposed very soon. One of the more well concealed of these little time-bombs is the notion that we have a near inexhaustable supply of uranium fuel for our reactors.

This supposition would be accurate if we were utilizing a "Breeder" economy, but since we have adopted a "Burner" mentality this concept is simply not true.

Some would argue that nuclear power is only an interim solution until we develop Fusion power into an economical and commercially viable entity. To date however, only three suitable and largely

undeveloped energy sources are of any consequence, which are; (i) Solar Energy, (ii) Fission Breeder Energy, and (iii) Fusion Energy [1]. Given the current status on the development of both Fusion and Solar energy technologies, it seems abundantly clear that a decisive and determined effort to advance Fission Breeder technology should be vigorously pursued without any further delay!

Perhaps I should digress for a moment or so. Currently in the United States there are some 100 commercially operating nuclear power plants and 36 in various stages of completion. This represents a net energy resource of more than 118.9 GWe [2] or in other words, better than 15% of our entire nations energy needs.

In the world of fission reactors there are three basic varieties; Burners, Breeders, and Converters. The Burners are plants that in some aspects operate like cars in that they consume, or rather "burn", fuel to deliver a useable output. The Converter will also consume it's fuel as well, but it goes on to make up a fraction of it fuel loss by changing some non-fuel materials into useable fuel. The Breeder is a peculiar type of

machine in so far that it will actually create or "breed" more fuel than it will consume. These last two aspects are unique to nuclear power and serve to justify the earlier comments.

The only major draw-back to these last two types of reactors is that in order to reclaim this useable fuel one must reprocess the exposed fuel. As you may recall, the only reason that the United States does not reprocess it's fuel is a wholly political one. Hence, while the rest of the world avails itself of the benefits of reprocessing and actively develops their breeder technology, we here in the U.S. find ourselves saddled with 136 nuclear reactors [2] that simply continue to burn up our limited supplies of precious uranium in a wasteful and scientifically foolish manner.

In anticipation of political enlightenment and the re-discovery of the forth comming over depletion of available uranium reserves, it is strongly felt that research activities directed towards the Breeder option would be timely and well advised.

1.1 RESEARCH GOALS

It is the objective of this thesis and the associated research to show that a suitable reactor core can be developed with acceptable characteristics to permit a smooth transition from our current Burner technology to either a Converter or Breeder technology. This type of alteration must be equally applicable to both currently operating systems as well as to the next design generation of reactors. A secondary goal is to show that this design can be reasonably expected to attain extended specific burnup levels of 50,000 MWdays / MTU.

1.2 GENERAL ORGANIZATION

In order to satisfy these objectives, investigative research was pursued and the findings of that effort are contained within the body of this thesis. With a strong desire to present the information in the most logical and forthright manner possible, the following organizational format was adopted.

Chapter 1 is dedicated to the introduction of

the topic and the intended objectives connected to the research activities.

Chapter 2 offers a fairly detailed accounting of numerous contributing factors that served to shape and direct the efforts associated with investigation. This chapter is important since it sets the tone and offers the necessary background needed to bring the reader up to date with respect to this small area of the ever widening concerns relating to nuclear technological advances.

Chapter 3 is the major effort of the thesis and is divided into 2 major sub-parts. Part 1 treats the thermohydrodynamic considerations of the investigation and encompasses both the steady state and transient behavior aspects of the designs considered. Part 2 examines the neutronic implications imposed by the suggestions of Part 1 and continues on to show overall reasonability and feasability of this direction.

Chapter 4 outlines some of the advantages of considering a mixed oxide fuel material. In this section some general neutronic investigations are performed in order to asses the reasonableness of

this direction.

Finally, Chapter 5 offer the authors' concluding thoughts pertaining to this area of investigation and the application of it's findings to the nuclear industry.

1.3 PLANT SELECTION

Whereas over 65% of all the United States nuclear reactors are of the Pressurized Light Water variety, the Trojan nuclear power plant was selected to serve as the design base reactor. Having a rated net electrical output of 1130 MWe it is one of the larger type PWR's in service and should serve well as an indicative model for both retro-fitting and the new generation core design parameters.

2. MOTIVATIONAL FACTORS FOR THIS RESEARCH DIRECTION

2.1 THE OVERVIEW

As in all of the succeeding chapters in this work, a basic overview will be provided at the beginning. This should permit the reader a quick and easy method to preview the chapter and hopefully serve to stimulate some interest in the ensuing pages.

The reason for this entire chapter is to provide some needed background material and give a little continuity to this thesis. It is strongly felt that in order to understand and appreciate the authors' intentions, one must first be aware of various competing factors, important within their own right, that greatly effect this industry, and in that way, the saliency of this research.

As such, the first topic to be examined will encompass the energy situation and posture for both the United States and the World. Second, a close look at the available supplies of uranium and the

implications that may arise if the status quo is permitted to continue unchanged. Third, the economic factors must be weighed on both a large (global - national) and small (industrial - utility) scale. Fourth, a review of the advantages of Converter and Breeder reactor technology is covered, and how it will interface with the above items. Fifth, a quick glance at the current state of progress with respect to the world and U.S. in this subject area. Sixth, a quick recapitulation of major points that were brought out in the earlier sections will be supplied.

2.2 ENERGY DEMANDS AND PROJECTIONS

All creatures need energy to survive. Taking food and chemically extracting it's energy is quite probably the most elementary form that immediately comes to mind. One may inquire as to the disposition of all this necessary energy and it is suggested that a reasonable conclusion might be that any living organism that needs energy will utilize it to either adapt to or change it's environment to assure it's own survival. If one was searching for the most basic justification

needed in order to consider the question of energy supply and demand, the above should be as acceptable as any.

Man however, has taken this idea one step further along than that of any other creature yet known in that man uses energy to alter his environment not solely for plain existence but for self-perceived comfort as well. Man started eeaking out an existence in a cold, damp, cave not even a million years. Meanwhile, the fish has neither altered himself or his environs over this same timeframe. Now man has to "survive" in a penthouse (a contemporary type of cave) that overlooks all of his compatriots. It has been suggested that a fish is relatively content in being a fish and does not desire to build cities or go to the moon. Man although, is different because he has always been driven to improve his lot in life, to somehow better himself and the world as well. Therefore, it should come as little surprise that there exists a direct link between Man's energy consumption and his overall prosperity.

The International Institute for Applied Systems Analysis (IIASA) has conducted, what is

felt by the author, the most in-depth and carefully considered analysis for the world's future energy needs to date. The results of this 7 year study were published in 1981. The combined efforts of more than 140 scientists representing more than 19 countries has yielded the most credible energy projections to date. Whereas the regional results of this study as applied to the United States agree quite nicely with those of the U.S. National Academy of Sciences 1980 report, "Energy In Transition 1980 - 2010" [3], the author has elected to adopt the IIASA report as the primary source of energy projections for this effort.

The IIASA report has developed 3 credible scenarios to explore: (i) The high scenario refers to the situation of rather high energy demand which is associated with unrestrained global growth. (ii) The "low" scenario which equates to a moderate energy demand and is commensurate with a modest and controlled growth. (iii) The zero growth or "16 TW" case which is the base minimum amount of energy needed to maintain the status quo and have no growth or improvement at all.

The projections of these three scenarios into

the future as related to the world energy consumption is depicted in figure 2-1 [4]. This plot shows that for the 55 year period covering 1975 to 2030 one can expect the global energy consumption to rise by more than 66% for the low and almost 73% for the high scenarios respectively.

In more concrete terms, when historical tendencies are taken into consideration we see that a final projected energy level in 2030 would be 41 TW yr/yr for the high scenario and almost 22 TW yr/yr for the low [5]. For the benefit of comparison, the 1975 energy level was only 8.208 TW yr/yr [6]. Perhaps a better way to express this data is to offer table 2-1 [7]. This table breaks down each region into a per capita final energy consumption basis. Of particular interest are the implications to the United States (Region I) of the low scenario. This depicts a rather small increase of only 0.480 KW yr/yr which is to say a near zero energy growth for the next 50 years. The author maintains that it is unrealistic to expect the United States to accept, let alone tolerate, a zero growth rate in either it's energy consumption or Gross National Product (GNP) as indicated by this data. (the GNP link will be covered momentarily)

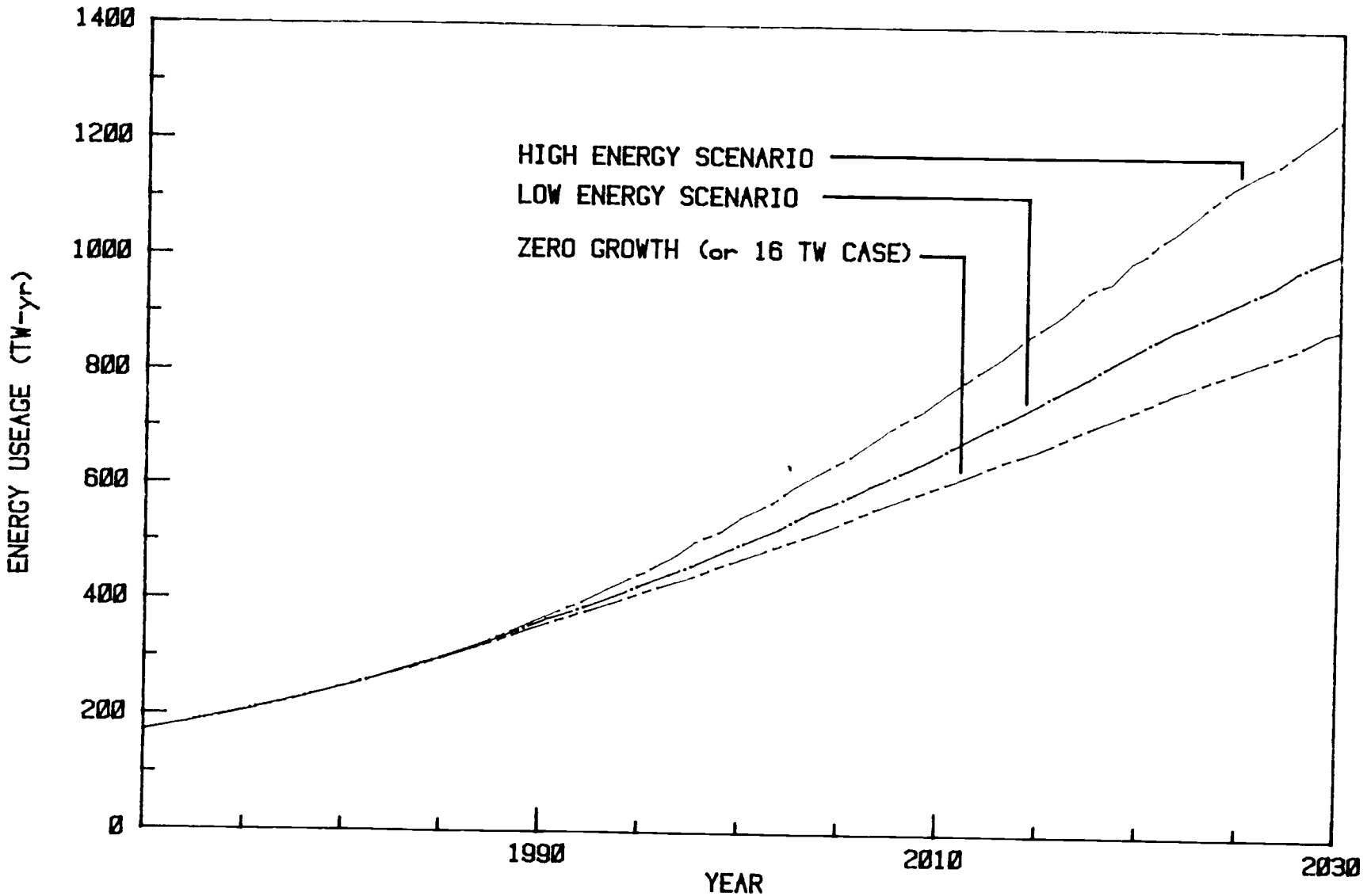


FIGURE 2-1 World energy consumption

TABLE 2-1 : PER CAPITA ENERGY CONSUMPTION

Final energy in the two scenarios compared to final energy calculated with historical elasticities (2030).

Region	*High Scenario (GWyr/yr)*	*With Historical e_f [a] (GWyr/yr)*	*Difference[b] (%)*	*Low Scenario (GWyr/yr)*	*With Historical e_f [a] (GWyr/yr)*	*Difference[b] (%)*
I (NA)	3665	6921	47	2636	4036	35
II (SU/EE)	4114	5355	23	2952	3850	23
III (WE/JANZ)	4375	6037	28	2987	3761	21
IV (LA)	2640	4385	40	1656	2481	33
V (Af/SEA)	3173	6900	54	1876	3121	40
VI (ME/NAf)	1638	2590	37	868	1015	16
VII (C/CPA)	3196	8849	64	1589	3536	55
World	22,801	41,037	44	14,564	21,800	33

[a]Calculated using historical (1950-1975) final energy-to-GDP elasticity (ϵ_f) for each region.

[b]Calculated as final energy, using historical ϵ_f minus IIASA scenario projection divided by final energy using historical ϵ_f.

When considering the United States apart from the rest of the world, recent USDOE projections of 1985 place our energy consumption level in the year 2000 at 97 Quads per year [8] which converts to 3.245 TW yr/yr. This is just barely (~1%) below the IIASA Region I low scenario projection of 3.310 TW yr/yr and only 9% off the high scenario level of 3.810 TW yr/yr. When it is recalled that Region I consists of both the USA and CANADA, the relative importance of the difference between these two reports is further diminished and tends to greatly enhance the credibility of both the IIASA and USDOE reports. A chart that shows the different regions is supplied in Appendix C.

The earlier references to a GNP - energy use connection may seem tenuous and a bit far fetched but the fact remains that it does indeed exist as is clearly demonstrated in figure 2-2 [9]. Furthermore, in a recent speech delivered by the former U.S. Deputy Secretary of Energy and retired Vice President of Bechtel Power Corporation, the Honorable W. Kennith Davis, supports the validity

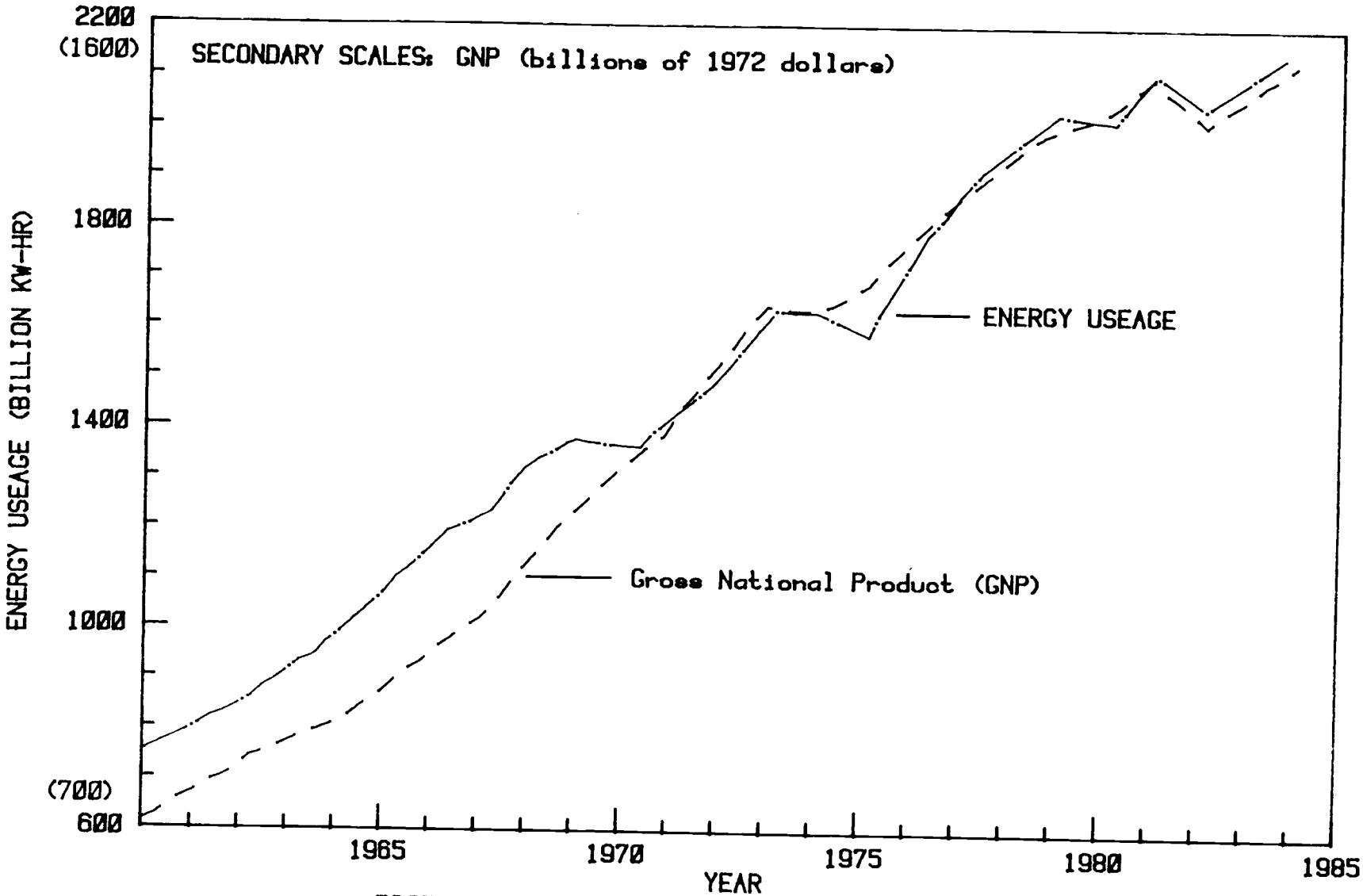


FIGURE 2-2 Energy and GNP growth

of this contention:

> "It is also the judgement of most observers that the increase in the demand for electric power can most prudently be assumed equal or exceed slightly the growth rate in the GNP. Thus, for planning purposes, a growth rate for electric power demand of 3% p.a. seems a realistic and prudent basis." [10]

The implementation of this projection was pointed out by Mr. Carl Walske, president of the Atomic Industrial Forum Inc. during a presentation that he gave to the Japanese just a couple of months ago. In that speech he stated:

> "... if demand grows at a moderate 3 percent per year--which is a little less than it has averaged during the past three years--the U.S. would need to add some 21 gigawatts of new capacity each year. This is equivalent to 21 large, 1000 MWe generating stations, not counting the need for additional capacity to replace aged or uneconomical units." [11]

He went on to build a strong case that allowed for nuclear power to capture 50% of this new market in the United States. This outlook is again comparable with the conclusions in the IIASA report. It forecasts that by the year 2030, nuclear power and coal could be expected to contribute upwards of 65% of our total energy needs. All this while oil and gas would drop to

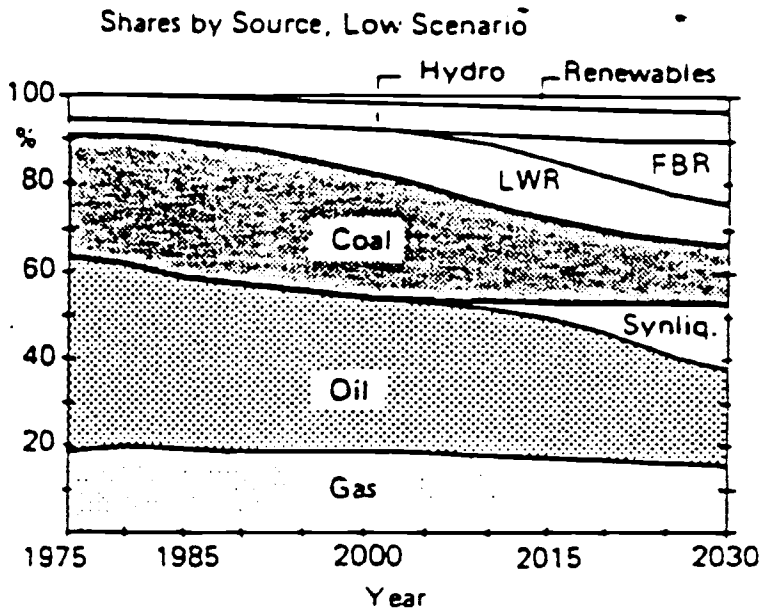
less than 30% which is a virtual flip-flop of the 1975 partitioning. The report observed that:

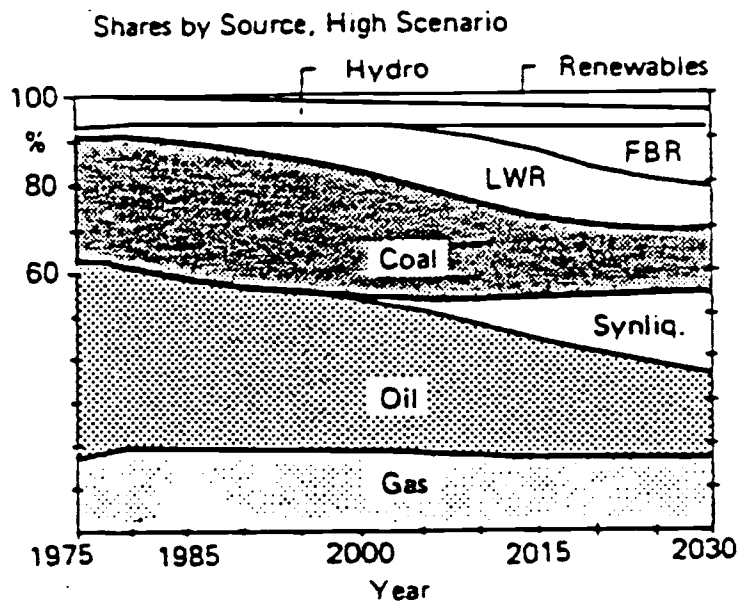
> "The supply mix of the future seems to be driven toward coal and nuclear energy, based on the aggregate of supply constraints and market restrictions." [12]

Globally, this trend appears to be identically transferrable regardless of the scenario adopted as depicted in figure 2-3 [13]. Again we can see that nuclear starts to increase it's share of the energy supply picture starting around 1985. (note; nuclear includes both LWR and FBR types)

2.3 IMPLICATIONS ASSOCIATED WITH THE CONTINUATION OF CURRENT BURNER TECHNOLOGY

When nuclear power was first introduced to the world, it was a god-send, the ultimate answer to our future energy needs. After all, it not only produced cheep, clean, efficient electricity, it had a limitless fuel supply to boot! Well, as was pointed out in Chapter 1, this last point was overstated and depending upon the implementation of Breeder reactors, may or may not be absolutely correct. The problem is the overwhelming resistance the United States has exhibited to the



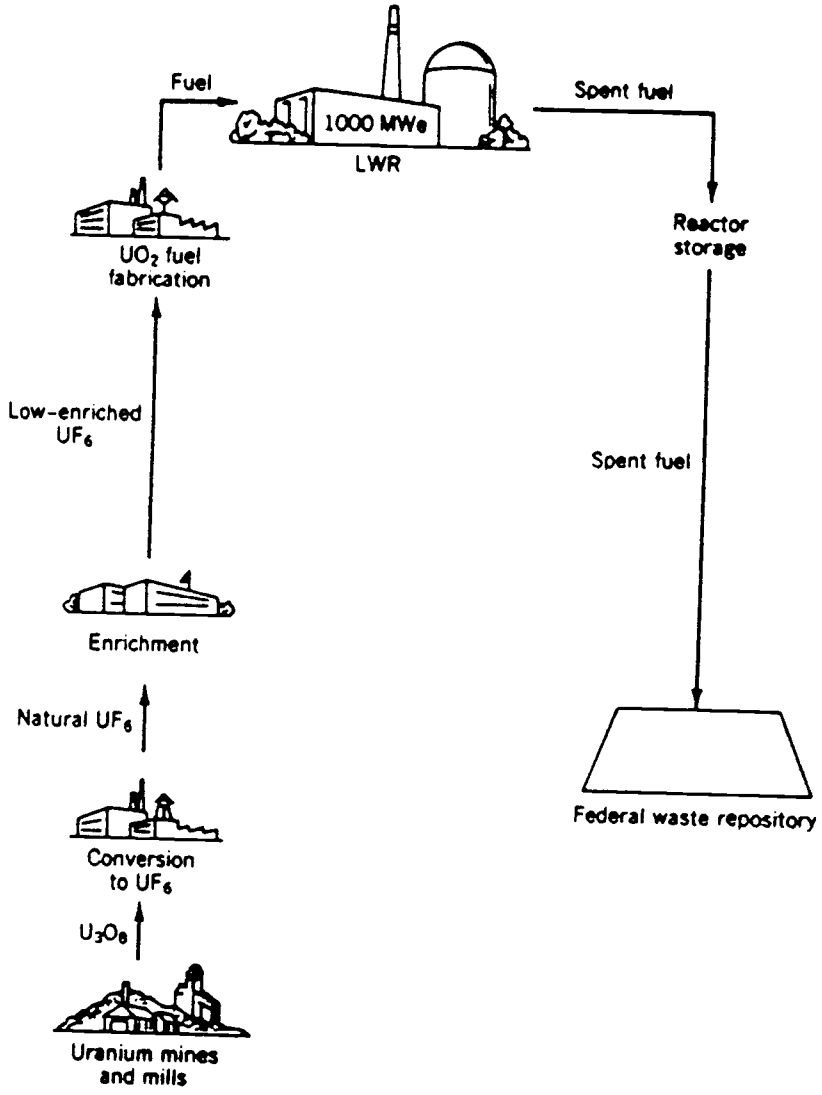


Global primary energy, two supply scenarios, 1975-2030.

FIGURE 2-3 Global energy supply scenarios

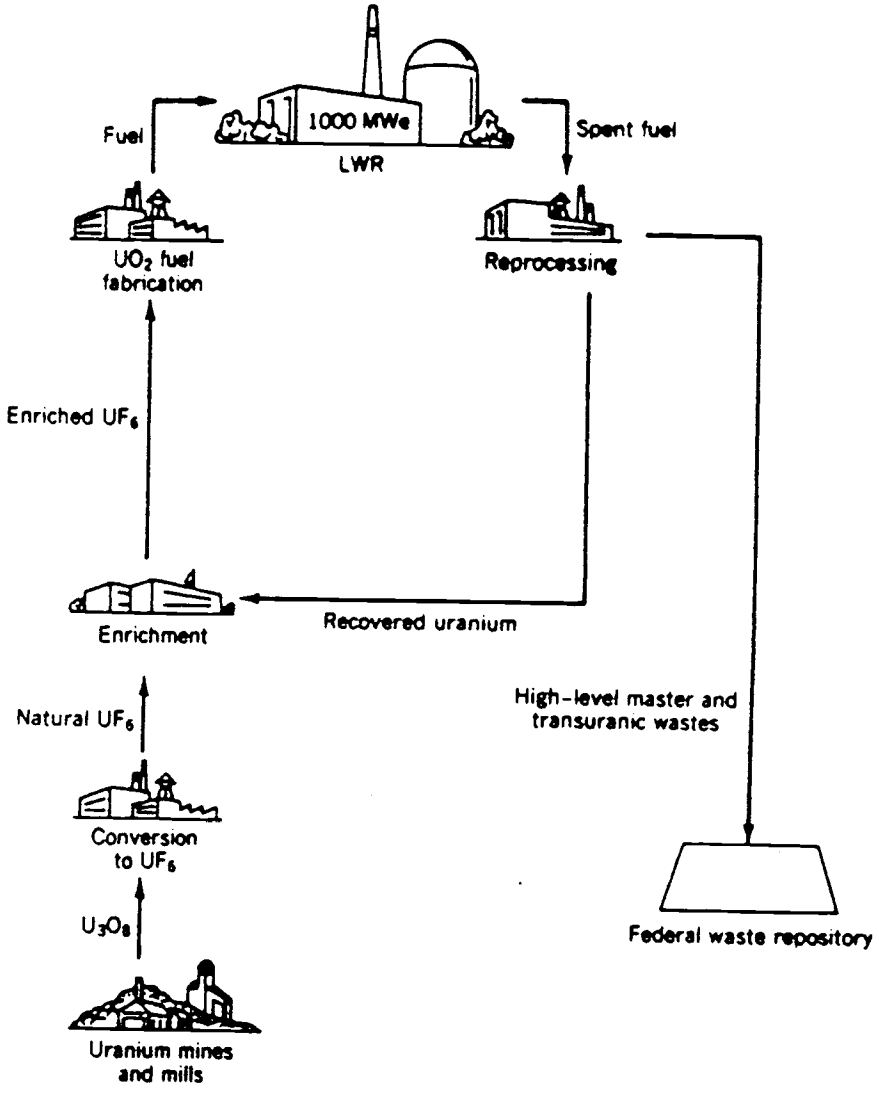
replacement of the once through fuel cycle. After all, it takes no great amount of insight to see that this method needlessly squanders millions of watts of useable energy as well as the limited supplies of uranium.

To clarify the difference between a once through fuel cycle and a complete one, figure 2-4 [14] and figure 2-5 [15] are provided for purposes of comparison. It should be noted that a substantial amount of fuel grade material is disposed along with the irradiated fuel which directly translates into useable kilo-watts of energy. To illustrate this point, an example is in order. Upon each refueling of a 1000 MWe plant, the discharged fuel contains about 180 kg of fissile plutonium and 220 kg of uranium-235 due to a poor fuel utilization factor. If this material were to be reclaimed and used to refuel a reactor, it would deliver enough energy to displace almost one million tons of coal [16] which is roughly equivalent to 4.8 million barrels of oil. Questions now raised as to how much uranium is available and why it is so much of an issue will each be addressed in turn.



A once-through fuel cycle. In this cycle spent fuel is handled as waste and is prepared for permanent disposal.

FIGURE 2-4 Once through fuel cycle



Fuel cycle with uranium recovery only.

FIGURE 2-5 Recycle fuel cycle

When one discusses the amount of uranium bearing ores that are available, one is immediately confronted with the two seemingly identical terms reserves and resources. This confusion needs to be dispatched before proceeding further. A reserve is a term used in connection with a reliably known, well explored and assayed mine; hence it's relative quality and quantity are fairly well understood. A resource on the other hand, is used when a site has not been examined in detail and the estimates as to it's holdings are based solely upon indirect evidence and generalized associations [17]. Put another way, the money in a nations' treasury is a resource and your bank account is a reserve.

To give a numerical sense to this subject, two reliable projections deserve consideration and are outlined in table 2-2 [18] and tables 2-3 and 2-4 [19] below. The first comes from the IIASA report and utilizes a global averaged smear model approach to secure an estimate. This is accomplished by taking a relatively well explored region, say like the United States, and divide the total amount of reserves by the total exposed surface area. This returns a gross average or "smeared" value for the

TABLE 2-2 : ADJUSTED URANIUM RESOURCES

IIASA World Regions	*Area* (10^6 *km*^{2})	*OECD-NEA/IAEA Estimate* (10^6 *tons*)[a]	*IIASA Estimate* (10^6 *tons*)
I (North America)	21.5	2.53	3.87
II (Soviet Union and E. Europe)	23.5	–	4.23
III (W. Europe, Japan, Australia, N. Zealand, S. Africa, Israel)	15.5	1.26	2.79
IV (Latin America)	20.6	0.08	3.71
V (Africa except N. Africa and S. Africa, South and Southeast Asia)	33.6	0.33	6.05
VI (Middle East and N. Africa)	9.8	0.08	1.76
VII (China and Centrally Planned Asian Economies)	11.5	–	2.07
World	136	4.29 (14.2-26.4)[b]	24.48

[a] Excluding regions II and VII.
[b] Including the speculative resources given in *OECD-NEA/IAEA* (1977).

TABLE 2-3 : U.S. URANIUM RESERVES - 1981

	Thousand Tons, U_3O_8
$30/lb	470
$50/lb	787
$100/lb	1034

Source: From GJO-100(81).

TABLE 2-4 : PROBABLE U.S. URANIUM RESOURCES

Probability Distribution Values for Potential U.S. Uranium Resources,

January 1, 1981[a]

Forward-Cost Category	Thousand Tons U_3O_8		
	Mean	Ninety-fifth Percentile	Fifth Percentile
$15/lb U_3O_8			
Probable	295	185	448
Possible	87	42	156
Speculative	74	30	162
Totals	456	280	704
$30/lb U_3O_8			
Probable	885	659	1161
Possible	346	194	530
Speculative	311	155	600
Totals	1542	1094	2097
$50/lb U_3O_8			
Probable	1426	1102	1802
Possible	641	346	973
Speculative	482	251	890
Totals	2549	1845	3369
$100/lb U_3O_8			
Probable	2080	1646	2573
Possible	1005	521	1526
Speculative	696	378	1225
Totals	3781	2766	4923

Source: Adapted from GJO-100(81).

[a]Estimated potential uranium resources include losses from mining; losses due to milling are not included and may range from 5 to 15%.

concentration of material over unit of land area. This value is then multiplied by the total land area of the world to arrive at a somewhat reasonable value for the amount of total resource to be had. This is how the IIASA group arrived at a figure of 24.48 million tones of uranium as an ultimate reclaimable level for the world. At first glance this method might seem capricious and unrealistic,but it must be realized that as current rich deposits become depleted, it becomes increasingly more economical to develop deposits of lower concentration that were unacceptable before. These figures should be contrasted with the projections of table 2-3 which represent the position of the United States as of 1981. A more detailed summary of the U.S. posture can be found in table 2-4 which shows both the levels of reserves and resources projections broken down into units of dollars per pound of yellowcake. (U_3O_8 is often referred to as yellowcake) All other estimates that were reviewed ran fairly close to the values depicted in these two samples.

If one now realizes that it takes almost 6,000 tones of Uranium to operate a 1,000 MWe LWR for it's entire 30 to 40 year lifetime, then we can

gain some valuable insight with just a few simple calculations. Based on the IIASA data somewhere between 4 and 715 separate 1,000 MWe plants can be ultimately supported, or if the other information is used, apparently 1,394 such plants could be supported to supply part of the world's energy. To put it another way, for each 1,000 MWe of installed LWR capacity, 130 tons of natural uranium is required annually. Thus, for an installed capacity of 10 TWe, which is projected by IIASA, more than 1.3 million tons of natural uranium would be required each year. Accepting the generous prediction of 24.48 million tons for a moment, this clearly indicates that the world's entire useable supply of uranium would be exhausted by the year 2030, a brief 44 years from now [20].

No matter how one chooses to figure it out, the conclusion that is arrived at is always the same: If the once through or burner cycle is continued unaltered, all of the available uranium supplies in the world will be completely exhausted within our lifetime. Any remaining deposits would be of such low quality that the cost to reclaim any useful amount would drive fuel costs to economically prohibitive levels. In short, the

nuclear power industry would be bankrupt.

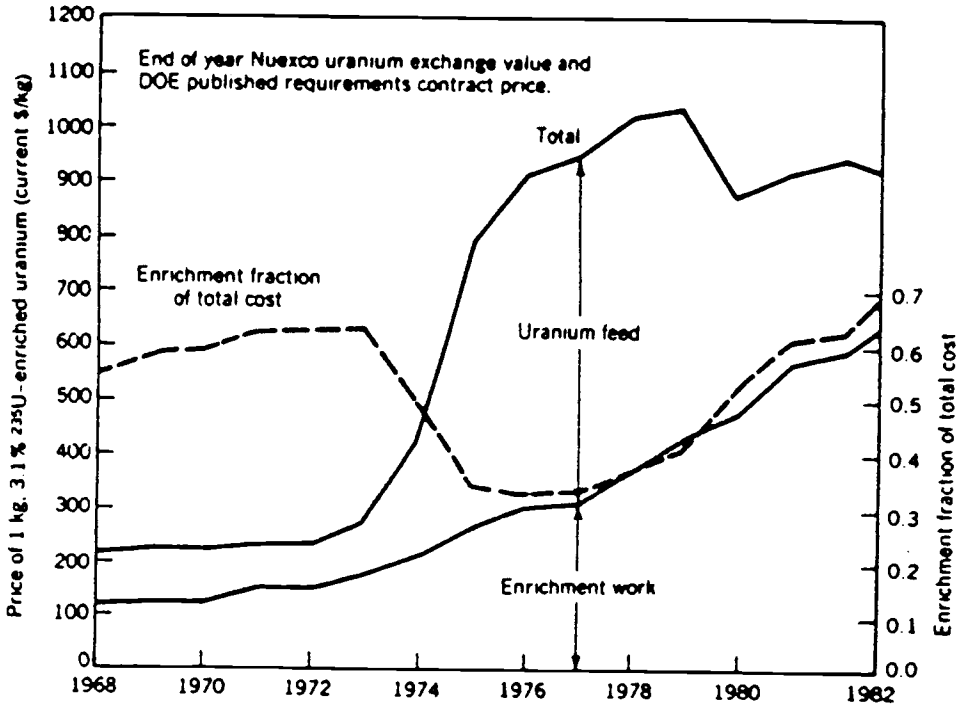
Now we arrive at the answer to the second question posed earlier which is to ask "so what ?" Often one may encounter the refuteing argument to the above scenario which contends that Fusion power will arrive just in the nick of time, and failing in this, the utilization of either plutonium or thorium will extend the cycle until it does. The only flaw with this position is the fact that in order to get thorium to transmute into uranium - 233 or activate the uranium - 238 / plutonium chain, it must undergo irradiation within an operational reactor. In order to accomplish this a certain amount of "seed" fuel is required to prime the pump, so to speak. This "seeding" will permit the establishment of a level of fissile isotopes (Pu-249, U-233) in sufficient quantity to achieve the desired prolongation of nuclear power. Obviously, without sufficient supplies of uranium - 235, this system will, like an unprimed pump, become an uterly useless piece of machinery. Add to this the fact that uranium - 235 is the ONLY naturally occuring fissile isotope at the meager level of 0.072 wt %, the scope and seriousness of the dilemmia becomes crystal clear.

The loss of the prospective contribution that nuclear power would have to be compensated for with a substantially increased reliance upon the remaining finite resources. This in turn will cause their depletion to hasten and become exhausted much more quickly than originally anticipated. The bottom line of this line of thought is cruelly evident...the continuation of current trends could leave the United States and the world saddeled with a severe shortfall of energy. In fact it could be of such magnitude so as to preclude the development of a replacement energy source thus completing the cycle and taking us back to the cave except that this time there will be no fire to provide warmth. This may appear to be fanatical ravings or a fantastic fabrication on the authors' behalf, but given the few rather reasonable caveats from above, a logical analysis will inevitably yield the same conclusion.

2.4 ECONOMIC CONSIDERATIONS

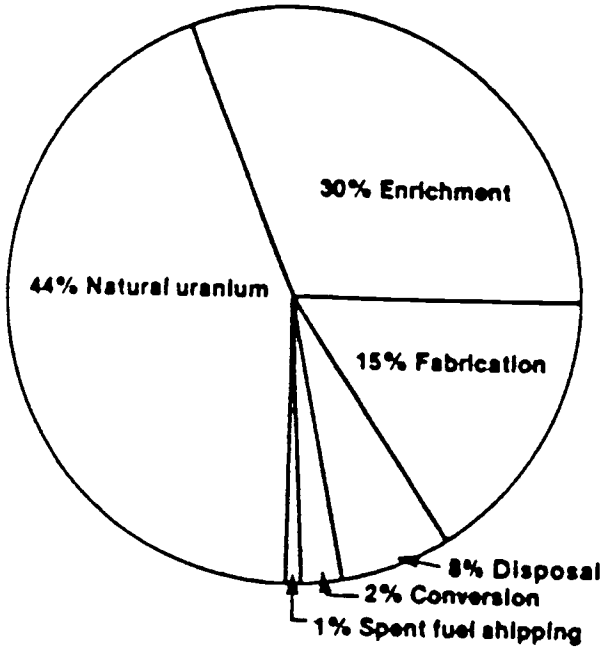
If we now retreat from the crystal ball of Doom and Dispair, we find ourselves living in a world of dollars and cents realities. Here where the final costs of goods and services weigh heavily, almost to the point of excluding everything else, in the decision making process. In the last section it was shown that the link between money and resources acts as a prime indicator of both availablity and demand. As the quality or yield of ore deposits plummet with increased demand coupled with the attendant resource depletion, the costs for obtaining a given amount of end product is increased. This price escalation does not stop with the mining chain, but will percolate up throughout the entire fuel cycle and eventually to the customer and beyond.

A clear demonstration of this can be seen in figure 2-6 [21] which shows that as ore asseys decrease, enrichment effort increases along with feed requirements thus leading to a total cost escalation. In order to show the disproportionate behavior inherent to the fuel cycle, figure 2-7 [22] should prove helpful. Note the great



Cost of 3.1% enriched uranium.

FIGURE 2-6 Relative fuel enrichment costs



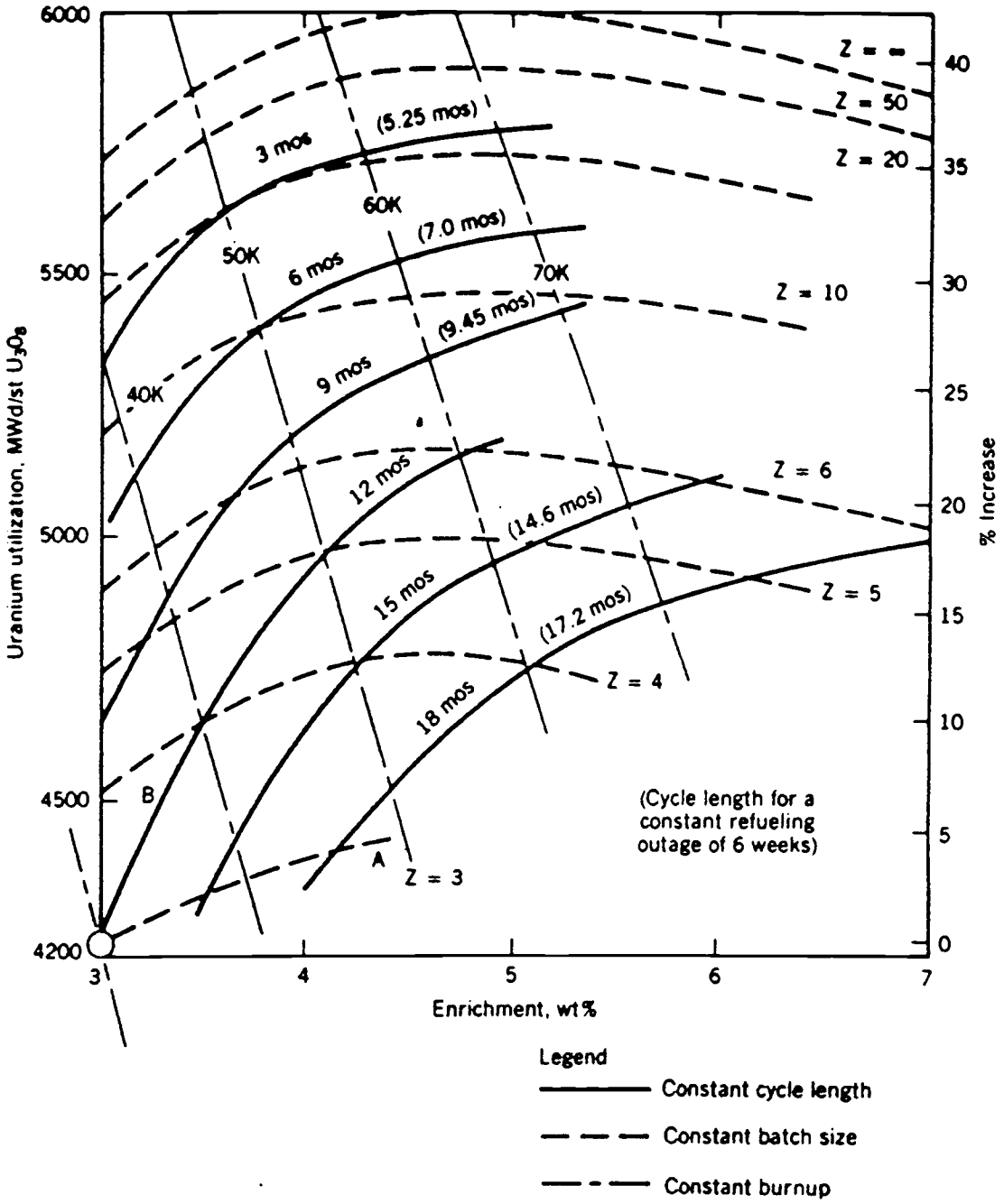
Nuclear fuel cost inputs. Distribution of nuclear fuel cost components (based on 30-yr levelized mills/kWh). All prices in 1982 dollars. (1) Yellowcake, \$31/lb U_3O_8. (2) Conversion, \$3.10/lb U. (3) Enrichment, \$130.75/SWU. (4) Fabrication: (a) first core, \$225/kg; (b) reloads, \$200/kg. (5) Spent fuel transportation, \$36/kg. (6) Spent fuel disposal, \$275/kg.

FIGURE 2-7 Nuclear fuel cycle costs

disparity in share-costs between the front and back end of the fuel cycle.

The question now arises as to how a utility that forthcomming fuel acquisitions will begin to escalate. The complexity of this issue is self-inflating due to the interdependancy of a variety of issues, of which fuel burnup, enrichment work, storage and replacement power costs are but a few, combine to determine the final costs.

If the specific burnup is raised, better fuel utilization is achieved through the extraction of more energy from the fuel. As exposure is increased, greater amounts of U-235 are fissioned thus allowing more total heat energy to be released and simultaneously lowering the amount of residual unburned U-235 that remains resident within the discharged fuel. The price incurred for this improvement is the cost of the increased fuel enrichment necessary to maintain criticality and operate the reactor. This additional cost will be off-set with the reduction of fabrication, spent fuel storage, replacement power and reprocessing costs (if applicable). In figure 2-8 [23] the utilization of uranium is indicated as a function



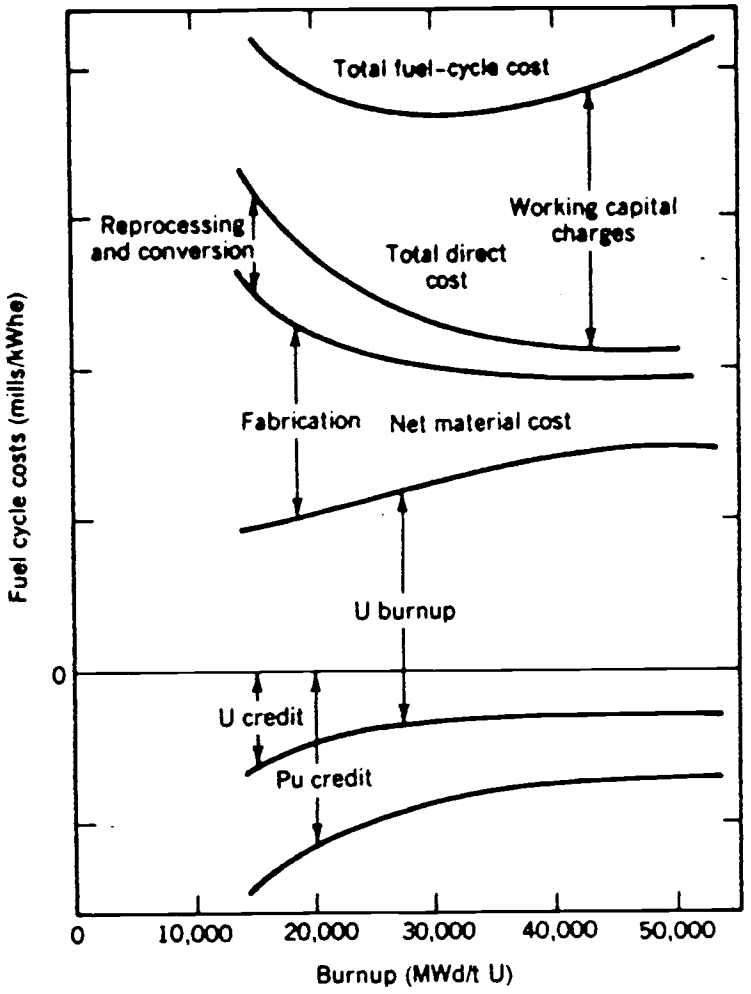
Uranium utilization as a function of enrichment and cycle length.

FIGURE 2-8 Uranium utilization chart

of enrichment and burnup.

As a illustration, if a plant would increase it's burnup from 33,000 to 50,000 MWD/MTU for a 9 month cycle, it would need to increase the fuel enrichment to around 4%. However, uranium utilization would rise 24% and require only 1/8 core replacement each outage. As an additional benefit of this arises from the fact that as the replacement fraction of the core is lowered, total fuel cycle costs are reduced as well, which is easily seen in figure 2-9 [24].

Another feature of this scenario is a reduction in replacement power needed by the utility. Replacement power is that which would normally be supplied by the nuclear plant but must be purchased from external suppliers for the duration of the outage. For a large 1,000 MWe plant these costs are significant as they range in the vicinity of 1 million dollars per day. So it is intuitivly obvious to the casual observer that replacing only 1/8 (12.5%) of the core instead of 1/3 (33%) will necessarily reduce the down time and thus serves to further off-set the additional enrichment costs. In fact, under some



Fuel-cycle costs versus discharge burnup.

FIGURE 2-9 Fuel cycle lifetime costs

circumstances the replacement power savings in addition to the other factors may serve to make it less expensive to pursue this fuel cycle than the one currently in use.

2.5 BENEFITS OF CONVERTER-BREEDER TECHNOLOGY ADOPTION

If nuclear power is expected to remain a growing and viable energy resource in the next century, then the facts as presented overwhelmingly point out that the adoption of Breeder and improved Converter technologies has become imperative. As illuminated in the IIASA report, a one-sided development of Converter technology will buy us some time, perhaps as much as 15 years in extended fuel material supplies [25] but the results are inescapeable. As observed in the report back in 1981:

> "One can play with the numbers, but the fact remains that burning only fissile atoms makes Nuclear power a short affair when viewed in the global context considered here." [26]

So it would appear that a duel purpose strategy is in order, one that will assure improved fuel utilization will be achieved through the

creation of Converters out of currently deployed Burners and two, to see that Breeder technology is implemented and phased in as soon as possible. Only with a full deployment of the Breeder can the original promise of Nuclear power be realized, that of being a near inexhaustable supplier of energy. It is interesting to note that the introduction of the Breeder would not only ease demands upon our precious Uranium resources but very low grade ores, on the order of a few ppm, would eventually become economical in terms of fuel. This would translate into a energy potential of some 300,000 TW-yr [27] which is equivalent to the potential offered by Fusion energy. In essence, a near infinite power capability is at hand for the taking. What other advantages or reason does one need??

2.6 U.S.A. vs WORLD : CURRENT DEVELOPMENTAL TRENDS

Historically, the United States has led the rest of the world in Nuclear Technology development and implementation, but the future of that position is now called into question. The cancellation of the Clinch River Breeder Reactor Project, which was our only major undertaking in the area of Breeder reactors, delt a crippling blow to our position and

research. Now France and Japan are the only countries that are actively pursuing this area. France has introduced it's first generation Breeder, the PHOENIX, just a year or two ago.

Recent events have only seemed to heighten the U.S. public's anxiety and mistrust of this industry which only serves to further retard the healthy progress and exploration of new concepts and ideas in this area. All in all, a recipe appears to be in the making that will one day require the U.S. to purchase some of the latest technology from a foreign nation which could have been developed domesticaly. An alternative that is wholely unattractive on grounds of National pride, economics and is in direct contradiction to a policy of energy independence.

On the issue of Converters and Breeders, one recent article was brought to my attention concerning work which was being conducted in the Federal Republic of Germany. This article is entitled, "GENERAL FEATURES OF ADVANCED PRESSURIZED WATER REACTORS WITH IMPROVED FUEL UTILIZATION" [28] which piqued my interest and directly inspired me to examine this issue in detail. The findings and

conclusions are contained herein.

In the article, a somewhat detailed design description of an advanced pressurized water reactor (APWR) is offered, but is noticeably lacking in reporting in areas such as pressure drops across the core. This single issue is at least as important if not more than any of the others that were covered in the report. Following a rather disappointing response from a direct solicitation for more details concerning this area, an investigation using the available data as put forth within the article was undertaken. Using a thermohydrolic code called COBRA IV-C as implemented on a CYBER 170/710 mainframe computer, the preliminary results indicated a across the core pressure drop of 300 psia. These results are unmanageable and unacceptable when considering the retro-fitting of a currently operational reactor. This point raised some questions pertaining to improved Converters and the feasiblity of retro-fitting improvements into on-line plants and consequently gave rise to this research effort.

2.7 SUMMARY

In this section we have seen that future energy supplies are projected to rely heavily upon Nuclear power as a significant contributing source. That continued use of Burner technology will only serve to prematurely kill this important and vital source via the over depletion of available raw resources.

An effort was also made to show some of the opportunities and possible actions that are currently at our disposal as well as indicating the various complexities involved in the balancing of numerous factors that affect the future and utilization of Nuclear power. At the end, the final push that instigated this research was reviewed.

Admittedly, this is a somewhat lengthy chapter replete with a flurry of diagrams and tables. This was done for two reasons; first to act as a refresher and fairly through outline of some of the more important areas concerning this subject and two, to indicate what motivated the author to pursue this topic instead of something else.

3. THE SEARCH FOR AN IMPROVED CORE DESIGN

3.1 THE OVERVIEW

The design of a nuclear reactor core is one that necessitates many compromises in order to arrive at viable alternative. A functional core will exhibit both good neutronic and thermohydrodynamic characteristics. It will also blend materials that provide strength and durability under prolonged radiation and heat exposure and still afford low levels of corrosion, heat induced deformation and neutronic absorption. The core must be able to not only endure the rigors of normal operation, but handle the plausable accident conditions that may occur throughout its designed lifetime as well. The many factions that gauge the afore mentioned characteristics will inevitibly play off each other to varying degrees. This then is the mission of a core design engineer, to investigate and be cognizant of these factors in order to find suitable combinations that will render improved fuel performance and maintain economic viability.

In this chapter, many of these characteristics will be explained as we seek to develop a viable core design alternative to current PWR cores that will improve fuel utilization and prepare the way for the eventual introduction of breeder reactors into the U.S. nuclear electrical supply economy. Subpart I will cover the thermohydrodynamic aspects of this design from both a steady state and selected transit basis. Subpart II will cover some of the more important neutronic considerations for the design set forth in Subpart I.

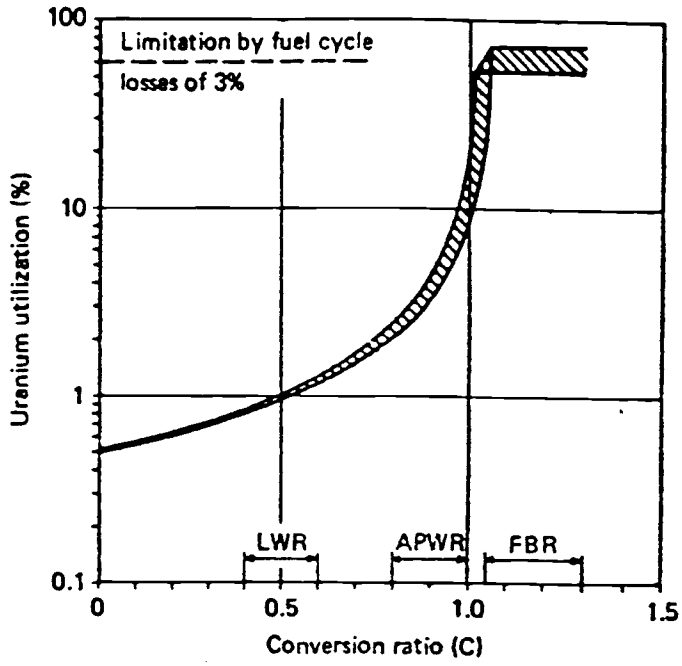
3.1.1 THERMOHYDRAULIC CONSIDERATIONS

The overriding concern is regards to this area when designing a new core deals with the components of flow, heat movement, and the materials physical limitations. Items like pressure and flow regimes and their effect on the efficiency of heat transport become of paramount importance since a failure in either aspect will completely negate any gains made in neutronic considerations. With this in mind, we should then proceed to discuss some of these limitations and the convenient ways that exist to evaluate these attributes as they relate

to the core design process. Following this we will review the reference core characteristics and attending assumptions that were employed to facilitate this analysis. With the boundaries thus outlined, we will then proceed to examine the steady state analysis and develop a base design. Establishment of this design will then be subjected to a series of transient analysis that should give a good indication as to the design's overall performance under both normal and abnormal circumstances.

In the current generation of LWRs only 0.6% of the natural uranium is consumed, the addition of reprocessing and plutonium recovery would only increase the LWR's fuel utilization to 1%. If the conversion ratio was increased from 0.60 to 0.90, the uranium utilization factor could be advanced to 4% as indicated in figure 3-1 [29] which is a substantial improvement. If a conversion ratio exceeding 1.0 could be obtained this factor could reach values ranging from 8% to 30% and very quickly attain the maximum value of 70% with conversion ratio as little as 1.05!

This may seem to be mindless hypothesizing,



Utilization of natural uranium in closed fuel-reprocessing cycles as a function of conversion ratio.

FIGURE 3-1 Natural uranium utilization curve

but results just announced from the Shippingport Reactor, a light water breeder reactor experiment, shows unquestionably that a LWR breeder is not only plausable, but readily available. In the report released during the 1986 ANS Winter Meeting, this 60 MW(e) nuclear power plant showed that a conversion ratio exceeding the self-sustaining level, which allows for a reprocessing loss of 0.5% have been achieved, which is to say that breeding has occurred [30].

In order to achieve these results, considerable pitch reduction appears to be necessary. This would in turn reduce the core's water volume fraction and therefore harden the neutron spectrum. Clearly, a pitch reduction will also increase the pressure drop across the core. To compensate for this, one could reduce the core heighth, alter the flow rate, or a judicious combination of these. To alter the flow rate, the designer could either replace the reactor coolant pumps, one option that is by far too expensive when considering a operational plant, or redirect the water through a series of baffles and orifices inside the reactor vessel. This latter option is the most favorable since the core internals would

need some modifications to accommodate newly designed assemblies at any rate so the attending costs would appear to be well within the range of acceptability.

This then seems to dictate a reduction in actual water flow within the core. A reduction in thermal output of the core would seem to be necessary which can be partially compensated with an increase of cross core temperature input. This mandates a reduction in core inlet temperature since increasing the exit temperature could bring on the onset of bulk boiling within the core's hot channels. In order to accomplish this, the operating temperatures and pressures of the secondary systems would be dropped. This finally results in a reduction in the amount of steam available to operate the turbines. Steam flow is conserved but the density is sacrificed, which leads to an electrical output reduction. This down-rating of the reactors performance should be marginal (1-2%) [31] as indicated through previous studies.

With the general aspects now presented, more design specification material will be offered.

3.1.2 REFERENCE CORE PARAMETERS

As in any study of this type, a comparison is needed with a current generation design that is operational in older to conduct a valid evaluation. With that in mind along with the desire to assure retro-fitability for on-line power station reactors, the author selected the Trojan Nuclear Power Plant as a design reference base criterion. This nuclear system was designed by Westinghouse Nuclear which is a company that is responsible for most of the operating PWRs in the United States. Considering the numerous differences between PWRs and BWRs, the author opted to restrict his research to PWRs only.

Trojan is rated to produce 3411 MW of thermal power and deliver 1130 MW of useable electric power to the regional power grid system. Primary system pressure is 2250 psig and the primary coolant mass flow rate is 132.70 E+06 Lbm/Hr as supplied by two reactor coolant pumps. The remaining specific details can be found in Appendix B for ready reference. Also supplied in that appendix are

graphic representations of the core and fuel assembly constructions. Although the reactor was initially designed for 16 x 16 fuel assemblies, the current fuel assemblies are 17 x 17. This change was done for both economic and safety considerations and consequently the 17 x 17 core was selected for this study.

To assure total compatibility with current cores in order to facilitate retro-fitting, the author restricted the overall dimension of the square assemblies to that of the 17 x 17 design. Further, reduce the number of variables, the overall dimensions of the fuel rods were maintained at their original values. This not only simplifies the parametric analysis but largely eliminates the re-configuration of the fuel fabrication facilities involved. Again for convenience, the fuel rod has been diagrammed within Appendix B.

One remaining issue is that in order to assure structural stability, grid spacers are used in the 17 x 17 core. This grids are equally spaced over the entire 13 foot length of the assembly and of the "Type R" variety. A sample of this grid is also supplied in Appendix B. It should be

mentioned here that if rod to rod spacing is 0.0394 inches (1 mm) then these spacers become too difficult to manufacture and lead to unacceptable pressure losses stemming from the increased flow restrictions [32]. If spacing this close is required, changing to helical finned fuel rod is purported to be a superior choice. Whereas pressure loss correlations and coefficients were withheld under proprietary protection and since our analysis code was not equipped to this accommodate this option, the author elected to continue on with the more conventional grids to assure comparability.

3.1.3 COMPUTER CODE SELECTION

The selection of a suitable analytical computer code was mainly one of convenience and accessibility. Considering that this research is commensurate with a first level or elementary position, the use of the VIPER code was immediately dismissed. This left the thermohydraulic code entitled COBRA-IV which is implemented on a CYBER 170/710 Mainframe computer.

This code performs a subchannel analysis to determine enthalpy flow and temperature distribution for rod bundles under steady state and transient conditions. Results are offered in individual rod and bundle averaged formats.

COBRA-IV was developed through Battelle Northwest Laboratories for detailed discussions concerning the equations and models employed the reader is urged to consult reference 33. This code offers a sufficient level of analysis to answer the demands associated with this level of research. The time required to run this code using the mainframe computer on a time shareing basis failed to over-extend our permissible levels which most other codes would have likely exceeded. This economizing of a resource that was limited to begin with was a distinctive factor in selecting COBRA-IV as well.

3.1.4 MAJOR CORRELATIONS OF INTEREST

Although many correlations are employed within the code, a number deserve an appearance within the text of this thesis. These will be limited to the relationships pertaining to Critical Heat Flux (CHF), fuel thermal conductivity and fluid friction factors.

The first is the CHF correlation which was selected for use to facilitate the analysis. Due primarily to its wide acceptance within the field and long term survival as a realistic and valid relationship, the author selected the W-3 correlation by L. S. Tong [34]. The correlation is as follows:

```
q" crit / 1.00E+6 =                                  (EQN. 1)
     [(2.002 - 4.302E-4*p)
   + (0.1722 - 9.48E-5*p)
   * EXP((18.177 - 4.129E-3*p)*X)]
   * [(0.1484 - 1.596*X + 0.1729*X*|X|)
   * (G/1.00E+6) + 1.037]
   * (1.157 - 0.869*X)
   * [0.2664 + 0.8357 * EXP(-3.151 * De)]
   * [0.8258 + 7.94E-4 * (Hsat - Hin)]
```

where the limits of this correlation are:

p = 1000 - 2300 psia
G = 1.00E+6 to 5.00E+6 Lbm/Ft^2-Hr
De = 0.20 to 0.70 inch
X = -0.25 to +0.15
L = 10.0 to 144.0 inch
q"crit = Uniform Heat Flux
Heated/Wetted Perimeter = 0.880 to 1.00
Geometry: *Circular tube
*Rectangular Channel
*Bare Rod Assembly

Spacer Grid Correction Factor:

$$Fs = q''crit\ spacer\ /\ q''crit\ bare\ rod \qquad \text{(EQN. 2)}$$

$$= 1.00 + 0.030 * (G/1.00E+6) * (TDC/0.019)^{+0.35}$$

where:

TDC = ϵ / V a
ϵ = eddy diffusivity
V = axial velosity
a = gap distance between rods

The TDC correction factor is 0.180 for a PWR of this type.

The parameter of major significance is not the CHF values but rather the ratio of the critical heat flux based on the correlation's results to the most limiting heat flux condition within the reactor. This ratio is frequently known as the Critical Heat Flux Ratio (CHFR) or more commonly as the Departure from Nuclear Boiling Ratio (DNBR). This ratio is a strong figure of merit which is used to analytically assess the probability of a boiling crisis occurrence at a given location. The term "boiling crisis" refers to the onset of

significant boiling and subsequent voiding of the subchannel region under consideration. A diagram of the different boiling regimes is offered in Appendix C. This becomes important when it is recalled that the uniform heat transfer from the rod is dramatically reduced when significant ratios of the coolant are in fact steam voids. These voided pockets localize the heat and lead to hot spots which enhance the possibility of cladding failure, a situation that we would prefer to avoid. Characteristic minimum values of DNBR for PWR plants are in the vicinity of 1.20 or higher which is the result of allowing for a safety range to compensate for the inherent uncertainties associated with the CHF correlation projections and the materials' heat transfer properties. For the Trojan plant the minimum DNBR (MDNBR) value is 1.30 [35].

With regards to fluid flow friction factors, COBRA-IV differentiates between laminar and turbulent flow regimes. The input of COBRA-IV allows these relationships to be entered in the form of:

$$f = AA\ (Re)^{BB} + CC \qquad \text{(Eqn. 3)}$$

From reviewing a variety of sources treating this subject the following standard values were finally selected for the AA, BB, CC constants of eqn (3) above [36] :

Turbulent friction factor:	AA =	0.079
	BB =	-0.250
	CC =	0.000
Laminar friction factor:	AA =	16.00
	BB =	- 0.50
	CC =	0.00

To handle the rod to coolant heat transfer, the code was permitted to use the widely accepted Dittus-Boelter correlation which is already implemented within the code.

Finally, to assume a more realistic modeling of the real world fuel designs, the author elected to employ a correlation to permit variable thermal conductivity in the fuel itself. This option is much preferred over the use of a single value since the fuel rod can see variations and temperatures ranging from 60 F to 1200 F or more. Without any doubt, the definitive work in this area is under the supervision of Argonne National Labs and was published as a part of the material properties or

MATPRO report [37]. The main drawback to the MATPRO correlation is that is is expressed only in an exponential format which cannot be used within the context of the COBRA-IV code. This necessitated the development of KLIEWER psuedo-correlation which is a polynomial best fit to the MATPRO data and is thoroughly discussed in Appendix A of this work.

3.1.5 DEVELOPING THE INPUT FOR COBRA-IV

Great care went into the development of the COBRA-IV input deck for it is this vehicle that will determine the accuracy and viability of the code's results. Whereas this code was formulated using FORTRAN-IV, its sensitivity on input spacing is notorious and unforgiving thus demanding the strictest attention to detail. A sample input file is given in Appendix C.

A couple of items of mild interest should be discussed before proceeding to the subchannel model development. First the axial heat flux profile was selected to be a conventional cosine shape as depicted in figure 3-2 which was "area normalized"

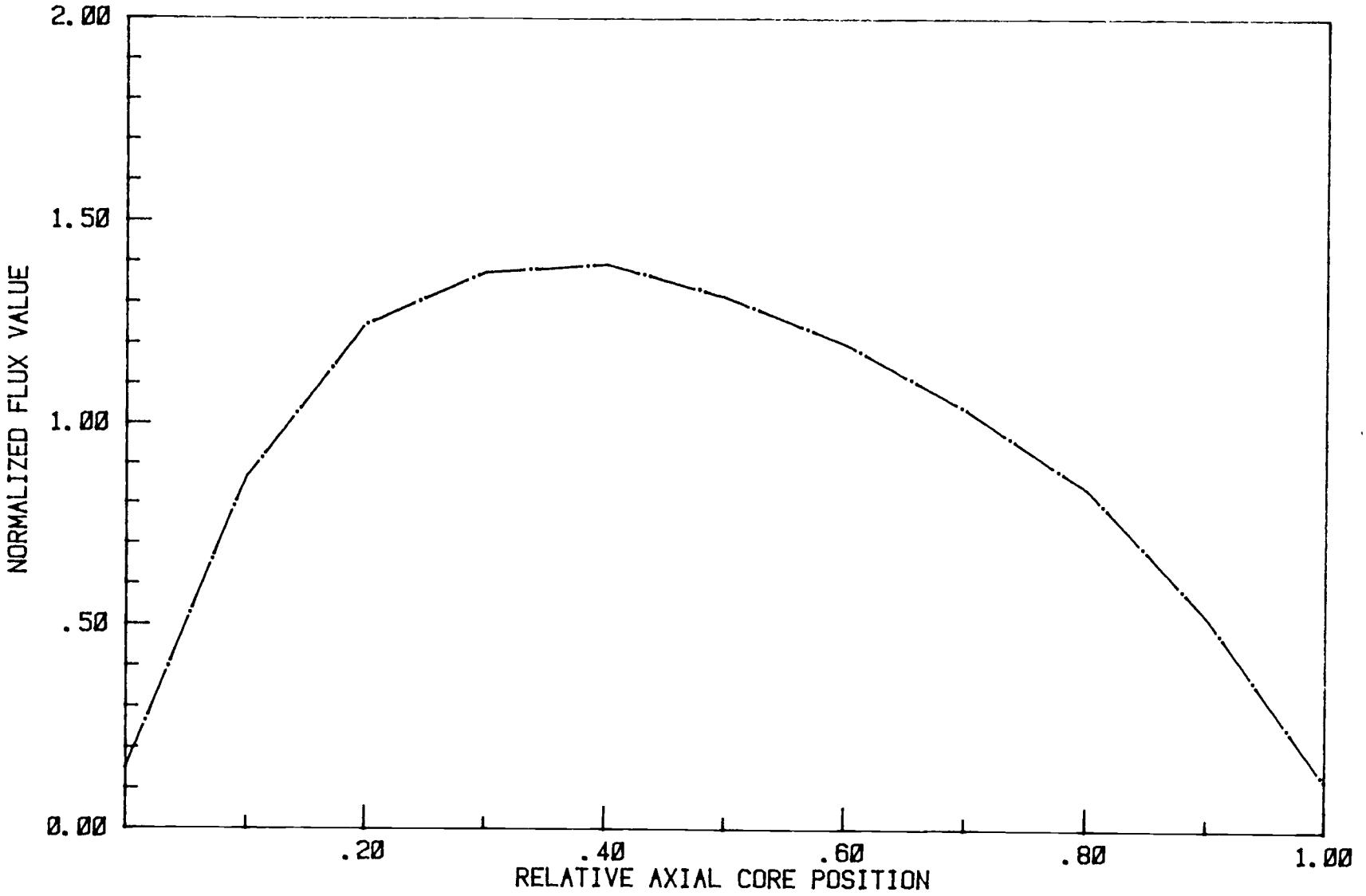


FIGURE 3-2 Axial heat flux profile for core

to unity as required by COBRA-IV. It was recognized that this is an optimized choice, especially with a radial power factor chosen equal to 1.00 and establishes a square core geometry. However, it is maintained that this method yields results suitable for meaningful comparisons and evaluations of the proposed design alternative. Lastly, the term pitch will be used to define the centerline to centerline distance between fuel rods. Probably the most tricky part deals with the establishment of the sub-channel diameters. To assist in the visualization the reader is referred to figure 3-3 which shows both square and hexagonal subchannel layouts. The shaded regions represent the cross-sectional flow areas and can be determined as follows:

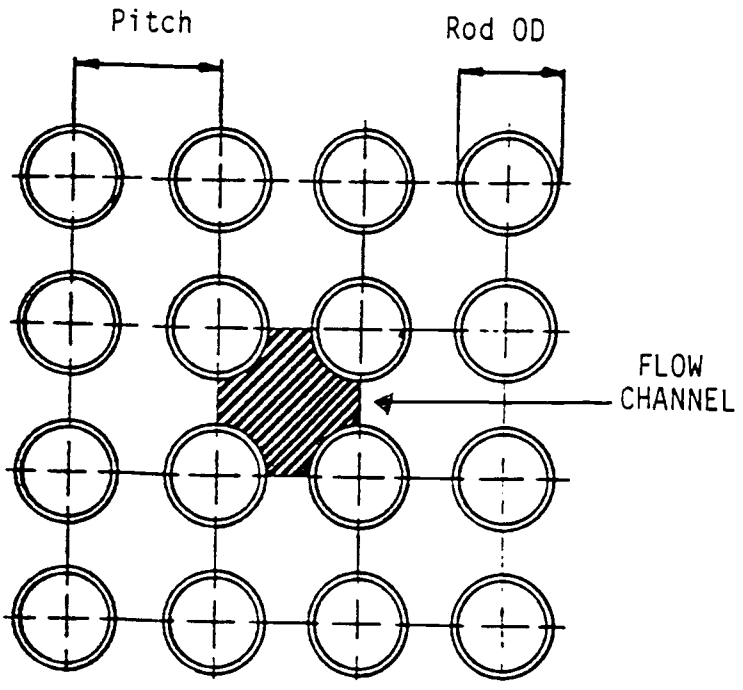
$$\text{Square Array} = \text{Pitch}^2 - \frac{(\text{Pi} * \text{OD})^2}{4.00}$$

$$\text{Triangular Array} = \frac{(\text{Pitch}^2 * \text{Sin } 60^\circ)}{2.00} - \frac{(\text{Pi} * \text{OD}^2)}{8.00}$$

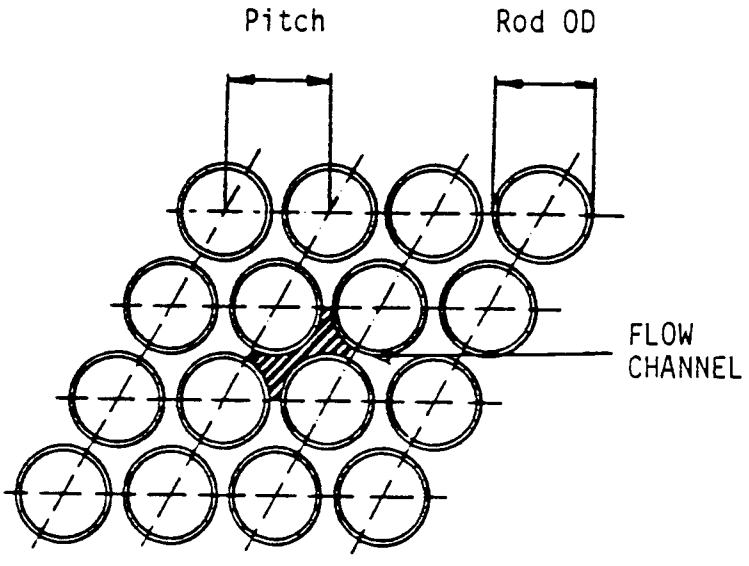
but for our five rod hex array there are two such triangular arrays so:

$$\text{Hex. Array} = (\text{Pitch}^2 * \text{Sin } 60^\circ) - \frac{(\text{Pi} * \text{OD}^2)}{4.00} \qquad \text{(EQN.4)}$$

The subchannels used in the analysis can be



SQUARE ARRAY



HEXAGONAL ARRAY

FIGURE 3-3 Square and Hexagonal arrays

seen in figure 3-4. One note, at various times the term "% Pitch Reduction" will be used in the text. This term references to a reduction of the reference pitch value of 0.4960 inch by the specified amount. [e.g., a 10% reduction is equal to 0.4960 - (0.10)(0.4960) = 0.4464].

Since not all the rods of a 17 x 17 assembly are not fuel but rather guide, structure and control rods, the percentage of these non-heated rods was held at a constant 8.65% throughout the analysis to assure consistency. This same idea was used when computing the presented flow area using a 2.196% factor when called for. These percentages were obtained by using the values obtained for the parameter in question from the FSAR and comparing it with the gross computed factor. The overall presented cross-sectional core area was maintained at a constant value for the hexagonal array and the total number of assemblies was held constant for the square array cases.

The term "hexagonal equivalent" should be interpreted to mean that the pitch along one of the rays of the assembly in the same as for the square array counterpart. If there is persisting

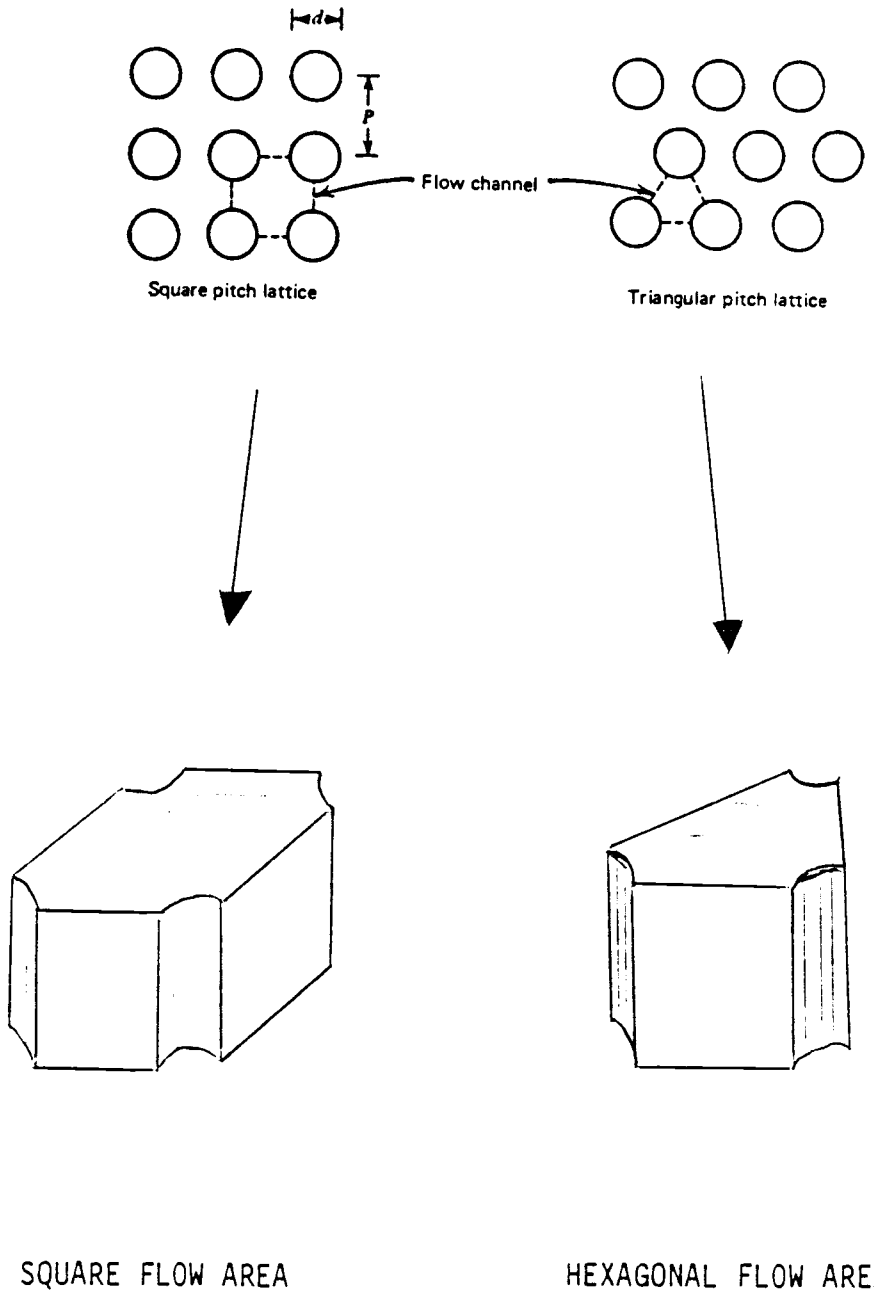
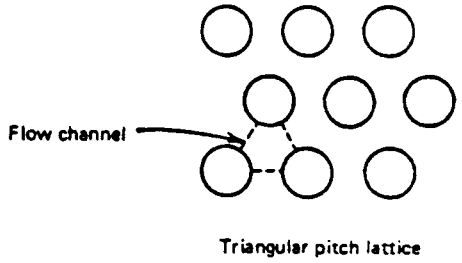
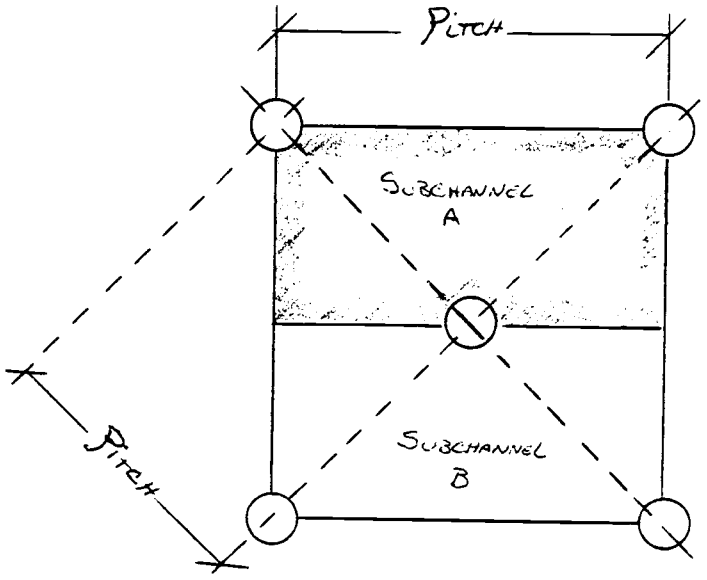


FIGURE 3-4 Assembly subchannel depictions

confusion, please consult to figure 3-5.

In order to assume as much technical accuracy as possible the mass flux and average core heat flux were adjusted as needed accounting for changes in pitch, which directly affected both total flow and heated areas associated with the core. As would be expected with a pitch reduction of any significance, more rods (and fuel too) will be introduced into the core. If the core's overall power rating is to remain constant, then the linear heat generation rate for each rod must be reduced. This increased the DNBR and consequently drives the core and most importantly the hot rod and hot channel, further away from CHF and the attendant boiling. All this adds to the safety of the core and its operation and could lead to either a reduction in core diameter or heighth if previous levels of LHGR are desired.





Effective 5-rod Hexagonal flow channel

FIGURE 3-5 Hex-equilivent development diagrams

3.1.6 RESULTS FROM STEADY STATE ANALYSIS

The primary concern, which was an overriding impetus to look into this topic in the first place, dealt with the attendant pressure drops associated with tight lattice cores. As has been brought out before, excessive pressure drops across the core will eliminate any possibility of a retro-fitable design option for online power stations and therefore reduce this entire effort to merely another "next generation maybe we'll build it" kind of design exercise.

With the above observation a parametric analysis based upon a systematic pitch reduction was clearly mandated. This comparison would be conducted for both square and hexagonal array configurations over some reasonable range, say up to a 20% pitch reduction. The results of that analysis are shown in figure 3-6. The slight discontinuities in the square array data are due to the required array spacing shift (e.g., 17 x 17 to 18 x 18) which was needed to maintain a constant total number of assemblies within the core. The strong divergence of the hexagonal array lead to unacceptable cross core pressure drops which could

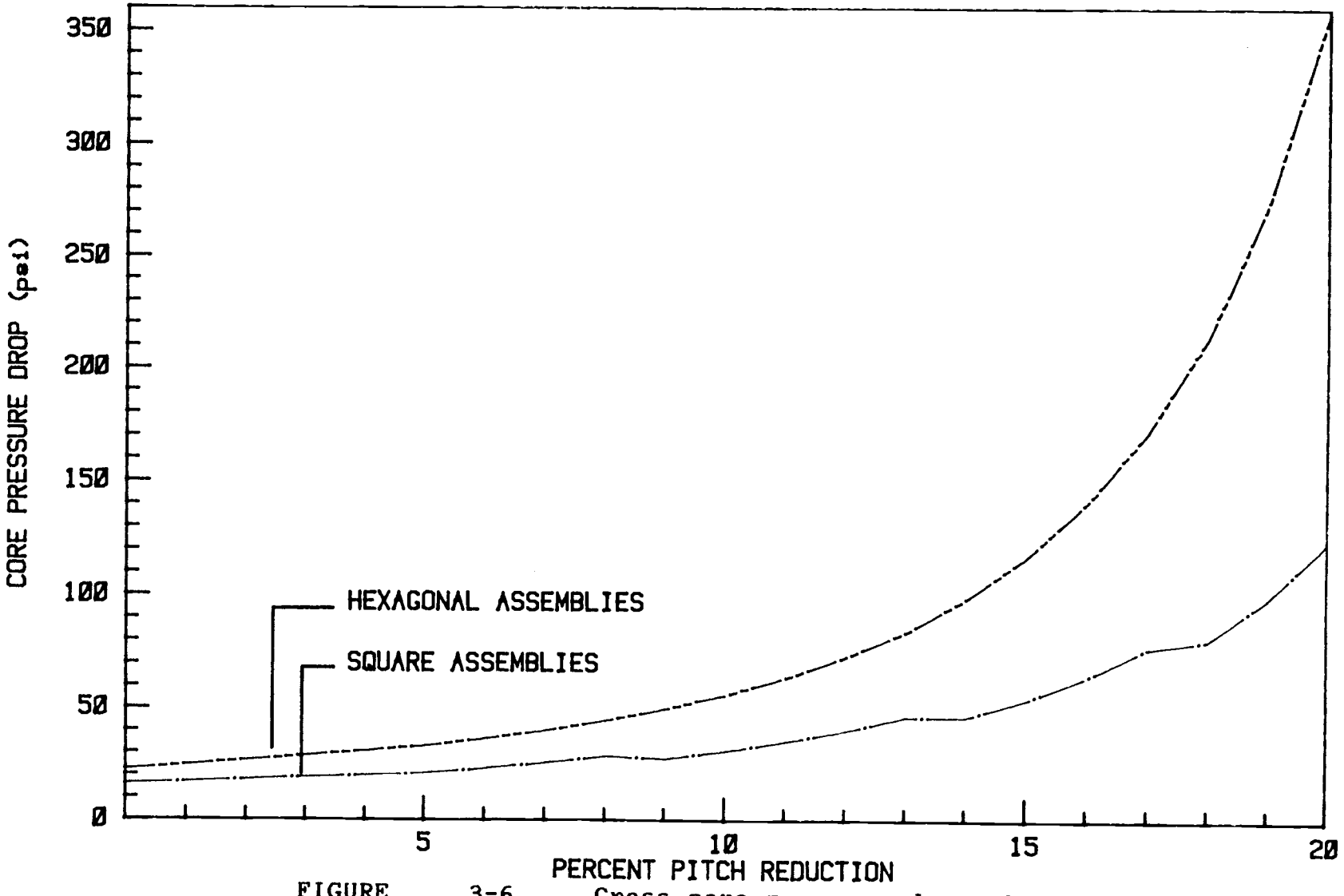


FIGURE 3-6 Cross core pressure drop plots

not be accommodated by flow and core heighth reductions. This factor alone is sufficient to warrant the rejection of the hexagonal array from further analysis in this study. In order to assure the validity of the W-3 correlation, pitch reductions of more than 18% for hexagonal or 20% for square arrays were dropped since the required mass flux would exceed the limitations of the correlation.

Considering these ideas and trying to account for factors like ease of fabrication and overall reasonableness, a 20 x 20 optimized square array was selected for further study. Actually, the results of the neutronic pitch reduction were considered as well but that information is not covered until Section 3.2.

In using the term "optimized square assembly" the author is constructing a 20 x 20 square array assembly with the largest possible pitch which will have the same outer dimensions as the 17 x 17 reference core. This is indeed possible to do as indicated by the fact that the reference assembly measures 8.4260 inches on a side and the optimized design is 8.42599 inches on a side, which seems

close enough for the author. This criteria permits much easier retro-fitting into a given operational core for reasons that are intuitively obvious. As an aside, this 20 x 20 design corresponds to a pitch reduction of ~14.56 %, an overall heat flux of 137.28 E+03 BTU/Ft^2-Hr, and an adjusted coolant mass flux of 3.58 E+06 Lbm/ Ft^2-Hr; all of which lie well within the limitations imposed by the W-3 CHF correlation.

Given the axial flux profile of figure 3-2, the steady state temperature profiles are now offered in figures 3-7 and 3-8 for both the reference and design core configurations. This supports the earlier analysis that suggested that a tighter lattice would in fact improve the inherent safety margins which directly equates to lowering the fuel rod temperatures. This is also supported by the increase witnessed in MDNBR values from 7.049 for the reference core to >10.0 for the 20 x 20 core.

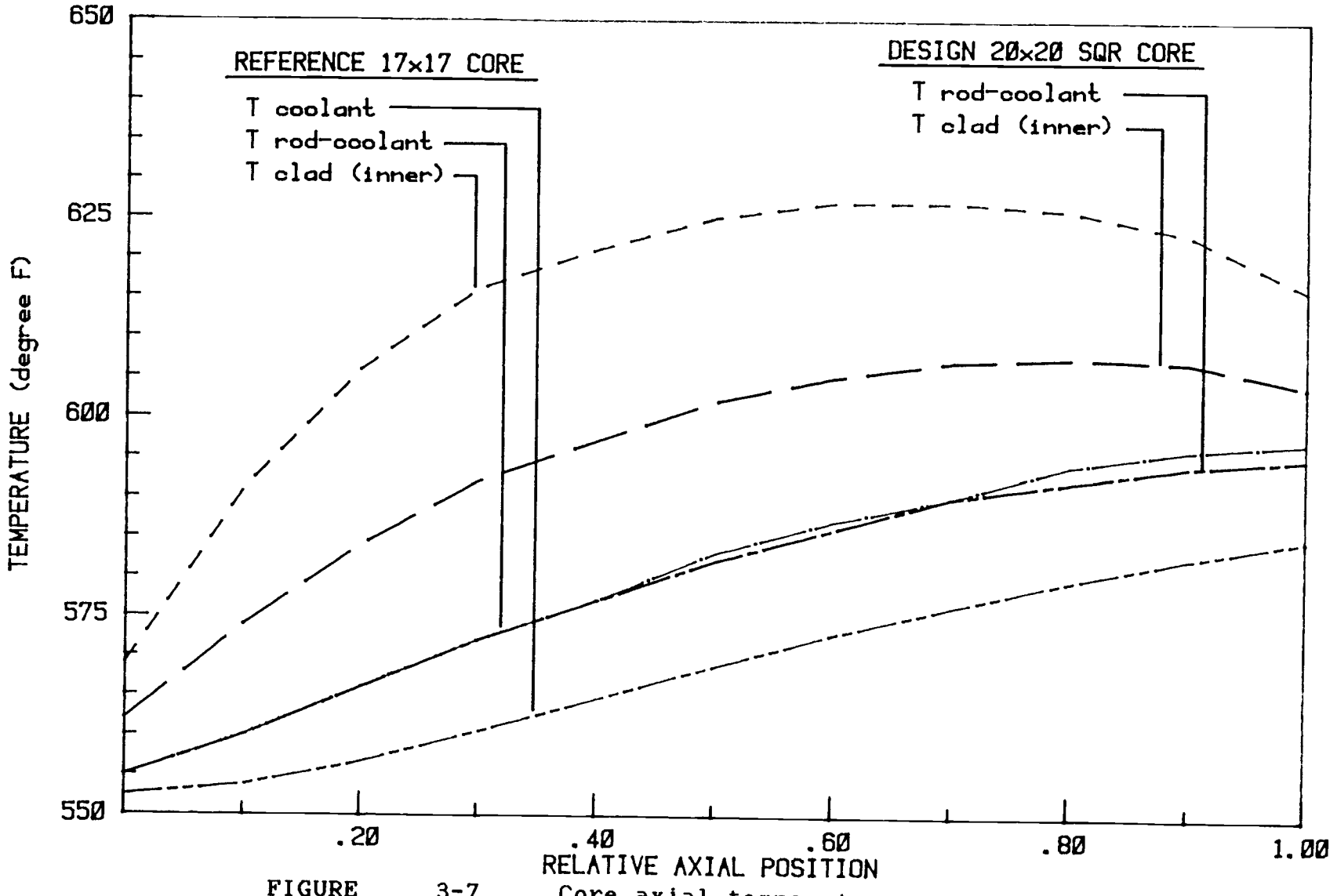


FIGURE 3-7 Core axial temperature profiles

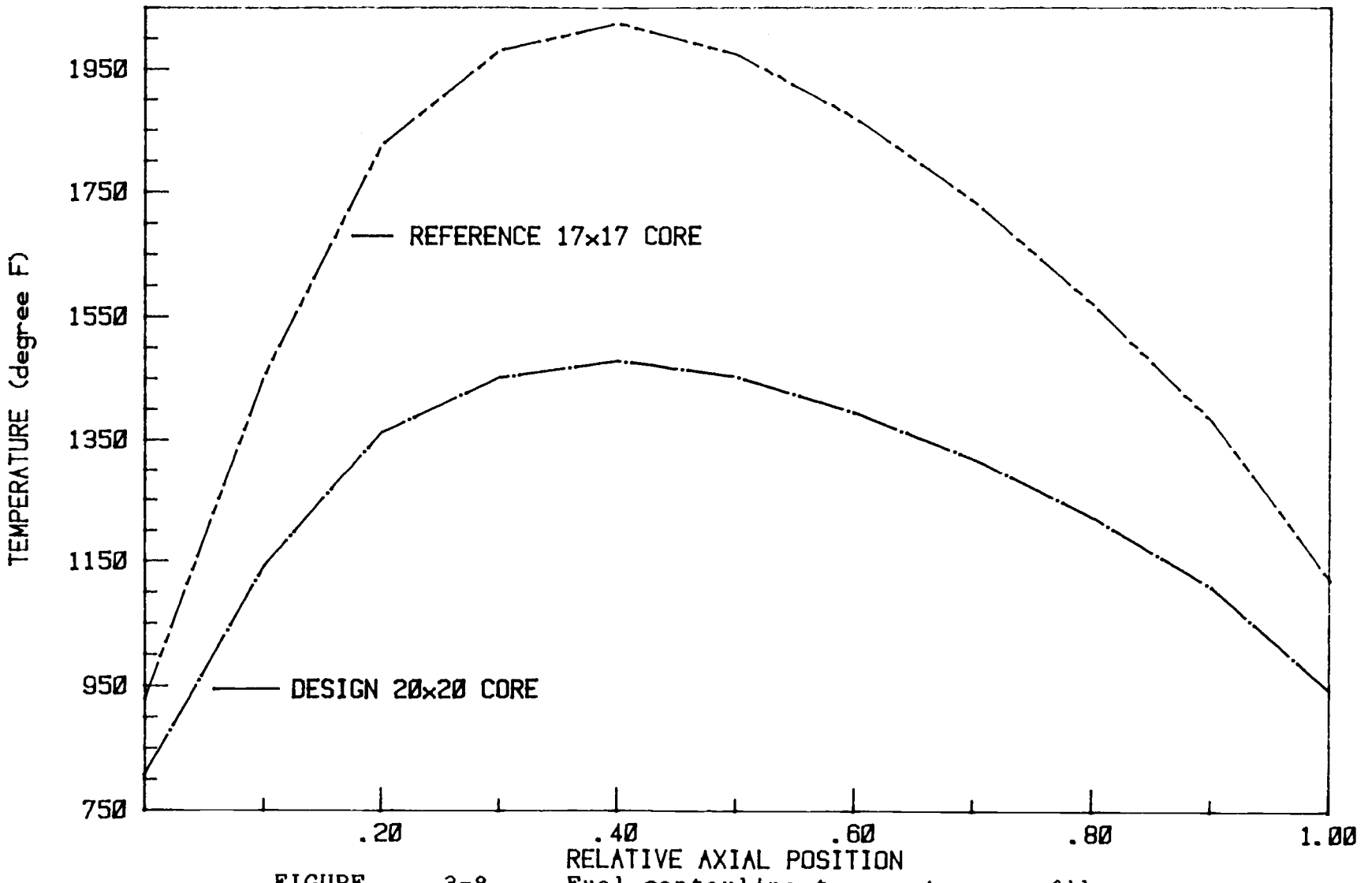


FIGURE 3-8 Fuel centerline temperature profile

3.1.7 RESULTS FROM SELECTED REACTOR TRANSIENTS

Any thermohydraulic study would be incredibly incomplete without a transient behavior analysis. The selected transients should be related to a fairly wide range of normal and abnormal situations as well as rendering a fair appraisial of a prospective design's ability to maintain safe operating conditions, or more bluntly, to assure that a core melt is not incurred. Since the author did not want to be derelict in his responsibilities, the following selected transient events were invesitigated.

3.1.7.1 ACCELERATED HEATING TRANSIENT

In this event the reactor is at full power steady state (FPSS) conditions and experiences an increase of 10 F in coolant temperature within 5.0 seconds after which the temperature remains in the elevated state. This corresponds to a 2 % temperature excursion and could arise from a number of circumstances such as a turbine load reduction event, or a partial loss of condenser vacuum. Either of these events, if severe enough, could

lead to a reactor power reduction or complete shut down.

The results are shown in table 3-1 and table 3-2 for both reference and prototype core designs. Although the results are lowered for the new design as compared to the reference core, the overall behavior is maintained for both pressure and temperature parameters. The DNBR behavior is also normal for this type of transient and carefull inspection of the latter times for this event indicates that DNBR is almost reached its minimum value which appears to be ~6.432 for the 17 x 17 core, and somewhere > 10.0 for the 20 x 20 core. Both of these values are well above the 1.30 floor set fourth in the FSAR for the Trojan plant.

3.1.7.2 ACCELERATED COOLING TRANSIENT

In this event the reactor is again at FPSS conditions when it incurs the direct opposite to the event just described. That is to say a 10 F coolant temperature decrease within 5 seconds. This event could be caused by a major steam rupture on the secondary side of the system.

TABLE 3.1

HEAT UP TRANSIENT RESULTS - 17 x 17 CORE

Time (sec)	Pressure (psi)	Temperatures (degree F) mod.	clad	fuel CL	MDNBR
0	15.99	595.0	625.1	1969.5	7.049
1	16.05	596.5	626.9	1969.8	6.937
2	16.06	598.6	629.3	1970.5	6.797
3	16.05	600.4	631.2	1971.8	6.693
4	16.05	601.9	632.3	1973.4	6.607
5	16.06	603.1	633.7	1975.2	6.528
6	16.05	604.1	634.6	1977.1	6.475
7	16.05	604.5	634.9	1978.9	6.452
8	16.04	604.7	634.9	1980.4	6.442
9	16.04	604.7	635.0	1981.7	6.436

TABLE 3.2

HEAT UP TRANSIENT RESULTS - 20 x 20 CORE

Time (sec)	Pressure (psi)	Temperatures (degree F) mod.	clad	fuel CL	MDNBR
0	52.09	594.5	607.3	1479.9	> 10
1	52.27	596.3	609.3	1480.2	"
2	52.34	598.5	611.5	1481.0	"
3	52.35	600.2	613.0	1482.3	"
4	52.39	601.6	614.3	1483.9	"
5	52.43	602.9	615.6	1485.5	"
6	52.42	603.7	616.4	1487.2	"
7	52.41	604.1	616.7	1488.7	"
8	52.41	604.2	616.8	1490.0	"
9	52.41	604.3	616.9	1491.0	"
10	52.41	604.3	616.9	1491.8	"

The results of this transient are offered in table 3-3 and table 3-4, which also includes the reference core's results for easy comparison. Once again the similarities become evident and analogous comments can be made concerning the overall behavior and safety margins between the two. The general trend for DNBR is an increase to a new steady state value and consequently since it is increasing it is improving the safety margin and no adverse affects are anticipated. Whereas the 20 x 20 core never drops below the value of 10.0, similar behavior is assumed to occur as indicated by the reference core.

3.1.7.3 MAJOR REACTOR COOLANT FLOW REDUCTION

This situation is of much more serious nature than the previous two. Here the coolant mass flux drops to 50% of its original level within 25 seconds and remains at the reduced value. The total transient time for the purpose of analysis is 35 seconds which allows sufficient time for steady state conditions to be reached.

TABLE 3.3

COOL DOWN TRANSIENT RESULTS - 17 x 17 CORE

Time (sec)	Pressure (psi)	Temperatures (degree F) mod.	clad	fuel CL	MDNBR
0	15.99	595.0	625.1	1969.5	7.049
1	15.99	594.8	624.9	1969.5	7.061
2	15.97	594.2	624.2	1969.3	7.106
3	15.95	592.7	622.6	1968.9	7.205
4	15.93	590.6	620.4	1968.1	7.339
5	15.92	588.2	618.0	1966.8	7.482
6	15.93	586.4	616.2	1965.1	7.602
7	15.95	585.4	615.4	1963.3	7.640
8	15.95	585.1	615.2	1961.5	7.658
9	15.95	584.9	615.0	1960.0	7.667
10	15.95	584.9	615.0	1958.7	7.672

TABLE 3.4

COOL DOWN TRANSIENT RESULTS - 20 x 20 CORE

Time (sec)	Pressure (psi)	Temperatures (degree F) mod.	clad	fuel CL	MDNBR
0	52.09	594.5	607.3	1479.9	>10.0
1	52.07	594.3	607.1	1479.8	"
2	52.01	593.6	606.3	1479.7	"
3	51.92	592.0	604.6	1479.2	"
4	51.84	589.8	602.3	1478.3	"
5	51.78	587.4	599.9	1476.9	"
6	51.79	585.6	598.2	1475.1	"
7	51.82	584.9	597.6	1473.4	"
8	51.83	584.6	597.3	1471.8	"
9	51.83	584.5	597.2	1470.4	"
10	51.83	584.4	597.2	1469.4	"

This would most likely be the result of a reactor coolant pump tripping off-line in a multi-loop system, or a large scale rupture in the main coolant line. In the former choice, which is by far the more plausable, the pump would be coasting down in speed and this fly-wheel action would tend to smooth out the suddeness of the transient. This is important since it would reduce, if not eliminate, the "water hammer" that would be associated with an event like this and the subsequent mechanical stresses associated with this phenomenon would thus be reduced considerably.

The results for this abnormal evolution can be seen in tables 3-5 and 3-6. As would be expected, reduced coolant flow results in a diminished ability to remove heat from the fuel by convection with the coolant, which is directly manifested by an elevation of fuel rod temperatures. As a consequence of this, the internal fuel temperatures approach the critical heat flux which preferentially lowers the DNBR, which is depicted in the obtained results. Finally since the coolant is moving slower, the associated kinetic energy is likewise reduced and is seen as a reduction in

TABLE 3.5

REACTOR COOLANT FLOW REDUCTION TRANSIENT
RESULTS - 17 x 17 CORE

Time (sec)	Pressure (psi)	Temperatures (degree F) mod.	clad	fuel CL	MDNBR
0	15.99	595.0	625.1	1969.5	7.049
1	15.28	595.8	626.8	1969.7	6.873
2	14.46	597.1	629.3	1970.3	6.634
3	13.83	598.5	631.8	1971.4	6.425
4	13.50	599.6	633.4	1973.0	6.296
5	13.15	600.6	635.0	1974.8	6.176
6	12.85	601.5	636.5	1976.8	6.069
7	12.55	602.4	637.6	1978.8	6.044
8	12.26	603.3	638.9	1980.8	5.864
9	12.01	604.1	640.3	1982.9	5.771
10	11.76	605.0	641.7	1985.0	5.680
11	11.48	605.8	643.1	1987.1	5.678
12	11.27	606.7	644.6	1989.2	5.504
13	11.04	607.6	645.4	1991.3	5.419
14	10.81	608.6	646.9	1993.4	5.336
15	10.59	609.5	648.4	1995.6	5.253
16	10.43	610.4	649.7	1997.7	5.186
17	10.27	611.1	650.9	1999.9	5.123
18	10.11	611.9	652.1	2001.9	5.062
19	9.96	612.7	653.4	2004.0	5.002
20	9.80	613.5	654.7	2006.0	4.942
21	9.35	615.1	657.8	2008.2	4.799
22	8.92	617.4	658.7	2010.8	4.637
23	8.50	620.0	658.9	2014.1	4.435
24	8.08	623.0	659.0	2017.6	4.246
25	7.69	626.3	659.0	2021.0	4.058
26	7.69	628.2	659.0	2023.8	4.015
27	7.69	629.2	659.0	2026.1	4.002
28	7.69	629.5	659.0	2027.9	3.997
29	7.69	629.7	659.0	2029.3	3.994
30	7.69	629.8	659.0	2030.4	3.991
31	7.69	629.9	659.0	2031.3	3.990
32	7.69	629.9	659.0	2032.0	3.988
33	7.69	629.9	659.0	2032.5	3.987
34	7.69	630.0	659.0	2032.9	3.987
35	7.69	630.0	659.0	2033.2	3.986

TABLE 3.6

REACTOR COOLANT FLOW REDUCTION TRANSIENT
RESULTS - 20 x 20 CORE

Time (sec)	Pressure (psi)	Temperatures (degree F) mod.	clad	fuel CL	MDNBR
0	52.09	594.5	607.3	1479.9	>10.0
1	49.53	595.4	608.6	1480.0	"
2	46.48	596.8	610.5	1480.5	"
3	44.04	598.2	612.5	1481.3	"
4	42.73	599.3	613.5	1482.3	"
5	41.40	600.3	614.5	1483.5	"
6	40.25	601.1	615.6	1484.7	"
7	39.13	602.0	616.7	1485.9	"
8	38.02	602.9	617.9	1487.2	"
9	37.04	603.8	618.9	1488.5	"
10	36.08	604.6	620.0	1489.7	"
11	35.13	605.5	621.1	1491.0	"
12	34.22	606.4	622.3	1492.2	"
13	33.34	607.3	623.4	1493.5	"
14	32.47	608.2	624.6	1494.8	"
15	31.61	609.2	625.8	1496.1	"
16	30.98	610.0	626.8	1497.4	"
17	30.38	610.8	627.8	1498.6	"
18	29.77	611.6	628.8	1499.9	"
19	29.17	612.4	629.8	1501.1	"
20	28.58	613.2	630.8	1502.3	"
21	26.91	614.8	633.2	1503.6	"
22	25.20	617.2	636.3	1505.3	"
23	23.54	619.8	639.8	1507.4	"
24	21.93	622.6	643.0	1510.0	"
25	20.37	625.8	647.1	1513.0	9.609
26	20.27	627.6	648.8	1516.1	9.451
27	20.27	628.4	649.5	1518.8	9.380
28	20.26	628.8	649.9	1521.1	9.339
29	20.26	629.0	650.1	1523.0	9.313
30	20.26	629.1	650.2	1524.4	9.294
31	20.26	629.2	650.3	1525.6	9.280
32	20.26	629.3	650.4	1526.4	9.269
33	20.26	629.4	650.5	1527.1	9.261
34	20.26	629.4	650.5	1527.6	9.255
35	20.26	629.4	650.5	1528.0	9.250

cross-core pressure drop which is on the same order of magnitude as the mass flux reduction.

3.1.7.4 POWER EXCURSION EVENT

This is by far the most dangerous and potentially damaging event of all those evaluated and the least likely to occur. Almost exclusively associated with a rod ejection event, it is included because this type of transient is usually required in the reactor licensing process. In this analysis a total time of 5.0 seconds was selected with time steps of 1/20 of a second. Power rise time was ~0.380 seconds to a level of 150% of normal full power, a resident time of 1.50 seconds followed by a drop in power to 0% power in ~0.150 seconds. This extended time at the elevated state may perhaps seem artificially prolonged with respect to a real time event, but it does permit a better analysis through the benefit of reasonable exaggeration. Conceiveably, if the rod was impulsed with enough force into its fully extended position, it could momentarily stick before being driven back into the core. Based on this, the method of analysis should be acceptable, besides if

any error is introduced, it is to the side of conservativeness which is always acceptable.

The findings of this analysis is displayed in table 3-7 and table 3-8 where the core was initially at FPSS conditions. We see that the pressure drop rises to a maximum and then partially subsides to a new level that is somewhat higher than the initial value. As would be expected, the temperature's characteristic behavior closely tracks the power history. This heat-up and cool-down action also produces a characteristic DNBR plot which is also shown. Again as in all of the previous transients, the maximum attained values never exceed the prescribed safety limitations. In addition to this, the results indicate a superior safety margin for the 20 x 20 core than that of the current 17 x 17 core in use. This is primarily due to the derated value of the linear heat generation rate for the new design which would be required to maintain the overall thermal rating commensurate with the reference core.

TABLE 3.7

POWER EXCURSION TRANSIENT
RESULTS - 17 x 17 CORE

Time Step	Pressure (psi)	Temperatures (degree F) mod.	clad	fuel CL	MDNBR
0	15.99	595.0	625.1	1969.5	7.049
2	16.00	595.0	625.6	1979.8	6.965
4	16.02	595.3	627.0	2002.5	6.771
6	16.02	595.7	628.5	2027.3	6.575
8	16.03	596.4	630.1	2052.3	6.407
10	16.03	597.2	631.6	2076.9	6.265
12	16.03	597.9	632.9	2100.8	6.148
14	16.02	598.7	634.0	2123.7	6.051
16	16.02	599.3	635.0	2145.4	5.973
18	16.02	599.9	635.8	2165.7	5.910
20	16.02	600.5	636.5	2184.4	5.860
22	16.01	600.9	637.0	2201.5	5.821
24	16.00	601.2	637.2	2211.2	5.819
26	15.98	601.3	636.2	2200.8	5.953
28	15.92	601.0	633.5	2162.9	6.278
30	15.92	600.1	629.9	2112.3	6.725
32	15.91	598.7	626.0	2056.4	7.223
34	15.91	596.9	622.1	1997.9	7.744
36	15.91	595.0	618.5	1937.8	8.279
38	15.91	592.9	615.0	1877.1	8.824
40	15.91	590.9	611.8	1816.7	9.381
42	15.92	588.9	608.9	1756.9	9.945
44	15.92	587.1	606.1	1698.4	>10.0
46	15.92	585.4	603.6	1641.5	"
48	15.92	583.9	601.3	1586.5	"
50	15.93	582.4	599.1	1533.5	"
52	15.93	581.0	597.0	1482.6	"
54	15.93	579.8	595.0	1434.0	"
56	15.94	578.6	593.2	1387.6	"
58	15.94	577.4	591.5	1343.4	"
60	15.94	576.4	590.1	1301.5	"
62	15.94	575.3	588.5	1261.7	"
64	15.94	574.4	586.9	1224.0	"
66	15.95	573.4	585.4	1188.4	"

TABLE 3.8

POWER EXCURSION TRANSIENT
RESULTS - 20 x 20 CORE

Time Step	Pressure (psi)	Temperatures (degree F) mod.	clad	fuel CL	MDNBR
0	52.10	594.5	607.3	1479.9	>10.0
2	52.12	594.6	607.6	1487.3	"
4	52.17	594.9	608.5	1503.8	"
6	52.19	595.6	609.7	1521.6	"
8	52.21	596.4	611.0	1539.3	"
10	52.22	597.3	612.2	1556.4	"
12	52.22	598.1	613.3	1572.9	"
14	52.22	598.9	614.0	1588.4	"
16	52.22	599.5	614.6	1602.8	"
18	52.22	600.1	615.2	1616.1	"
20	52.22	600.6	615.7	1628.2	"
22	52.21	601.0	616.1	1639.1	"
24	52.19	601.2	616.3	1644.9	"
26	52.12	601.2	615.6	1635.3	"
28	52.01	600.6	613.9	1606.4	"
30	51.92	599.3	611.5	1568.8	"
32	51.88	597.5	608.5	1528.1	"
34	51.86	595.3	605.6	1486.2	"
36	51.85	593.1	602.7	1443.9	"
38	51.85	590.9	599.9	1401.8	"
40	51.85	588.8	597.4	1360.3	"
42	51.85	586.9	595.0	1319.8	"
44	51.85	585.1	592.8	1280.6	"
46	51.85	583.5	590.8	1242.8	"
48	51.85	582.0	589.0	1206.5	"
50	51.85	580.6	587.2	1171.9	"
52	51.85	579.2	585.6	1138.9	"
54	51.85	578.0	584.0	1107.5	"
56	51.84	576.8	582.6	1077.7	"
58	51.84	575.7	581.2	1049.4	"
60	51.84	574.7	579.9	1022.7	"
62	51.84	573.7	578.8	997.5	"
64	51.84	572.7	577.6	973.7	"
66	51.84	571.8	576.4	951.3	"

3.1.8 OBSERVATION ON COBRA-IV RESULTS

This code provided adequate analysis in the context of a primary level evaluation code in the area of thermohydraulics.

The results concerning the temperature and DNBR values as related to the transient analysis are a bit misleading in so far that the negative temperature reactivity coefficient effects are not incorporated into COBRA-IV. If they were we would see reactor power turn (ie: reduce) when coolant temperature rose. The usual outcome for transient cases offered here would be lower final temperatures associated with the fuel and coolant thus once again increasing the safety margins of concern.

Nonetheless, the results obtained from this code are acceptable even if they are inherently conservative in nature. It is particularly hearting to see that even with the extra conservative analysis, the safety margins are well above the critical values obtained from this analysis. This insures that the core would not be in peril under either steady state or abnormal

conditions as examined herein. This seems to be especially true for the proposed 20 x 20 fuel assembly of this study.

3.2 NEUTRONIC DESIGN ASPECTS

In this area, some fundamental concepts will be revisited and explained. Topics such as the basic neutronic economy cycle, conversion ratio and fuel to water ratios will be reviewed. Following this, we will quickly review the basic decay schemes and equations that are solved in LEOPARD (a pin-cell neutronic cross sectional & burn-up calculational computer code) and the concept of reactivity. This will then lead us to examine the obtained results and discuss the differences between the reference and proposal cases.

3.2.1 NEUTRON LIFE CYCLES

At the very heart of any nuclear reactor is the neutron life cycle, for without neutrons, the whole nuclear energy endeavor would be nothing but a wispy dream forever out of reach. However, hope springs eternal and neutrons do exist and also

follow a rather crude birth-to-death life cycle.

All neutrons are born at high energy (in MeV range) and are termed "fast." As such there are only three ways which reactors "lose" neutrons; direct escape in which the neutron simply travels through all materials until it is outside the established control volume, be that of the fuel pin, assembly, core, or building; absorption or capture which albiet different in their actual process, still result in the immobilization of the neutron; and for those fortunate few, self-annihilation or decay since neutrons have a half-life of 10.5 minutes and subsequently disintegrate into electrons and photons.

Frequently, the above catagories are further subdivided into fast and thermal energy zones to differentiate between reaction probabilities involved. As one may imagine, the various combinations of these elements can quickly become difficult to conceptualize and keep track of, which is why a symbolic life track schematic is often developed. One such depiction is offered in figure 3-9 and is given in lieu of further discussion on this line. This diagram also shows the basic

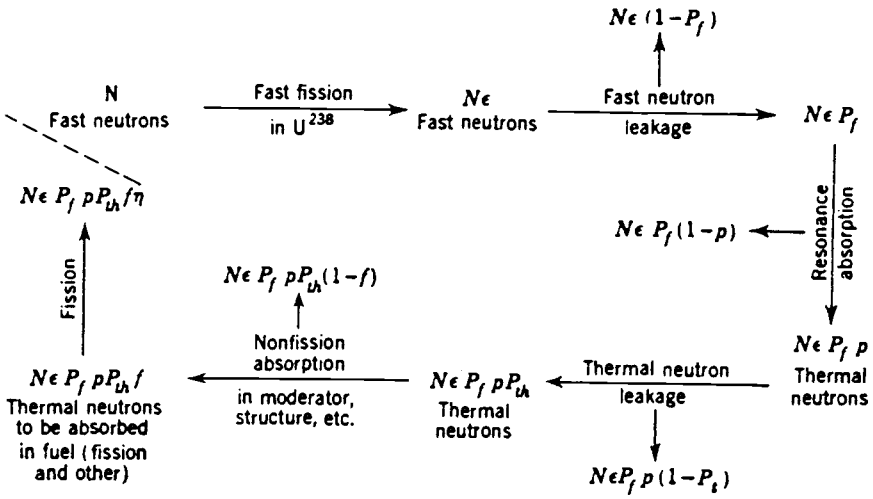


FIGURE 3-9 Neutron life cycle

relationships between the competing elements and necessarily contains the definitions of the neutron balance factors.

If the life cycle is the heart, then the six factor formula is the backbone of the neutronics field. This handy, compact formula contains all of the essential information pertaining to a given core design and is usually written as follows.

$$k_{eff} = \epsilon * Pf * Pth * p * \eta * f \qquad \text{(EQN. 5)}$$

where:

k_{eff} = the effective multiplication factor

ϵ = fast fission factor

$$= \frac{\text{\# of fast neutrons born from all fission}}{\text{\# of fast neutrons from thermal fission only}}$$

Pf = fast non-leakage factor

$$= \frac{\text{\# of fast neutrons failing to leak while slowing}}{\text{\# of fast neutrons that started to slow down}}$$

Pth = thermal non-leakage factor

$$= \frac{\text{\# of thermal neutrons that fail to leak out}}{\text{\# of thermal neutrons that started}}$$

p = resonance escape probability

$$= \frac{\text{\# of neutrons that reach thermal energy}}{\text{\# of thermal neutrons that started}}$$

η = neutron utilization factor

$$= \frac{\text{\# of fast neutrons born from thermal fission}}{\text{\# of thermal neutrons absorped by the fuel}}$$

f = thermal utilization factor

$$= \frac{\text{\# of thermal neutrons absorped into the fuel}}{\text{\# of thermal neutrons absorped within the core}}$$

And if we neglect the losses due to the leakage within the core, we then have a core of infinite radius and leads us to the four factor formula:

$$k_\infty = e * \eta * p * f \qquad \text{(EQN. 6)}$$

where:

e , η, p, f

carry the same definitions as above.

These various factors can be reduced further into relationships that are expressed in terms of macroscopic cross-sections which are, in fact, expressions of specific event occurence probabilities.

When first investigating a proposed design for a core, one assumes a reasonable value for material buckling (i.e. 1.00 E-6) and neglects the effects of leakage as well. Neglecting the leakage terms is permitted in preliminary analysis since these terms are strongly dependent upon specific design parameters unique to a given core design proposal. This then simplifies the analysis to indicate a design suitable for further consideration, hence the values of k infinity (k_∞) serve an important function in this process. Whereas this thesis is not strictly a neutronic based investigation, the analysis in this area is of the first order only. That is to say that the neutron analysis is employed to indicate feasibility and viability

rather than supplying in depth detailed aspects to any particular design configuration.

3.2.2 THE CONVERSION RATIO

The conversion ratio (CR) is of major concern within the context of this thesis and weights heavily in the design considerations of the next generation of reactor as indicated by its preeminence in recent published papers addressing this subject. In short, it is a direct measure of a reactor's performance in the area of fuel generation and utilization. Most often it is defined as:

$$CR = \frac{\text{new fissile atoms produced by absorption}}{\text{total fissile atoms consumed from initial fuel}}$$

New fissile atoms are produced by the absorption of a neutron by a fertile atom, or by atom that will become fertile or fissile by subsequent decay or transmutation. Since there are only a limited number of either fertile or fissile atoms, the number of available paths are somewhat limited. To maintain continuity, a fissile atom is

one that has the ability to undergo fission process upon the absorption of a neutron where a fertile atom becomes fissile upon the absorption of a neutron.

If the conversion ration is great than 1.00, then the reactor is said to breed fuel. That is to say, it will create more fissile fuel material than it will consume which is a feature that is unique to the nuclear process.

Early generation power reactors had conversion ratios of 0.35 to 0.45 and current on-line power reactors are ranging from 0.50 to 0.63 or so. Our desire is to raise this level to one ranging from 0.80 to 0.95 or higher for retro-fitted cores thus reclassifying them as "converters" and if possible to surpass the 1.00 barrier and make low level breeding possible.

3.2.3 MODERATOR TO FUEL RATIO

Water is a surprising and unique substance. It is the only normal material that expands when frozen, exhibits good thermodynamic properties as well as occuring in copious quantities and most importantly is an absolute necessity to sustain life as we know it. This remarkable material is also of profound importance to the way nuclear power is generated. Granted, there are a number of designs that employ gasses and molten salts, but in large part the nuclear energy generation capacity throughout the world is directly likened to light water reactors. The distinction is made between light water which is normal H_2O and heavy water which is D_2O since some reactors also employ heavy water in their processes.

In regards to nuclear operations, we get double duty out of water. Since it is 2/3 hydrogen, it is an excellent moderator. Recall that an ideal neutron moderator substance is one that has the same atomic mass as the neutron so that when a collision occurs, most if not all of the kinetic energy may be transferred directly to the other particle. This is indeed the case where

a hydrogen atom is less than 0.090% lighter than a neutron. Indeed, the only other substance that would make a better moderator would be pure monotonic hydrogen gas alone, but that is entirely too expensive and difficult to handle in the quantities that are needed for power production. On the other side of the coin, water performs as an adequate coolant transport medium which is the second function as referenced above. While it is true that there are other substances that far surpass water in this area, the fact that water is plentiful (therefore cheap), and that the heat transport mechanisms are well understood tends to mitigate its shortcomings to tolerable margins. Add to this water's outstanding moderating characteristics and we can begin to see why the LWR is such a popular design.

Conventionally defined, the moderator to fuel ratio is the ratio of hydrogen atoms to fuel atoms where fuel is confined to U-235 and U-238 atoms. Often times these quantities are given in terms of number densities, especially when plotted in publications. As would be expected in the context of an array of UO-2 fuel rods and water, this ratio is a function of the densities of both materials in

addition to the actual geometries. Whereas the temperature changes will affect the densities, atomic ratios are much preferred to volumetric ratios for use as a figure of merit during the design process to assume consistency in cross comparisons.

In selecting a moderator to fuel ratio for a partially enriched uranium reactor, two competing neutronic phenomenon become important to consider. First is the fraction of neutron absorptions occuring in the fast and resonance energy ranges. With an increase in the moderator to fuel ratio, the fraction of neturons that reach thermal energy increases while the fraction of epithermal neutrons absorptions within the fuel decreases. In a partially enriched fuel matrix the vast majority of epithermal absorptions are capture events within fertile material, say U-238, and consequently have no effect upon neutron multiplication processes. Whereas the fissile material in the fuel is preferentially the largest absorber of thermalized neutrons within the system, an increase in neutron thermalization will necessarily increase the fraction of the all neutrons that lead to thermal fission. If we were to propose that all neutrons

that attain thermal energies were absorbed into the fuel, then increasing the moderator to full ratio would result in a strictly linear increase of the infinite multiplication factor for the entire system, but such is not the case. The second factor competes directly with the first and consequently they play off each other to a large degree. Increasing the moderator to fuel ratio leads to more neutrons that reach thermal energy levels, but with this comes a larger probability of parasitic absorption of neutrons by the moderator, most notably by the oxygen component. This then acts to decrease the fraction of thermal neutrons that cause fission and in addition seems to lower the infinite multiplication factor as well.

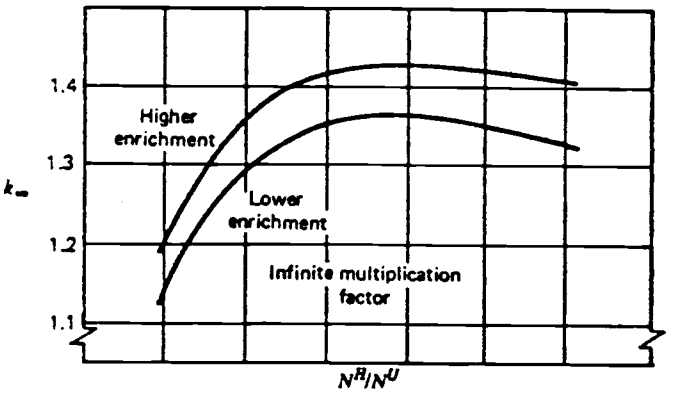
At low moderator to fuel ratios, the first effect becomes dominant and subsequent increase in moderator to fuel ratio leads to an increase in infinite multiplication. However, as the ratio is increased the thermalization process tends to saturate so that thermal neutron captures occuring in the moderator will begin to offset the reduction in the epithermal U-238 captures. Once again the competition between these two effects becomes significant and causes the characteristic variation

of the infinite multiplication factor as shown in figure 3-10 [38].

One item needs to be pointed out at this time with respect to the diagram just referenced. If a proposed design falls to the left of the apex of these curves, it is referred to as undermoderated while designs to the right of the peak are considered overmoderated.

All practical LWR reactors are designed to be undermoderated primarily on the basis of operational safety concerns. In the United States, the regulating licensing agency, the U.S. Nuclear Regulatory Commission, has mandated by law (ie: 10 CFR 50) that all reactors will be undermoderated at all times.

This inherent safety feature stems from the fact that if a reactor is undermoderated, it will have a negative temperatures reactivity coefficient associated with the moderator. Since the thermal expansion coefficient of water if approximately two orders of magnitude than that of either the fuel or structural material, a change in temperature in the moderator will have a preferentially larger effect



Effect of fuel enrichment on neutron multiplication characteristcs.

FIGURE 3-10 Enrichment - Moderator chart

upon reactivity than anything else. To illustrate, an increase in temperature of water causes its specific density to decrease, this in turn reduces the dynamic moderator to fuel ratio which then translates into a negative reactivity insertion. The exact opposite result occurs if the reactor is overmoderated. Consequently, for an undermoderated reactor, an incremental increase in moderator temperature in effect "turns" power to a lower level and establishes an important degree of inherent reactor safety. Conversely, for an overmoderated reactor the same incremental temperature change in the moderator will add positive reactivity and push up the reactor power level which will then cause the temperature to increase further and hence the situation quickly escalates to a point where control of the reactor is completely lost. This exact situation is what occurred earlier this year at the RBMK-I reactor in Chernobyl which is located within the Soviet Union. Although that type of reactor is not of a light water design, it was (and the remaining reactors of its type are still) operated in the overmoderated region.

Another strong reason to go for the

undermoderated design is linked to fuel cycle economics. Increasing the moderation decreases the ratio of U-238 captures to U-235 fissions, which is to say that the conversion ratio is degraded. In addition to this, the infinite multiplication factor begins to drop and must be compensated for by raising the level of initial enrichment required which is in turn more expensive.

3.2.4 METHOD OF EVALUATION

In order to conduct a reasonable neutronic analysis for the core designs suggested by the thermohydrolic analysis discussed in Section 3.1, the Lifetime Evaluating Operations Pertinent to the Analysis of Reactor Designs, or better known as LEOPARD, was the computer modeling code selected. This is a pin-cell model type code that was developed by Westinghouse Atomic Division and is based upon the Modified MUFT-SOFOCATE model. One strong advantage of LEOPARD is that with only a very basic description of a desired core design proposal, a reasonably accurate analysis can be performed to enable a first and occasionally second level evaluation of a particular designs merits.

The code determines energy dependent cross sections, thermal and non-thermal spectrum calculations, as well as performing an overall neutron balance. In addition to this, the code may perform these calculations as a function of burn-up. For a detailed explanation of the code the reader is referred to references 39 and 40.

Two items need mentioning concerning the operation of this code at this time. First is that this code has the ability to temperature convert various parameters, most especially the cross-section that are of influence to the analysis. This feature greatly reduces the error that would be indigent to the analysis if not performed since all cross-sections are expressed at a reference energy, normally 0.653 ev, and doppler broadening substantially effect these values for the temperatures associated with normal reactor operations. The second concerns for method that is employed to perform the burn-up calculations. LEOPARD uses a Runge-Kutta numeric solution scheme to perform the forward projections used in the burn-up calculations. To assure an accurate representation the initial steps would be small to

allow the establishment of an accurate trend-history that permits better advance projections.

In regard to input parameters for LEOPARD, the values can be extracted from the basic design information that was set forth in Section 3.1. For the benefit of improved accuracy in the analysis, the resonance temperature used for analysis was obtained from a linear average of fuel centerline temperatures for the 17 x 17 reference core. All other input parameters are self-explanatory as set forth in the input coding sheets, copies of which can be found in Appendix C.

While this code is a pin-cell model, a composite core evaluation would require a bundle-cell type of evaluation. Codes like 2-DB or PDQ for instance, would render whole core performance results. However, the LEOPARD code is sufficient for the purposes of this investigation. Clearly, to advance the design beyond this stage it will necessarily require analysis with one of these other more advanced codes.

One item that remains to be discussed prior to

exploring the results which is how LEOPARD evaluates the moderator to fuel ratios. In this code the moderator to fuel ratio is termed water to metal ratio wherein it converts the uranium contained in the oxide fuel into the equivalent volume that the uranium would occupy if it was in a metallic form. It does this for plutonium and thorium as well. In addition to this, the code will correct for the density changes that occur in the water coolant as a result of the increased operational temperatures involved. This is denoted with the prefix of "HOT" to the ratio of concern when the affected parameters appear in the output. Consequently, the "HOT-CELL" water to metal ratio values are used for the moderator to fuel ratio evaluations. It way be recalled that volumetric ratios were described to be inferior to atomic ratios in Section 3.2.3, but the conversion into metallic equivalencies mitigates this argument to a large extent. Hence with the accepted degree of rigorous analysis that this code may offer, the water to metal ratio is a satisfactory figure of merit as exposed to the atomic ratio.

3.2.5 RESULTS OF THE CORE NEUTRONIC EVALUATIONS

As was done for the thermohydraulic preliminary analysis, the retention of the square and hexagonal lattice arrays has been performed. This is done even though the hexagonal option was dropped in the thermohydraulic analysis on the basis of unacceptable pressure loss characteristics for the purpose of illustrative comparison.

When examining the effect that U-235 enrichment has upon the moderator to fuel ratio, results indicated a slight rise in value with the increasing of the fuel enrichment level. Interestingly, increasing the level of partial U-235 enrichment leads to an increase in the moderator to fuel ratio. These plots may be a bit contradictory to what may be indicated by intuition so a bit of explanation is in order.

The preferential increase in enrichment translates into a replacement of U-238 atoms with U-235 atoms. This means that there is a slight decrease in the overall density of the equivalent uranium metal value. It is this decrease, albiet very slight, that results in a slight increase in

the water to metal ratio. Although at a first glance the increase would appear significant, a carefull examination of the values for the water/metal ratio shows that the increase is indeed quite small. In fact when considering the range of enrichment (2-15%) the attending change in the ratio can be considered negligible. However, what is significant is that in comparing the 20 x 20 square lattice with the 20 x 20 hexagonal equivalent lattice, the values for the water/metal ratio are markedly different (~1.25 vs ~0.83).

As would be anticipated, the hexagonal lattice ratio indicates less moderation occurs there than in the square lattice arrangement. Naturally, with less moderation occuring, the neutron flux is hardened which is to say that the average neutron energy is shifted towards the epithermal range. The associated effects of this were discussed in Section 3.2.3 above.

Along more traditional lines we see from figure 3-11 the results that confirm our expectations when partial enrichment levels of U-235 were increased. Clearly, as we introduce a larger inventory of fissile fuel into the core, the

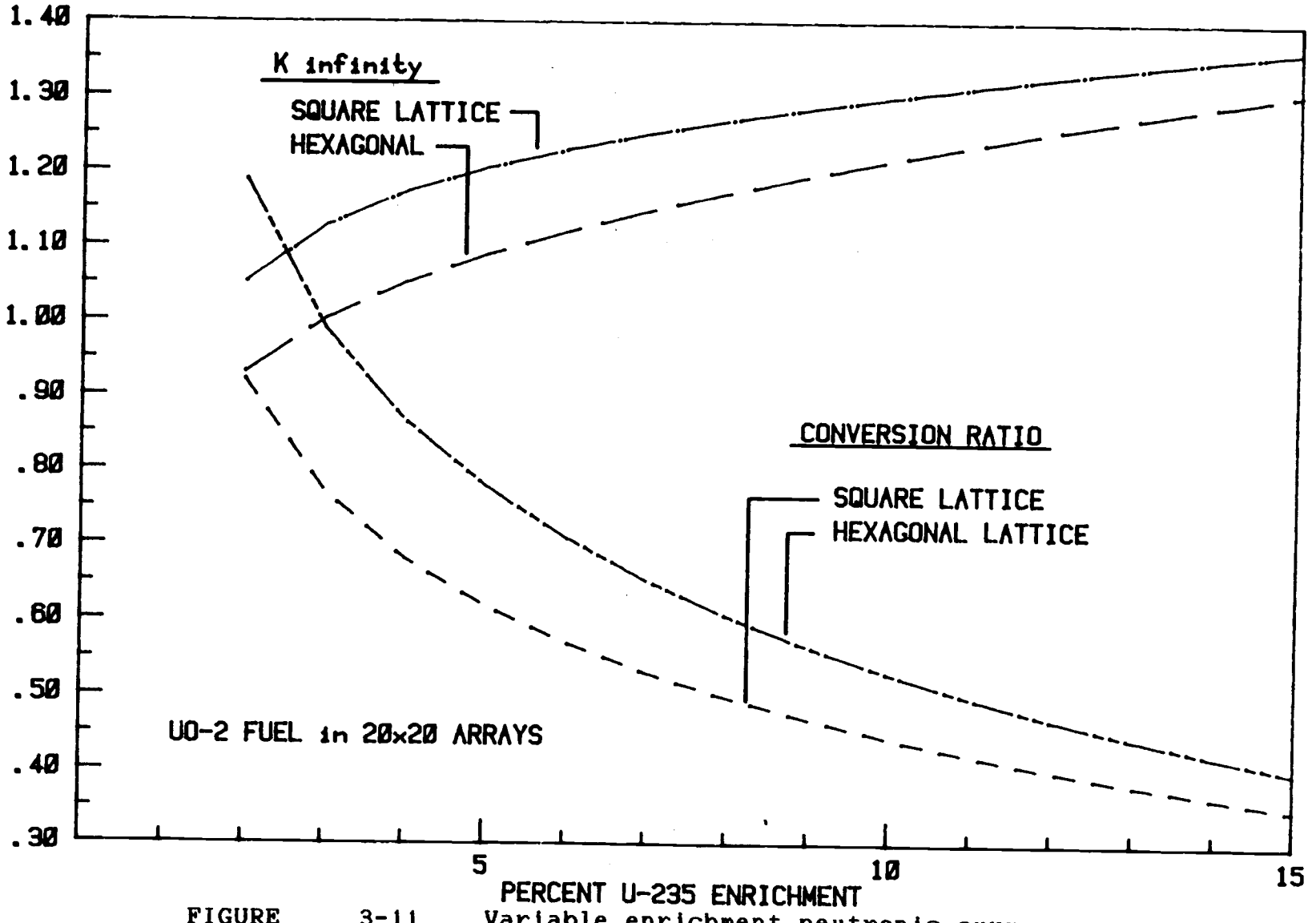


FIGURE 3-11 Variable enrichment neutronic curves

infinite multiplication factor rises and the conversion ratio diminishes. This plot plainly shows the inverse relationship between the infinite multiplication factor and the conversion ratio. Where the hexagonal lattice may exhibit a superior conversion ratio behavior, it has inferior performance in the area of supplying adiquate excess positive reactivity as compared to the square lattice for a given enrichment level. These plots indicate that while the 20 x 20 design as now envisioned can not become a breeder, it certainly can become a very good converter.

The other aspect that was of some importance to examine was the effects that pitch reduction would have upon the neutronics of the core. As illustrated in figure 3-12 we see that tightening the lattice tends to enhance the conversion ratio at the expense of the infinite multiplication factor. It also indicates that the square lattice offers a better blend of the combined characteristics for moderate pitch reductions as compared to the hexagonal lattice alternative. As would be expected, the water/metal ratio decreases with the tighting of the lattice as clearly indicated in figure 3-13. Combining this

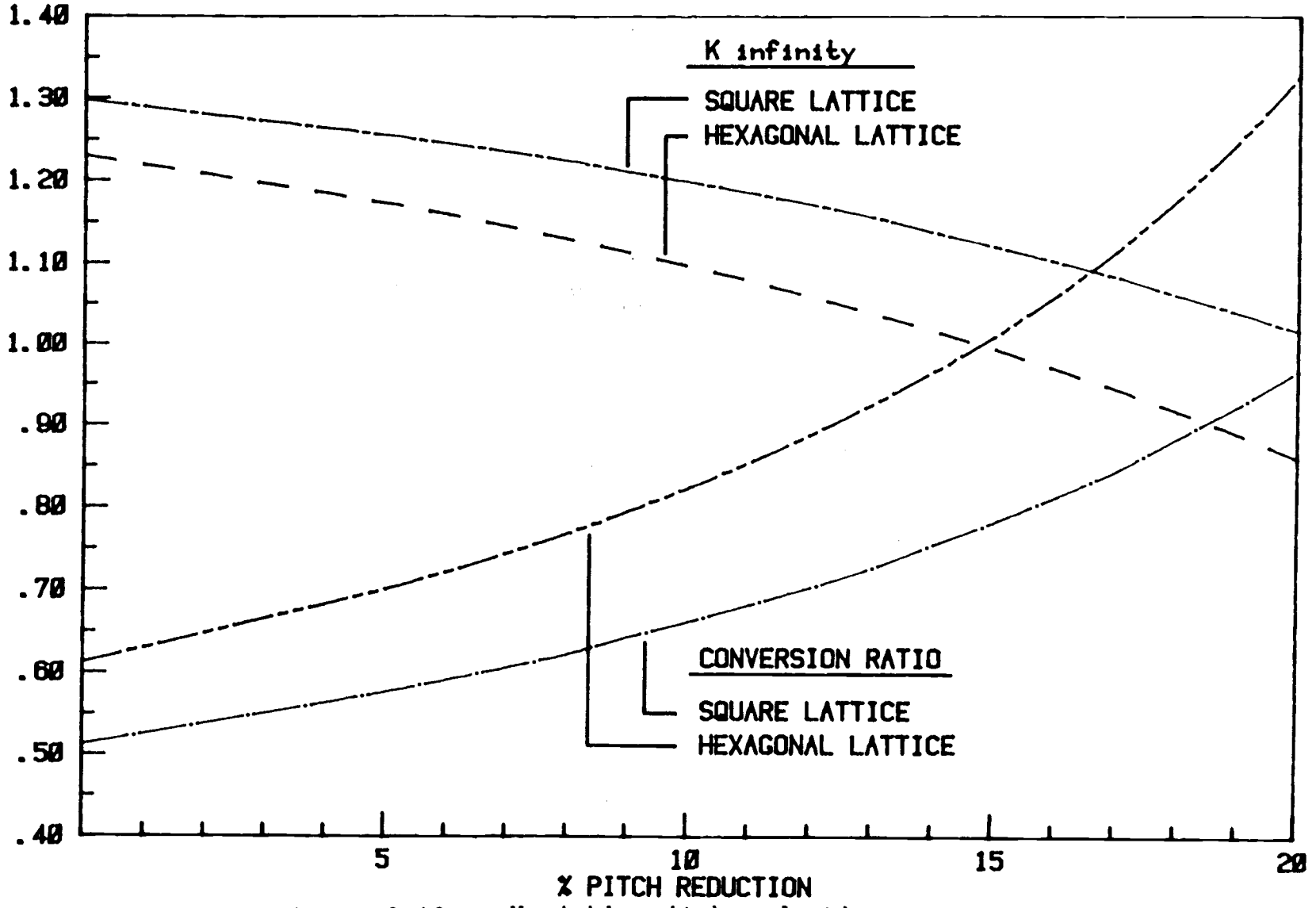


FIGURE 3-12 Variable pitch reduction curves

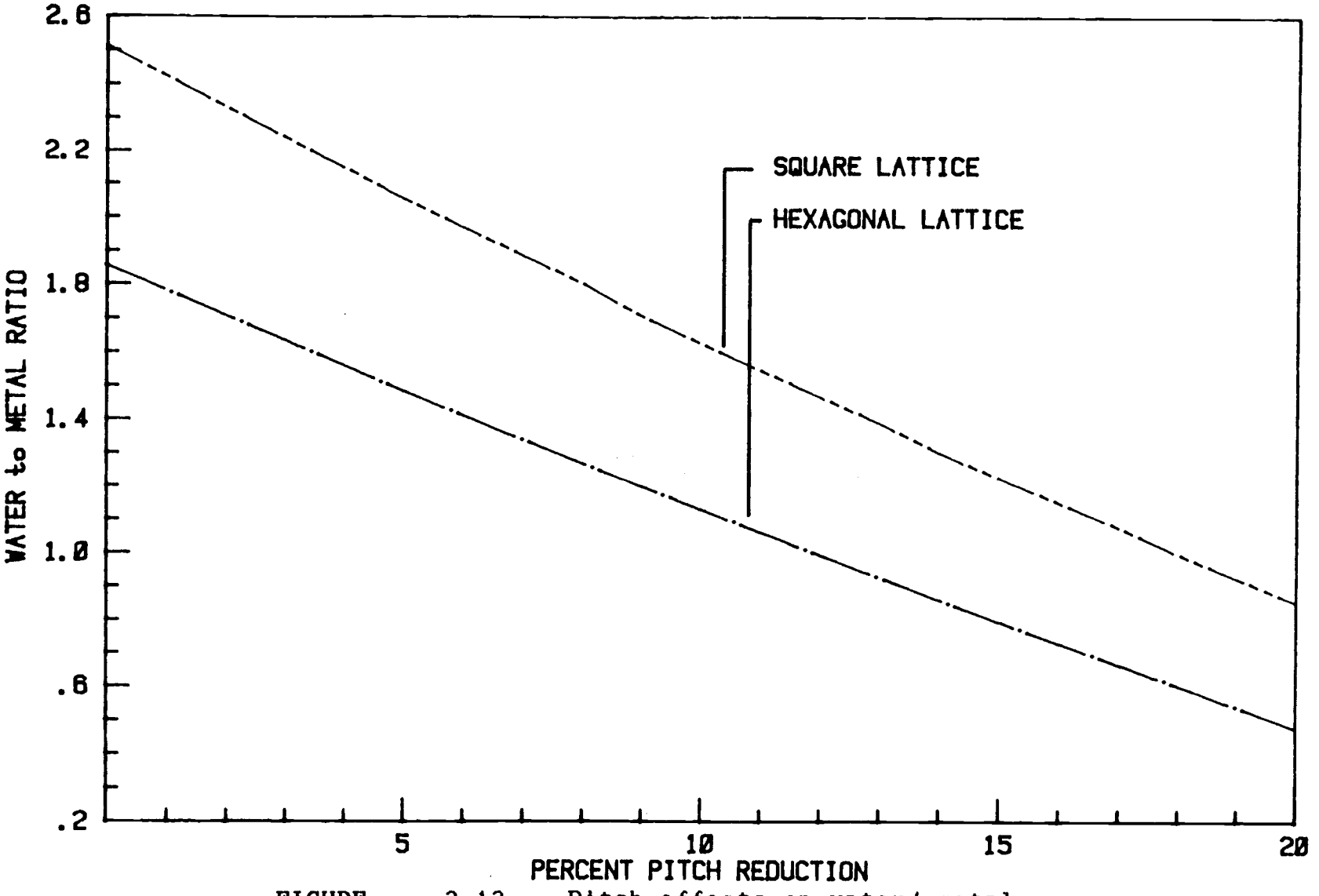


FIGURE 3-13 Pitch effects on water/ metal

information and creating a composite plot of k_∞ vs water/metal ratio shows that we are indeed operating in the undermoderated region for either design as brought out in figure 3-14.

This analysis would be woefully incomplete without continuing the investigation to consider the effects of burn-up. In keeping with one of the reported objective of this study, burn-up analysis was performed to the levels of 50,000 MWd/MTU. This level was selected since current work in metallurgy and irradiation damage indicates that improved methods of processing can yield materials capable of safely withstanding this level of exposure. This has been further supported through actual experimental investigations at FFTF and elsewhere and has been extensively documented in recent literature within the field itself.

The economics associated with an extension in the fuel burn-up was discussed in Chapter 2 to some length. One item that may have escaped the initial analysis is that a larger amount of fuel will be loaded into a given core when converting from a 17 x 17 to a 20 x 20 fuel assembly, almost 38% more U-235. This means that the fraction of the core

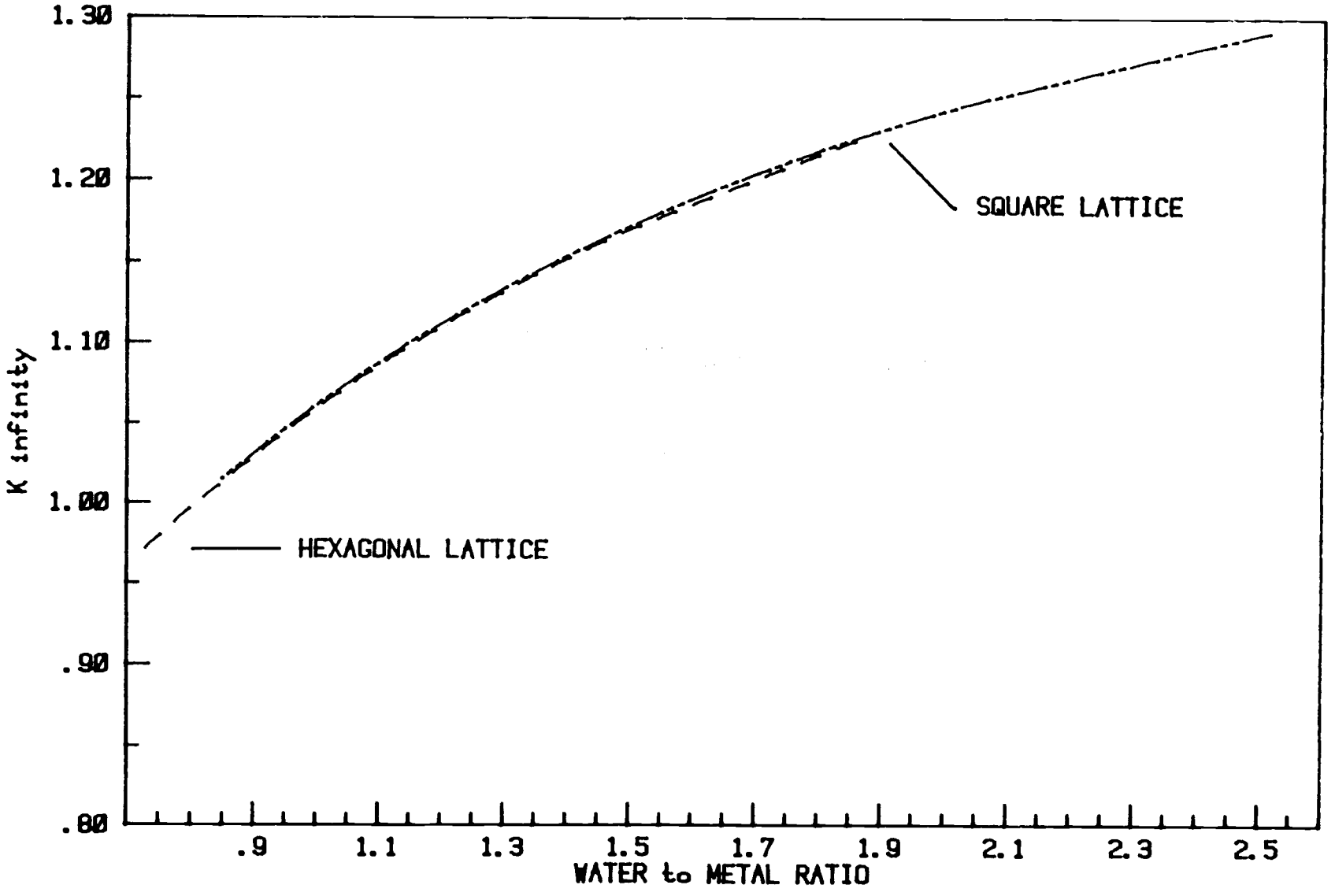


FIGURE 3-14 Core moderation characteristics

that would have to be replaced during fuel outages would be decreased. This translates to mean that the infinite multiplication factor for a given discharged fuel assembly could be lowered somewhat, and hence improve the fuel overall utilization.

The results of this burn-up analysis are presented in figures 3-15 thru 3-17. In figure 3-15 we see that the reduction of pitch associated with the adoption of a 20 x 20 lattice, be it hexagonal or square, tends to flatten out or reduce the depletion rate of positive reactivity. This is to say that the incremental consumption of U-235 is lessened a bit. This characteristic is also continued with regard to the conversion ratio performance as depected in figure 3-16.

A very carefull examination of these two plots leads to the selection of a probable design to receive more attention. In comparing the reference case with the 20 x 20 design, an optimum appears to exist somewhere between a 3% to 6% partial U-235 enrichment level. To assure adequate reserve reactivity, the square array would appear to be a superior choice. As an educated first guess it would seem that a 20 x 20 square lattice with a 4%

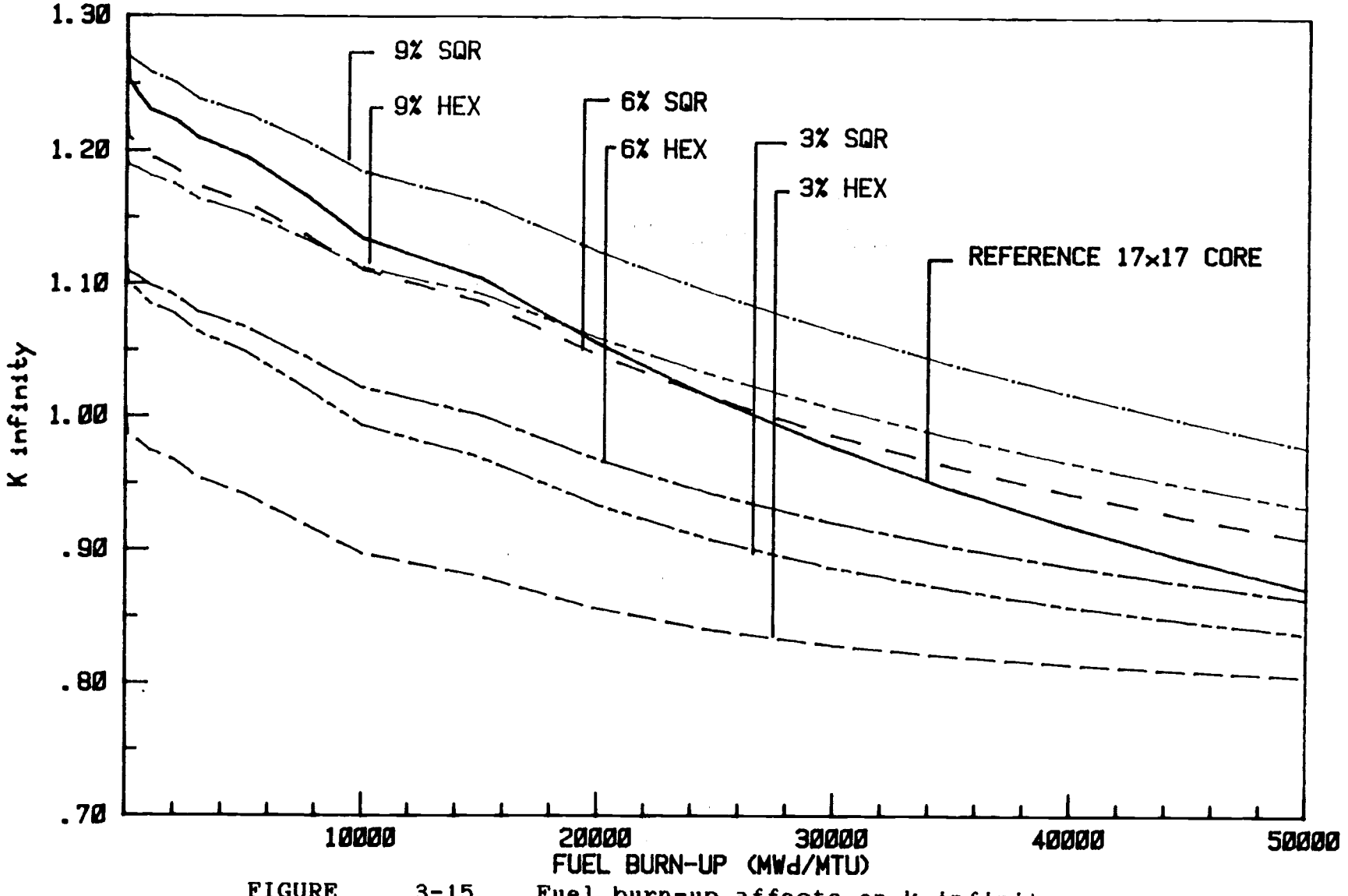


FIGURE 3-15 Fuel burn-up affects on k-infinity

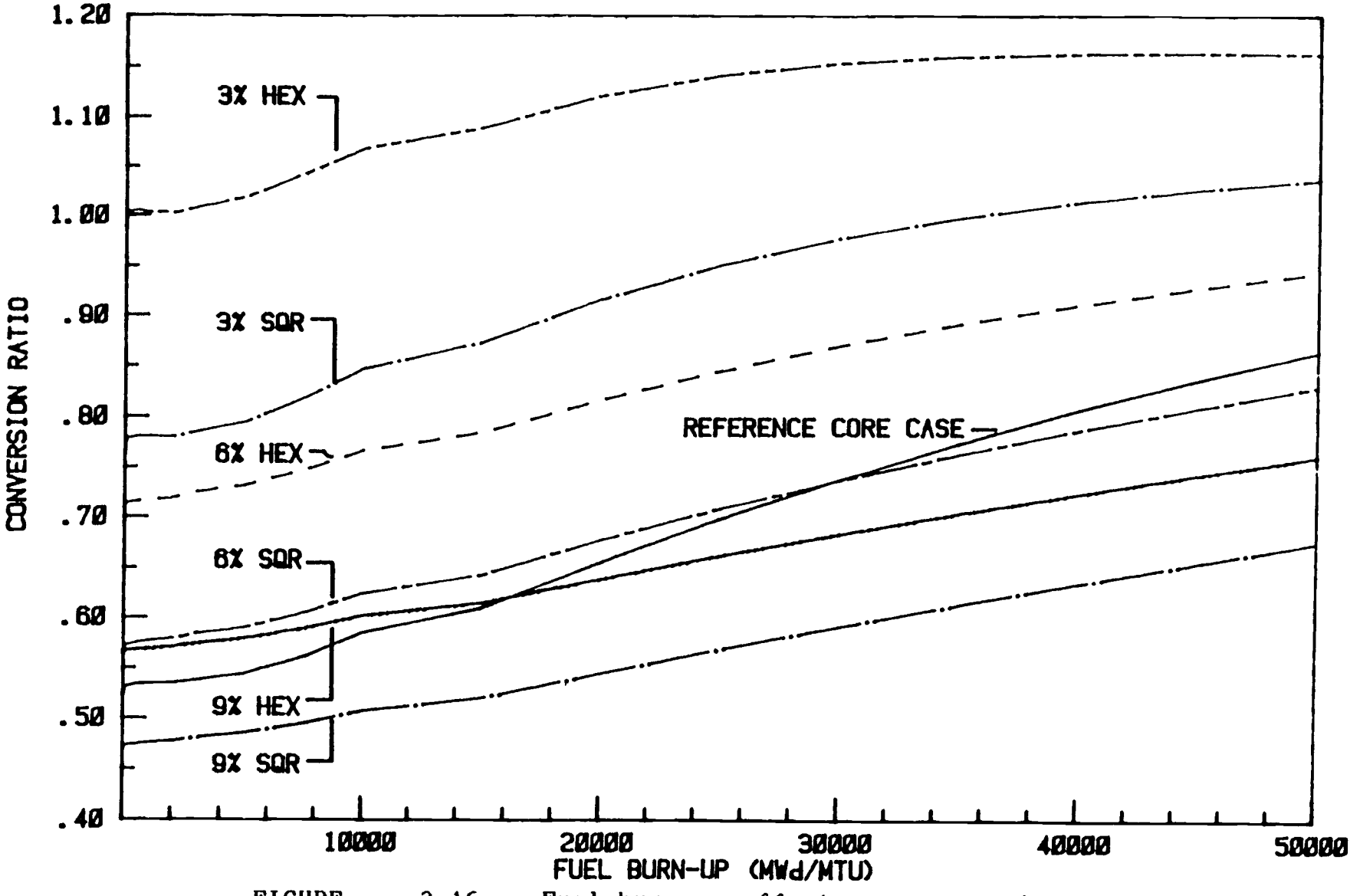


FIGURE 3-16 Fuel burn-up affects on conversion

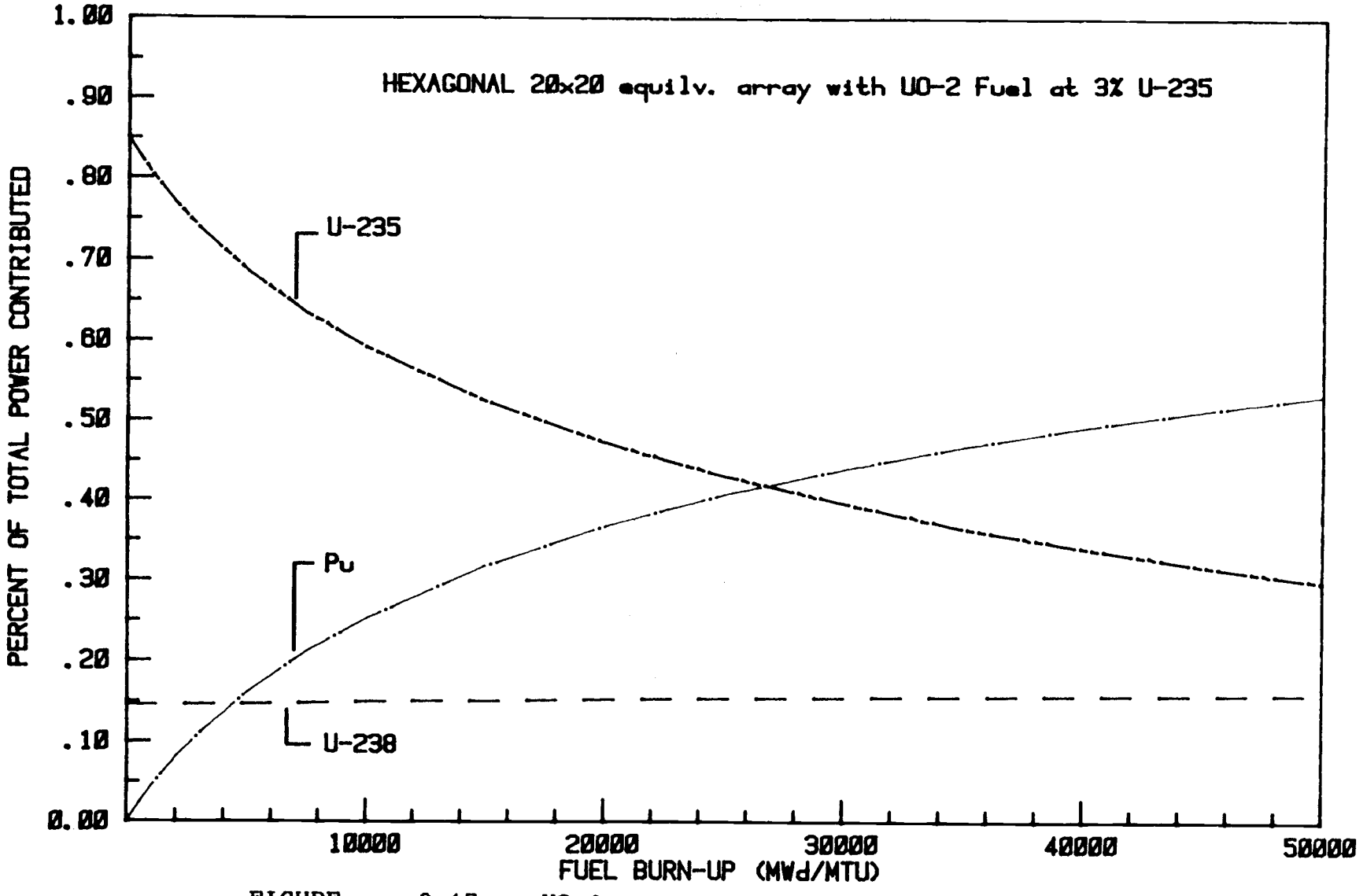


FIGURE 3-17 UO-2 power generation in fuel life

U-235 partial enrichment would be a close contender for a jump-off point needed to launch a more detailed investigation with more complex and involved neutronic codes. This choice appears to have a suitable reactivity depleation rate when compared with the reference core while offering a significant improvement in the overall conversion ratio.

This is further encouraging since enrichment costs associated with 4% as contrasted with 3% are not significantly different within the price structure now used by the industry. In addition to this, the enrichment facilities would not have to under go any significant re-tooling or expansion efforts in order to accommodate this change, nor apply for new federal licenses.

Finally it was thought to be instructive to show the partitioning of the elemential power generation over the span of the lifetime of a fuel assembly. As indicated in figure 3-17, the amount of thermal power generated from the fissioning of U-235 decreases as the fuel is consumed. It is rather interesting to note that at the current fuel discharge levels of around 30,000 MWd/MTU, we see

that a cross-over occurs where the power produced by the created plutonium exceeds that of the fuel itself. Although the plutonium is created via a capture event involving U-238, the overall inventory of U-238 is so much in excess that it's power contribution appears to remain almost constant thoughout the duration of the exposure. This particular plot does a very good job in depecting the improvement of fuel utilization that comes with extending the burn-up limits.

4. GOING THE EXTRA MILE: A FIRST LOOK AT USING THORIUM

In earlier sections the impending shortage of high quality uranium ore and therefore the amount of reclaimable U-235 was discussed in some detail. It was also brought out that recoverable thorium ore is quite plentiful. This started the author to begin considering the feasibility of incorporation of a thorium - uranium based fuel cycle within the confines of this research. This is not to say that exhaustive investigation on this issue was performed, but only as much as necessary to decide if further investigation would be well advised.

To forestall any unnecessary research, a preliminary neutron analysis was performed by substituting ThO-2 for UO-2 while maintaining a constant level of 3% U-235 enrichment. If the results of this proved promising, then further investigation would be under taken to assess the other aspects dealing with the question of feasibility.

4.1 A PRELIMINARY NEUTRON ANALYSIS

To conduct the analysis, it was decided to look at the extreme ends of the spectrum. This is to say that a complete replacement of UO-2 by ThO-2 is accomplished by keeping a constant inventory of fissile material, namely U-235, with respect to the fuel composition itself. To further streamline the analysis, a burnup analysis was performed in the place of doing separate steady state and burnup computer runs. In this manner the most information was obtained while conserving a scarce resource, namely mainframe computer time.

Whereas we are strictly replacing U-238 with Th-232 atoms, the water to metal ratio will remain unaffected and under moderation is maintained. In addition to this, the 20 x 20 lattice spacing was used to allow a direct comparison with the results of earlier investigations

4.2 THE PROCESS INVOLVED

It would seem appropriate to quickly review some important facts as related to thorium in contrast with uranium. First, to show the difference between the process under consideration figures 4-1 and 4-1a are offered. Here a couple of points are of some interest. The value of the neutron utilization factor for U-233 is 2.287 whereas for U-235 it is only 2.068; a reduction of almost 11%. This translates into more neutrons born per fission event using U-233 instead of U-235. The net result of this is an overall improvement in the conversion ratio potential as compared with the traditional fuel materials as indicated in Table 4-1 [41]. The general trend is depicted in the associated figure 4-2 [42] which clearly indicated the superior performance of U-233 over both U-235 and Pu-239 is the thermal range. Considering that with the adoption of a square lattice arrangement a softer neutron every spectrum is present, the advantage of introducing U-233 into the fuel cycle becomes significant.

U-238 Chain, Neglecting Short-Lived and Low Cross Section Stages

Captures in Pu-242 result
in a zero-cross section
daughter, arbitrarily.

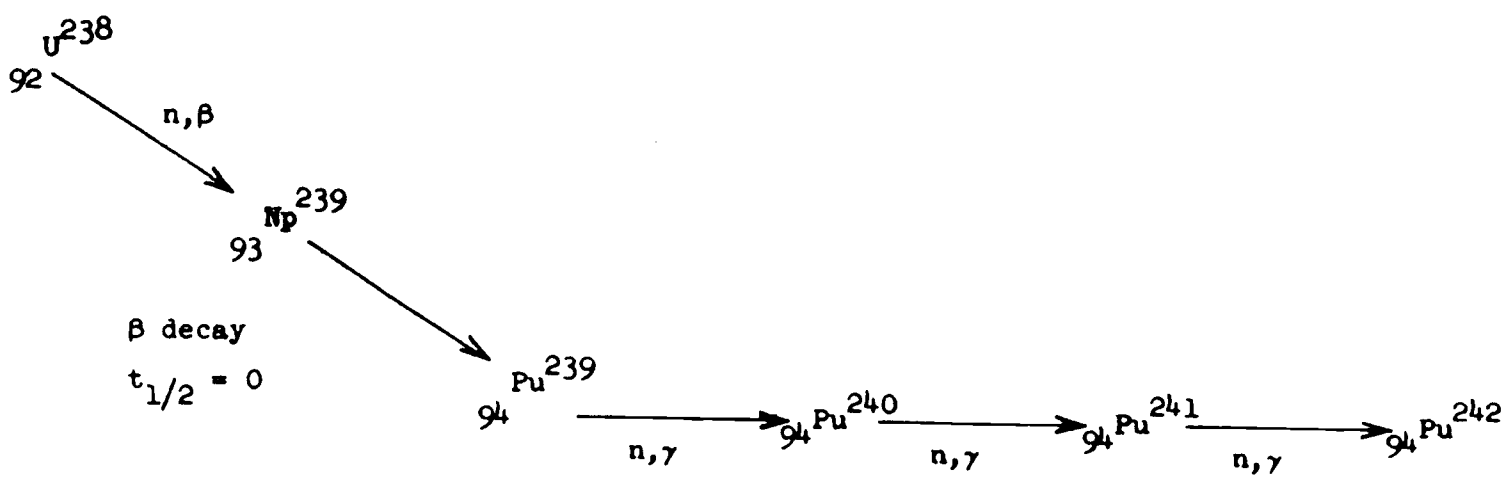


FIGURE 4-1 Pu-242 production chain

Thorium Chain, Neglecting Short-Lived and Low Cross Section Stages

Captures in U-236 result in a zero
cross section daughter, arbitrarily

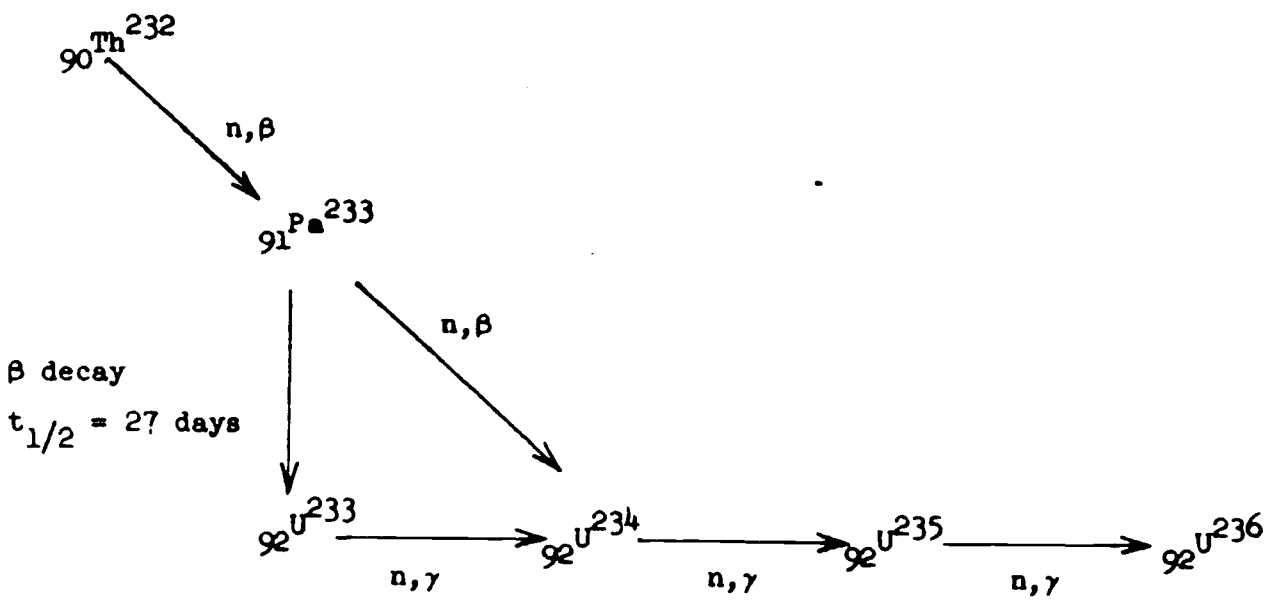


FIGURE 4-1a U-233 production chain

TABLE 4-1 : BREEDING OR CONVERSION POTENTIAL

Nuclide	Neutron Energy				
	Thermal	1 to 3000 eV	3 to 10 keV	0.1 to 0.4 MeV	0.4 to 1 MeV
Plutonium-239	1.09	0.75	0.9	1.6	1.9
Uranium-235	1.07	0.75	0.8	1.2	1.3
Uranium-233	1.20	1.25	1.3	1.4	1.5

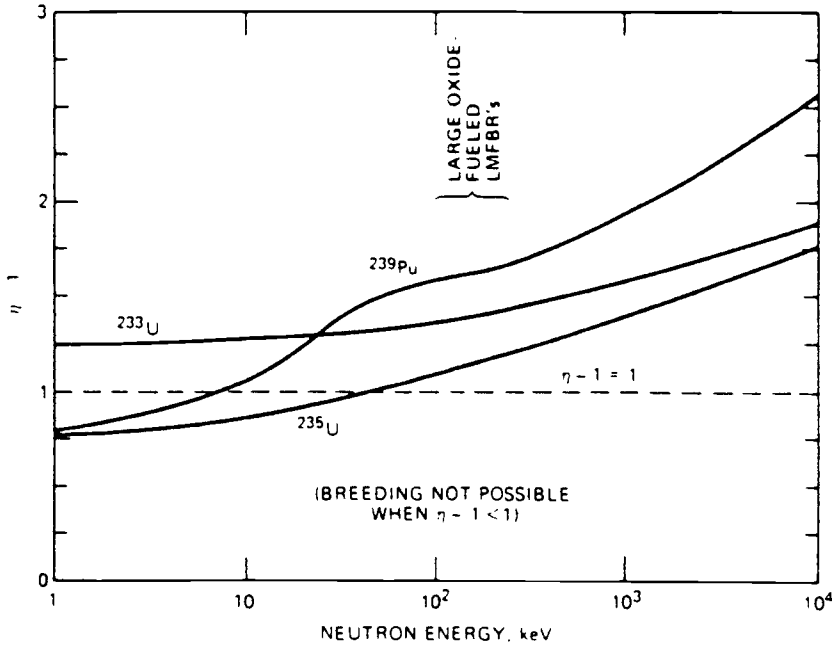


FIGURE 4-2 Breeding behavior curves

4.3 RESULTS OF PRELIMINARY ANALYSIS

As anticipated due to the higher absorption cross section of thorium-232 as opposed to U-238, the infinite multiplication value is reduced from initial values with UO-2 of 1.13 to 0.98 with ThO-2 for the square lattice. As indicated in figure 4-3 this is compared with the 1.28 value obtained for the design reference case. It is also noteworthy to point out the larger degree of evening out the incremental decrease of this factor over core life. This feature is very desireable since it would decrease the requirement for burnable poison loading and consequently enable a flatter radial core flux profile to be obtained. It is also evident that in order to obtain an operational value of reactivity for the hexagonal arrangement, the specific level of enrichment would be significantly higher than compared to the required amounts for the square arrangement. This is directly related to costs and results in a significant drawback to the adoption of a hexagonal array.

Conversely, a dramatic improvement occurs in the conversion ratio performance as depicted in

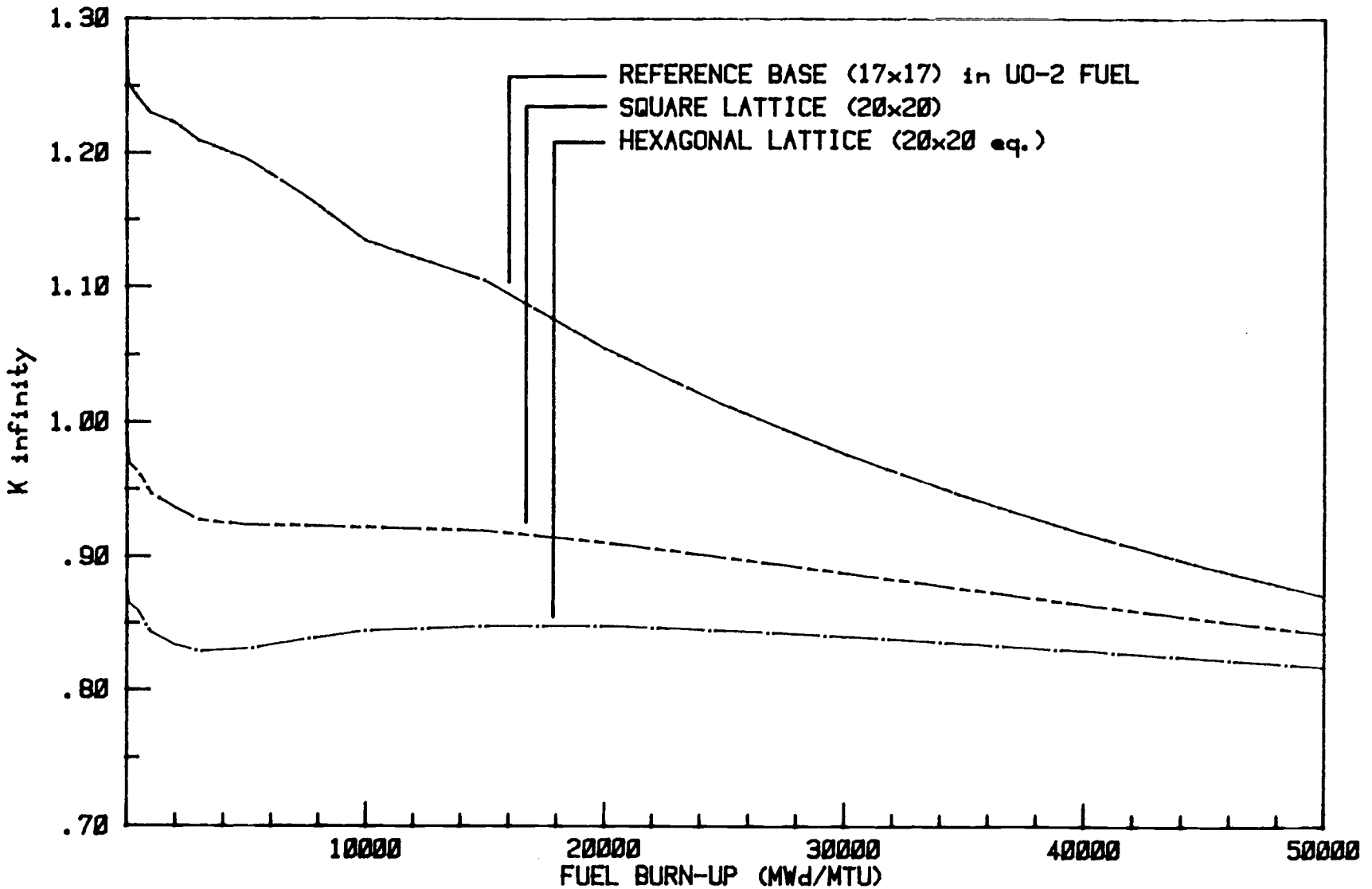


FIGURE 4-3 Burnup characteristics for Th-U fuel

figure 4-4. Initial values for the square array see better than 17.5% improvement (0.7650 vs. 0.8995). This combined with the improved performance will even permit the square array to become a breeder with prolonged exposure.

Clearly, using perhaps a partial enrichment of 4% U-235 might just be the magic recipe. In any respect, the addition of thorium shows distinct advantagous possiblities. Therefore, this area warrants further investigation in the opinion of the author.

As a final item, we examine the division of power production for this new fuel composition, though it's exposure. As indicated in figure 4-5, behavior similar to the UO-2 case is evident except U-233 has supplanted plutonium, and Th-232 the U-238 components. Since the thorium contributes less to the power production, the overall utilization of uranium resources is significantly enhanced.

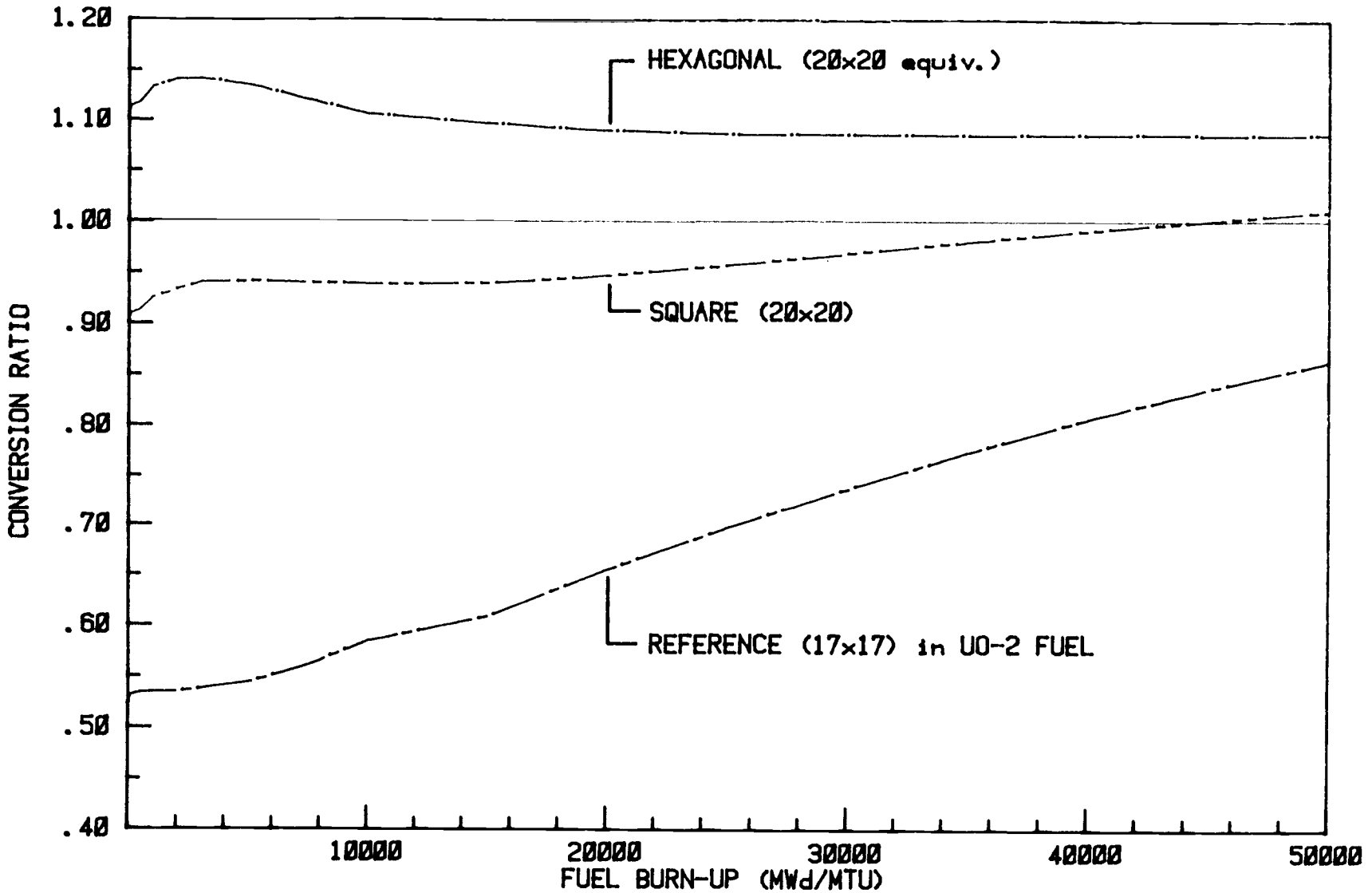


FIGURE 4-4 Conversion Ratio effect chart

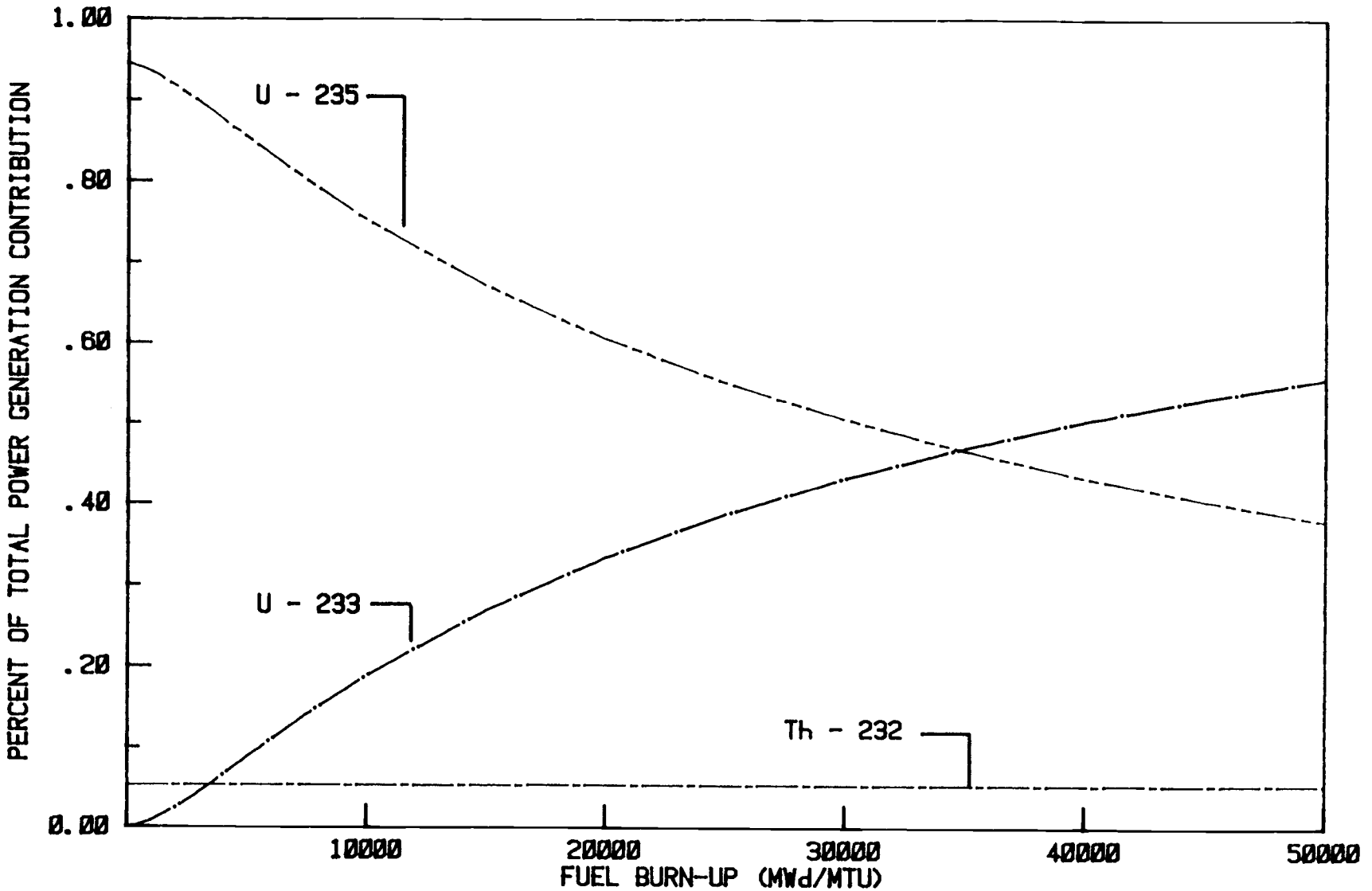


FIGURE 4-5 ThO-2 power production partitioning

4.4 SOME OTHER FACTORS OF CONCERN

Some unique complications arise in the Th - U-233 cycle. For example, during an intermediate stage in the U-233 production, protactinium-233 is created which has a relatively long half-live (27 days) and a fairly large cross-section for absorption in both the resonance and thermal energy regions. This latter effect is a strong function of the neutron energy spectrum and as such the power density of the core and may impose a significant effect upon the reactivity with only minor effects felt by the fuel conversion process.

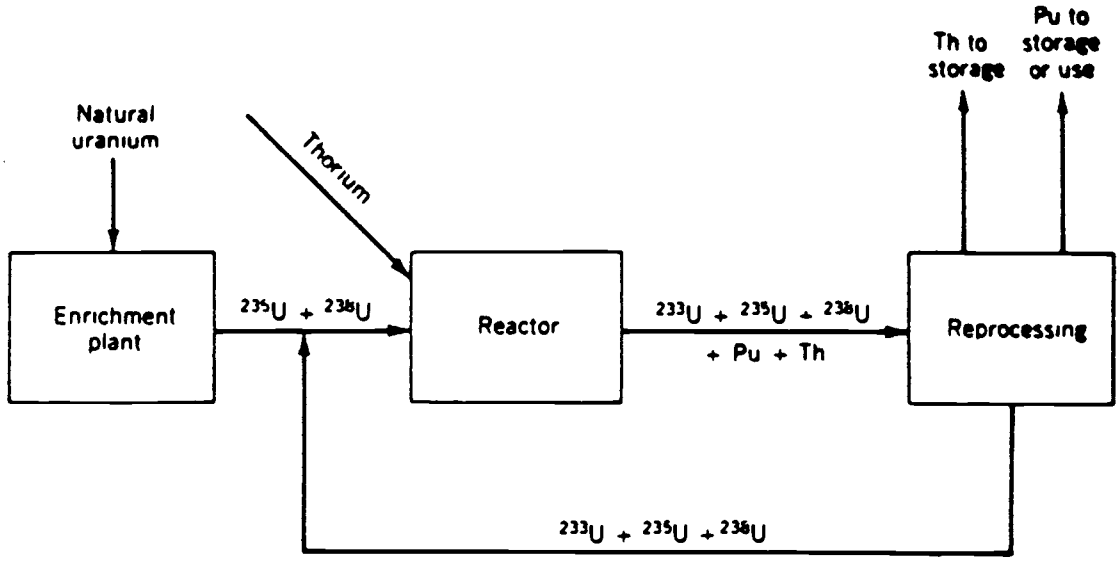
Yet another concern is related to the production of U-232 formed by (N,2N) reactions with U-233 and Th-231. Whereas some of the decay daughters have strong gamma ray emissions, shielding and remote handling is effectively mandated. With the current state of robotics therefore, this should not pose any unrealistic problems. In fact when one meets the problems associated with proliferation, this problem could turn around and prove to be a significant asset.

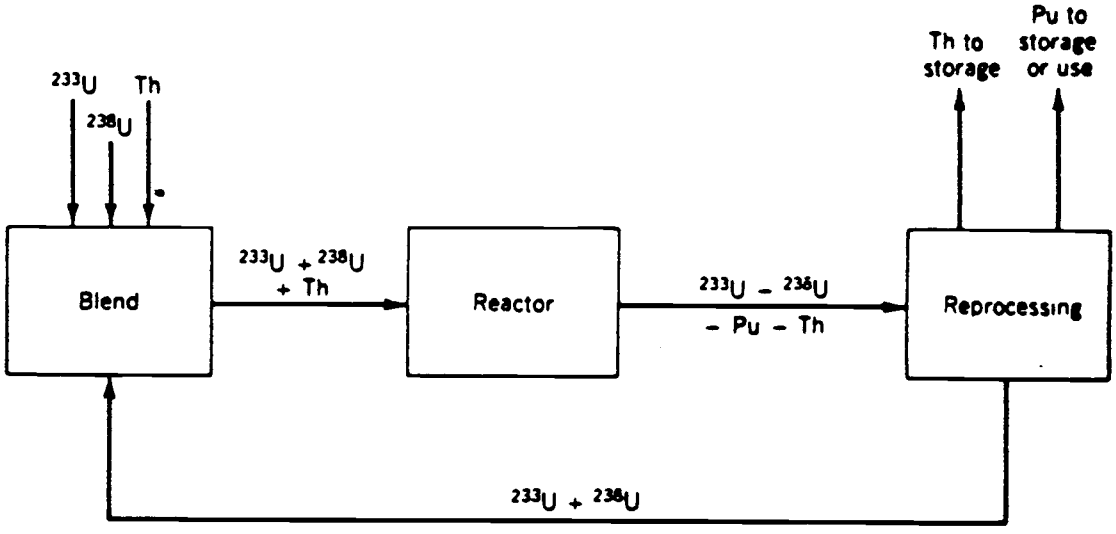
This concept is receiving increased attention

and as a result many types of fuel cycle proposals are being advanced. For illustrational purposes only, these are offered in figures 4-6 [43] and 4-7 [44] without much further comment. To that end, the author feels that the mixed oxide approach offers the most hope with the uranium and thorium configuration being of greatest interest within the context of a homogeneous matrix.

The reason behind this endorsement is that it has been shown that the fabrication and performance characteristics between the two oxides are similar and completely compatible [45]. The mixed oxide has already been shown to perform similarly to that of UO-2 under irradiation with less swelling, which is a major benefit, and fabrication plants would need only minor re-tooling to produce the product. The advantages associated with the adoption of this fuel form seem almost overpowering.

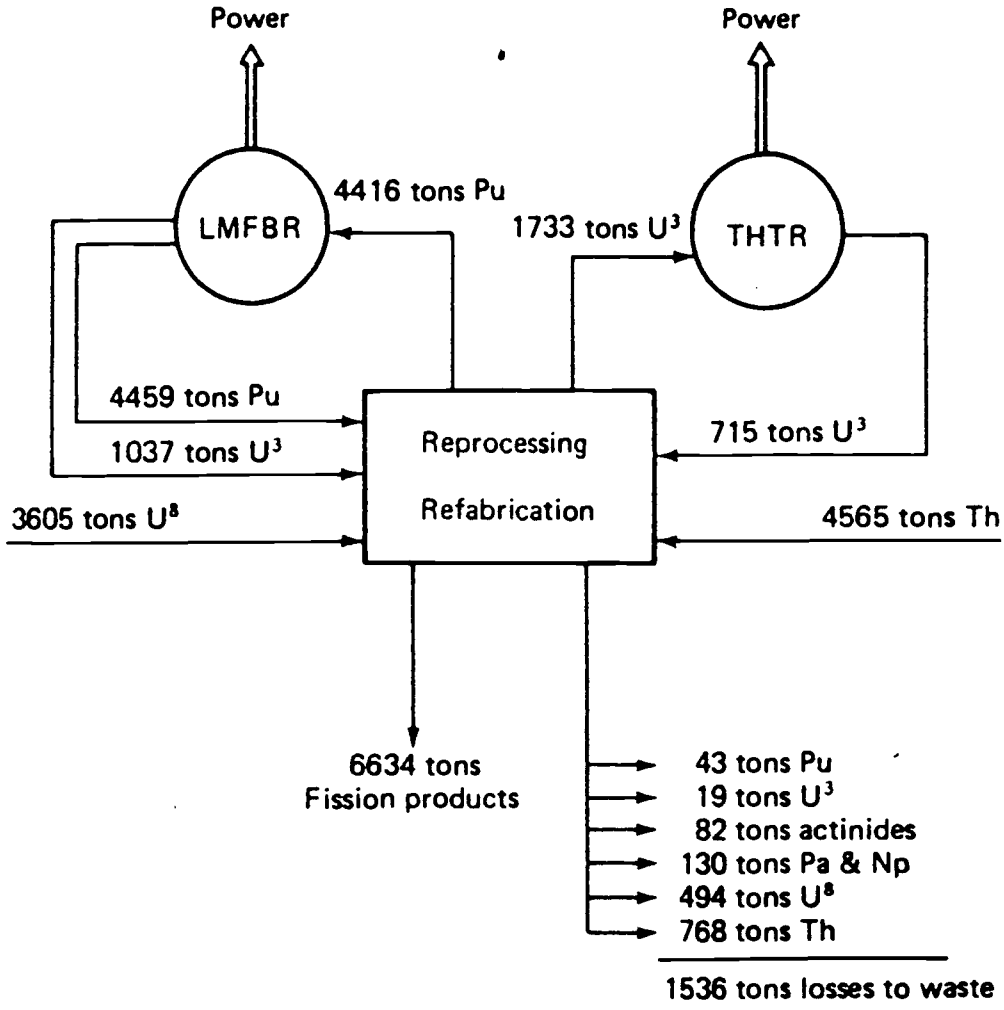
One last item is that through the adoptive use of thorium, current uranium resources can be prolonged. This is graphically displayed in figure 4-8 [46] which shows that for more than 2/3 of a reactor's lifetime, significant improvement of uranium ore utilization can be realized thru the





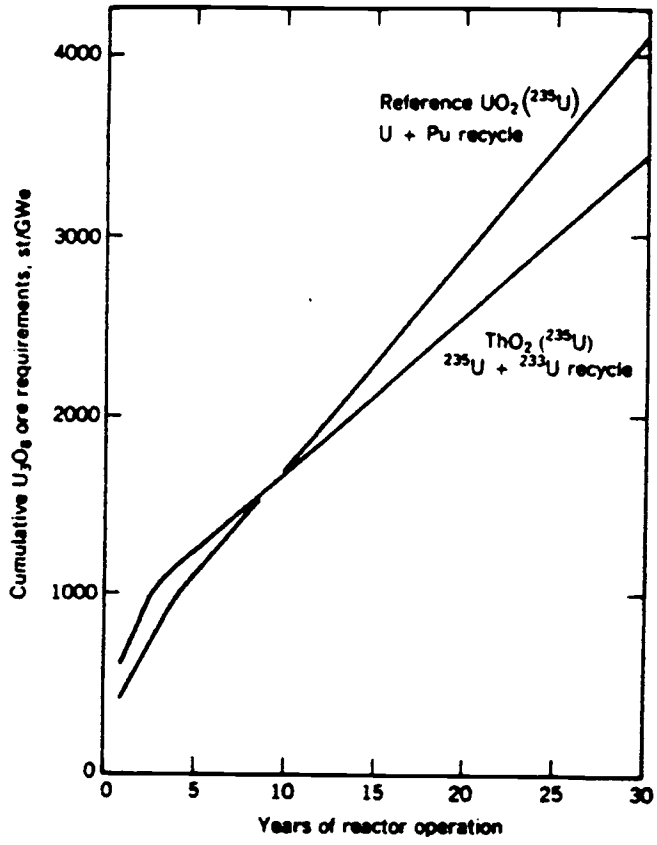
In the denatured cycles the fissile isotopes ^{233}U (bottom) and ^{235}U (top) are always mixed with nonfissile ^{238}U by deliberate addition of natural or depleted uranium to the thorium rods. This precludes the use of chemical separation of fissile material. However, plutonium is still produced at the reprocessing plants and left mixed (spiked) with radioactive fission products. ^{233}U is a weapons material but can only be separated from ^{238}U by isotope separation techniques.

FIGURE 4-6 Mixed oxide fuel cycle



Annual throughputs and losses (in tons) for a 17 TWyr/yr, FBR-HTR operation. Only closed balances for Pu, U^3, and total (U^8 and thorium) are shown. The 6 ton mass defect associated with 17 TWyr/yr is not accounted for.

FIGURE 4-7 Mixed oxide fuel cycle



Cumulative U_3O_8 ore requirements versus reactor lifetime.

FIGURE 4-8 Ore requirements vs. core lifetime

implamentation of the thorium - uranium mixed oxide fuel.

4.5 SUMMARY

Given the above information and data, it would seem clear that not only is thorium cycle feasible, but in most prospects exceedingly desireable. From the standpoint of the pragmatic engineer, the current system is adequately equipped to investigate this fuel cycle quite easily.

Considering the potential for extending the energy generating capabilities of nuclear power, the author is surprised to see so little effort being devoted to this area. The only catch is that thorium needs to be irradiated in order to perform as desired, and if we run scarce or ever completely deplete our U-235 supplies, this whole concept will never get off the ground!

5. THE FINAL SUMMARY

In review of the results obtained through this research effort it seems safe to conclude that a reasonable converter design or perhaps even a low grade breeder design can be developed. Furthermore the feasability of retro-fitable design that could be utilized by the currently operating PWR power plants is also attainable.

Specifically, as a result of this effort, the optimized 20 x 20 fuel assembly design can satisfy the above criteria rather well thus affording a more fuel efficient operation of the Trojan power plant. Moreover, with the increased fuel inventory that would be present in this choice, longer periods of exposure are possible which in turn would reduce the size and frequency of fuel replacement evolutions during the planned outages.

All this equates out to the potential of tremendous economic savings while promoting greater efficiency in fuel utilization as well as significantly extending the already dwindling uranium supplies not to mention ensuring the very future of the Nuclear Industry well into the 21st

century.

In regards to the specific design as treated within this work, the author maintains that its construction lies well within the capabilities of our current technology in both areas of design and fabrication. Additionally, it also satisfies all of the applicable safety and performance criteria currently desired. Since the new design and the reference design have matching outside assembly dimensions, the task of retro-fitting is considerably simplified. Finally, whereas the new design offers outstanding performance in both the areas of thermohydrolics and neutronics, the likelyhood that a round or two of optimized design analysis conducted with more exacting codes would produce a very workable design appears to be quite good.

The utilization of thorium within the current fuel cycle process is, in the opinion of the author, of paramount importance for the reasons afore mentioned. Furthermore, the introductory investigation results as outlined in Chapter 4.0 are exceptionally promising to that end.

It is frankly bewildering to the author why it is that something which posesses the vast potentials to produce energy in the amounts that rival Fusion energy itself, is left to the way-side as thorium has been up to now! Considering the prevailing and ever cautious attitude which desires that the United States becomes more energy self-sustaining, the apparent disregard of this option seems to be ill-advised and wrong-headed.

In closing it is very strongly felt that with all things considered, further research along this avenue should be vigorously pursued and is indeed well warranted.

REFERENCES

1. Robert A. Gross, FUSION ENERGY, John Wiley and Sons, New York, pp 2-3 1984

2. Editor, WORLD LIST OF NUCLEAR POWER PLANTS, Nuclear News, Vol.29 No.2, pp 67-86 Feb. 1986

3. The National Academy of Sciences, ENERGY IN TRANSITION 1985-2010, W.H. Freeman and Company, New York, 1980

4. Wolf Hafele, et al., ENERGY IN A FINITE WORLD-PATHS TO A SUSTAINABLE FUTURE, Ballinger Publishing Co., Cambridge Mass. pp 186 1981

5. Ibid, pp 138

6. Ibid, pp 136

7. Ibid, pp 138

8. A. M. Plat, et al., NUCLEAR FACT BOOK 2nd Ed., Hardwood Academic Publishers, New York, pp 1 1985

9. U.S.D.O.E., ANNUAL ENERGY OUTLOOK, 1983, WITH PROJECTIONS TO 1995, May, 1984

10. Kenneth Davis, A PERSPECTIVE ON NUCLEAR ENERGY IN THE USA, Atomic Industrial Forum, Bethesda, Md., pp 3 (oration) 15 April 1986

11. Carl Walske, FUTURE OF THE NUCLEAR POWER OPTION IN THE UNITED STATES, Atomic Industrial Forum, Bethesda, Md., pp 4 (oration) April 1986

12. Reference 2, pp 162

13. Reference 2, pp 143

14. Frank J. Rahn, et al., A GUIDE TO NUCLEAR POWER TECHNOLOGY, John Wiley and Sons, New York, pp 50 1984

15. Ibid, pp 51

16. John R. Lamarsh, INTRODUCTION TO NUCLEAR ENGINEERING - 2nd Ed., Addison Westly, Reading, Mass. pp 151 1983

17. Reference 14, pp 145

18. Reference 2, pp 49

19. Reference 14, pp 149-50

20. Reference 2, pp 49-51

21. Reference 14, pp 874

22. Reference 14, pp 861

23. Reference 14, pp 895

24. Reference 14, pp 894

25. Reference 2, pp 54

26. Reference 2, pp 54

27. Reference 2, pp 55

28. Werner Oldekop, et al., GENERAL FEATURES OF ADVANCED PRESSURIZED WATER REACTORS WITH IMPROVED FUEL UTILIZATION, Nuclear Technology Vol. 59, pp 212-227 Nov. 1982

29. Ibid, pp 212

30. W.C. Schick, et al., MEASURED PROOF OF BREEDING OF THE LIGHT WATER BREEDER REACTOR, American Nuclear Society Transactions - Winter Meeting, Washington D.C., pp 476-477 1986

31. Cornelis H.M. Broeders, CONCEPTUAL DESIGN OF A (Pu,U)O2 CORE WITH A TIGHT FUEL ROD LATTICE FOR AN ADVANCED PRESSURIZED WATER REACTOR, Nuclear Technology Vol. 71, pp 82-95 Oct. 1985

32. Ibid, pp 85

33. C.L. Wheeler, et al., COBRA-IV-I: AN INTERM VERSION OF COBRA FOR THERMAL - HYDRAULIC ANALYSIS OF ROD BUNDLE NUCLEAR FUEL ELEMENTS AND CORES, BNWL-1962/UC-32

34. L.S. Tong, BOILING CRISIS AND CRITTICAL HEAT FLUX, USAEC; TID-25887, pp 10, 1972

35. Portland General Electric, FINAL SAFETY ANALYSIS REPORT FOR THE TROJAN NUCLEAR POWER PLANT, Operating Licence NPF-1, Vol. 4, Dockett Number 50-344

36. James R. Welty, et al., FUNDAMENTALS OF MOMENTUM, HEAT, AND MASS TRANSFER 3rd Ed., John Wiley and Sons, New York, 1984

37. Donald L. Hagrman, et al., MATPRO VERSION II: A HANDBOOK OF MATERIAL PROPERTIES FOR USE IN THE ANALYSIS OF LIGHT WATER REACTOR FUEL ROD BEHAVIOR, Argonne National Labs, NUREG/CR0497 Feb. 1979

38. Harry W. Graves Jr., NUCLEAR FUEL MANAGEMENT, John Wiley and Sons, New York, pp 52 1979

39. Westinghouse, LEOPARD - A SPECTRUM DEPENDANT NON-SPATIAL DEPLETION CODE FOR THE IBM-7094, WCAP-3269-26

40. L.E. Strawbridge, R.F. Berry, CRITICALITY CALCULATIONS FOR UNIFORM WATER MODERATED LATTICES, Nuclear Science and Engineering, Vol. 23, pp 58-73 1965

41. Samuel Glasstone, et al., NUCLEAR REACTOR ENGINEERING 3rd Ed., Van Nostrand Reinhold Company, New York, pp 540 1981

42. Ibid, pp 540

43. Reference 14, pp 919

44. Reference 2, pp 57

45. Reference 14, Chapters 7, 11, and 20

46. Reference 14, pp 482

APPENDICES

APPENDIX A

APPENDIX A

MATPRO UO-2 THERMAL CONDUCTIVITY APPROXIMATION

In order to use the variable fuel conductivity option offered in the COBRA-IV code, a polynomial expression must be used. Within the industry, the correlation of choice is the one which is contained within the MATPRO documentation. To provide easy reference it has been reproduced here as eqn. 1 [37]:

$$\underline{0 < T < 1650\ C}$$

$$k = P * \left[\frac{K1}{K2 + T} + K3 * EXP\ (K4 * T) \right]$$

$$\underline{1650 < T < 2840\ C}$$

$$k = P * \left[K5 + K3 * EXP\ (K4 * T) \right]$$

where the variables and constants are:

k = UO-2 fuel thermal conductivity (Watts/cm-K)
P = Porosity correction factor
T = Temperature of the fuel (degree C)

K1 = 40.4 K2 = 464.0 K3 = 1.216E-4
K4 = 1.867E-3 K5 = 1.91E-2

As can be easily seen, the exponential behavior is a far cry from a polynomial type of treatment. Following a rather extensive search which yielded little direct success, the author

opted to create a polynomial approximation to the MATPRO correlation.

This was accomplished by first obtaining a fine mesh of data point values from the MATPRO correlation with the aid of a computer. These coordinates were then subjected to a regressional treatment in order to generate the coefficients for a polynomial equation of the desired degree. This was performed and resulted in a second and third order polynomial fit to the MATPRO correlation. For lack of a better name, these relationships were dubbed "The Kliewer Polynomials" and are of the form:

$$K_t = B1 + (B2 * T) + (B3 * T^2) + (B4 * T^3)$$

where:

K_t = UO-2 fuel conductivity (BTU/Ft-Hr-F)

	2nd ORDER	3rd ORDER
B1 =	3.939638834	= 3.841533289
B2 =	-1.8039685 E-3	= -1.65570152 E-3
B3 =	3.00324044 E-7	= 2.438677071 E-7
B4 =	n/a	= 6.076454364 E-12
Valid Temp Range =	400 - 5825 F	= 400 - 5900 F

These correlations yield values that are at worst in the neighborhood of 9 to 10 % off the MATPRO values and within the dynamic range of normal operations, either one closes to within 1 to 2% of the MATPRO values. This relationship can be seen in figure A-1.

As can be seen in the plot, excellent agreement has been achieved with these psudo-correlations in comparison with the MATPRO results. It was therefore concluded that these relationships would be sufficient for use in the COBRA-IV code for variable fuel conductivity evaluations.

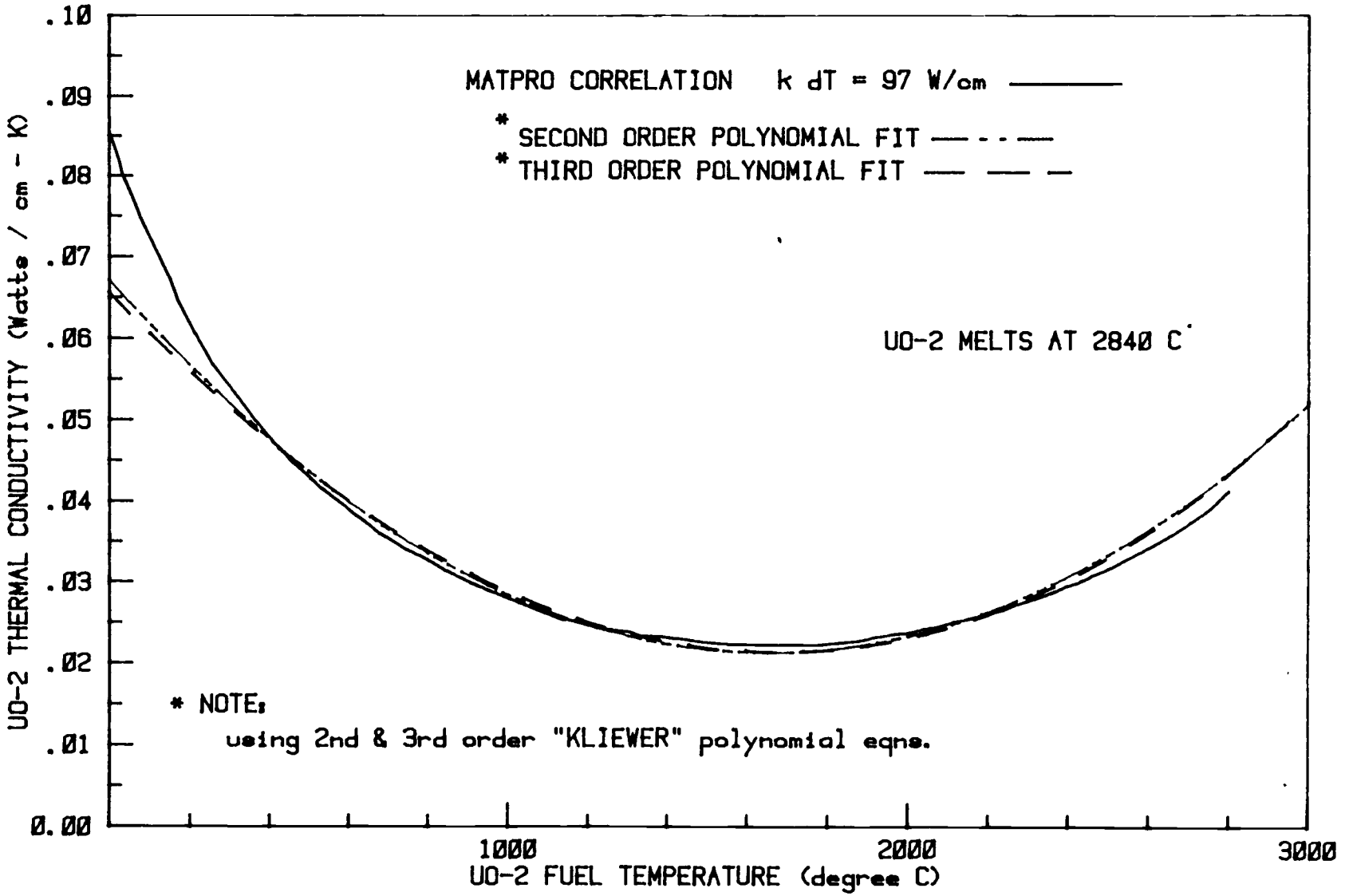
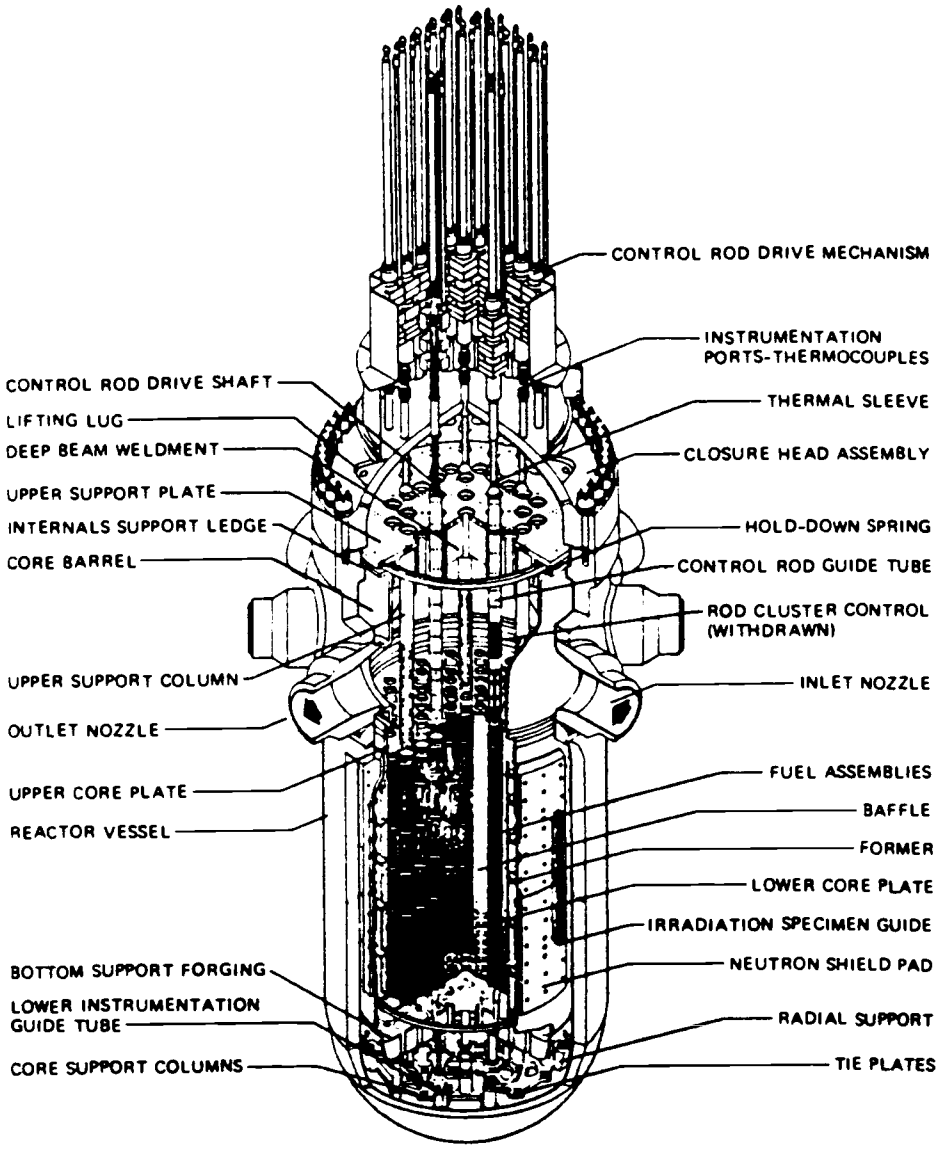


FIGURE A-1 MATPRO Correlation Comparison

APPENDIX B



Internal structure of a PWR (Westinghouse Electric Corp.).

FIGURE B-1 PWR Reactor Cut-away

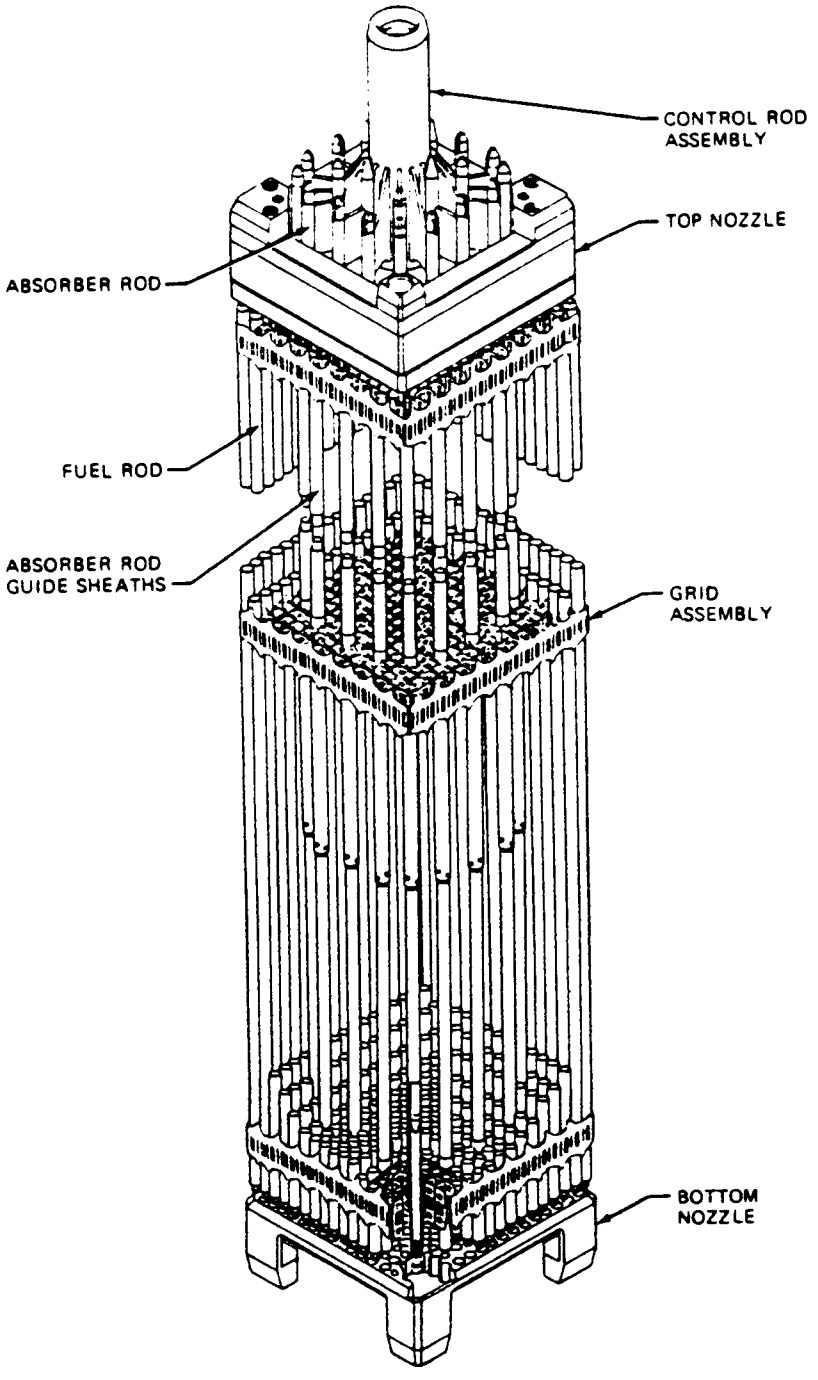


FIGURE B-2 PWR Fuel Rod Sub-assembly

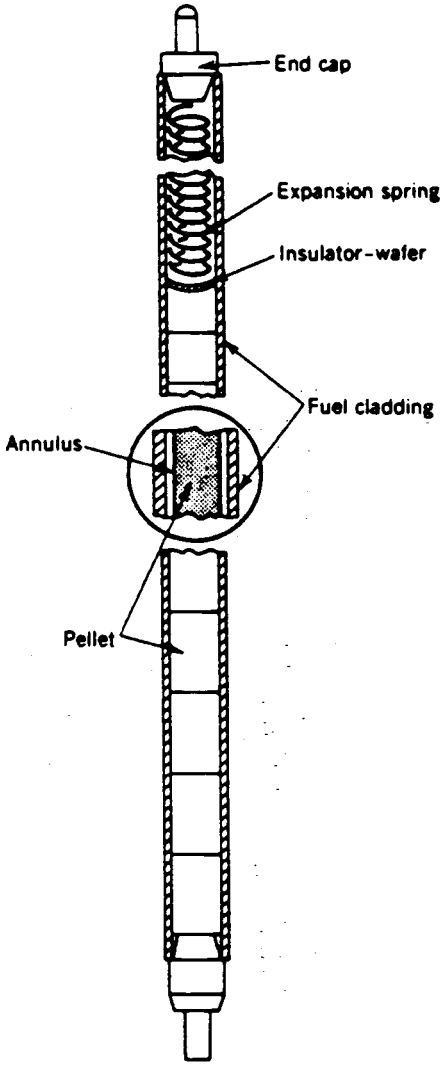


FIGURE B-3 PWR Fuel Pin Cut-away

TABLE B-1 Reactor Design Comparison Data

Thermal and Hydraulic Design Parameters	Trojan 17 x 17 Fuel Assembly with Densification Effects	Trojan 15 x 15 Fuel Assembly without Densification Effects
Reactor core heat output, MWt	3411	3411
Reactor core heat output, Btu/hr	$11,641.7 \times 10^6$	$11,641.7 \times 10^6$
Heat generated in fuel, %	97.4	97.4
System pressure, nominal, psia	2250	2250
System pressure, minimum steady-state, psia	2220	2220
Minimum DNBR for design transients	>1.30	>1.30
Typical flow channel	2.04	-
Thimble (cold wall) flow channel	1.71	-
Coolant Flow:		
Total thermal flow rate, lb/hr	132.7×10^6	132.7×10^6
Effective flow rate for heat transfer, lb/hr	126.7×10^6	126.7×10^6
Effective flow area for heat transfer, ft^2	51.1	51.2
Average velocity along fuel rods, ft/sec	15.7	15.5
Average mass velocity, $lb/hr\text{-}ft^2$	2.48×10^6	2.47×10^6
Coolant Temperature:		
Nominal inlet, °F	552.5	552.5
Average rise in vessel, °F	64.2	64.2
Average rise in core, °F	66.9	66.9
Average in core, °F	585.9	585.9
Average in vessel, °F	584.7	584.7

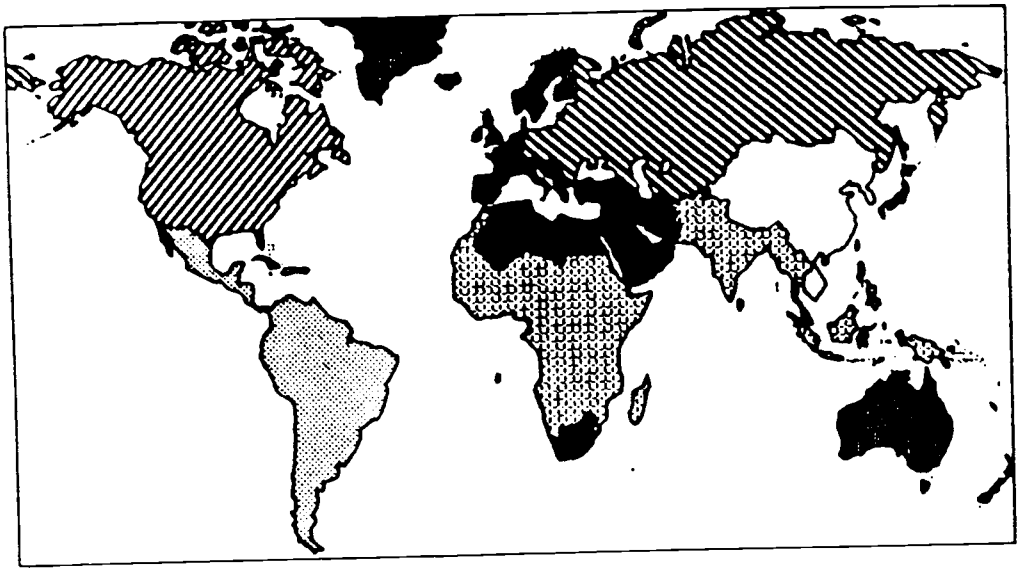
Thermal and Hydraulic Design Parameters	Trojan 17 x 17 Fuel Assembly with Densification Effects	Trojan 15 x 15 Fuel Assembly without Densification Effects
Heat Transfer:		
Active heat transfer, surface area, ft^2	59,700	52,200
Average heat flux, Btu/hr-ft^2	189,800	217,200
Maximum heat flux, for normal operation Btu/hr-ft^2	440,340	521,300
Average thermal output, kW/ft	5.44	7.03
Maximum thermal output, for normal operation, kW/ft	12.6[a]	16.9
Peak linear power for determination of protection setpoints, kW/ft	18.0[c]	21.1
Heat flux hot channel factor, F_Q	2.32[b]	2.40
Fuel central temperature:		
Peak at 100% power, °F	3400	3950
Peak at peak linear power for determination of protection setpoints	4150	4500

Core Mechanical Design Parameters	Trojan 17 x 17 Fuel Assembly with Densification Effects	Trojan 15 x 15 Fuel Assembly without Densification Effects
Rod Cluster Control Assemblies:		
Neutron absorber	Ag-In-Cd	Ag-In-Cd
Cladding material	Type-304 SS-cold worked	Type-304 SS-cold worked
Clad thickness, in.	0.0185	0.019
Number of clusters	53	53
Number of absorber rods per cluster	24	20
Core Structure:		
Core barrel, I.D./O.D., in.	148.0/152.5	148.0/152.5
Thermal shield	Neutron pad Design	Neutron pad Design
Structure Characteristics:		
Core diameter, in. (equivalent)	132.7	132.7
Core average active fuel height, in.	143.7	144
Reflector Thickness and Composition:		
Top - water plus steel, in.	~10	~10
Bottom - water plus steel, in.	~10	~10
Side - water plus steel, in.	~15	~15
H_2O/U, cold volume ratio lattice	2.43	2.52

Core Mechanical Design Parameters	Trojan 17 x 17 Fuel Assembly with Densification Effects	Trojan 15 x 15 Fuel Assembly without Densification Effects
Fuel Assemblies:		
Design	RCC canless	RCC canless
Number of fuel assemblies	193	193
UO_2 rods per assembly	264	204
Rod pitch, in.	0.496	0.563
Overall dimensions, in.	8.426 x 8.426	8.426 x 8.426
Fuel weight (as UO_2), lbs	222,739	218, 367
Zircalloy weight, lbs	50,913	~44,424
Number of grids per assembly	8-Type R	7-Type L
Loading technique	3 region non-uniform	3 region non-uniform
Fuel Rods:		
Number	50,952	39,372
Outside diameter, in.	0.374	0.422
Diametral gap, in., Regions 1, 2, and 3	0.0065	0.0075
Clad thickness, in.	0.0225	0.024
Clad material	Zircaloy-4	Zircaloy-4
Fuel Pellets:		
Material	UO_2 sintered	UO_2 sintered
Density (% of theoretical)	95	94
Diameter, in., Regions 1, 2, and 3	0.3225	0.3659
Length, in.	0.530	0.600

APPENDIX C

The IIASA world regions.



	Region I	(NA) North America
	Region II	(SU/EE) Soviet Union and Eastern Europe
	Region III	(WE/JANZ) Western Europe, Japan, Australia, New Zealand, S. Africa, and Israel
	Region IV	(LA) Latin America
	Region V	(Af/SEA) Africa (except Northern Africa and S. Africa), South and Southeast Asia
	Region VI	(ME/NAf) Middle East and Northern Africa
	Region VII	(C/CPA) China and Centrally Planned Asian Economies

FIGURE C-1 IIASA World Region Chart

FIGURE C-2 LEOPARD Input Form Sheets

LEOPARD CODE INPUT FORM, SHEET A Page ____ of ____

8-character label Analyst _____

_____ Date _____ LS _____

1. _____

TITLE _____

	a.	Change case	0	(Options always propogate.)
		Reference case	1	
	b.	Temperatures in F°	0	
		Temperatures in R°	1	
		Temperatures in K°	2	
		Temperatures in C°	3	
	c.	Lengths in centimeters	0	
		Lengths in inches	1	
	d.	Square cell	0	
		Hexagonal cell	1	
	e.	Search for L-238	0	
2.		Use input L-238	1	
OPTIONS	f.	Dimensions are hot	0	
		Dimensions are cold	1	(code will adjust)
	g.	σ_{rem} fitted for element*		(Usually enter 1 for hydrogen.)
	h.	Don't punch CANDLE library	0	
		Punch CANDLE library	1	
	i.	Include $\sigma_{a,r}^{3,28}$ in library	0	
		Don't include $\sigma_{a,r}^{3,28}$ in library	1	
	j.	Don't punch CANDLE thermal data	0	
		Punch MND CANDLE thermal data	1	
		Punch conv. CANDLE thermal data	2	
	k.	Punch FAB-2 data for material*		(Supply a number)
	l.	Punch ADM-5 data for material*		(Supply a number)
	m.	2-group MND	0	
		2-group conventional model	1	(See options k and l)
		4-group MND	2	
		4-group conventional model	3	
	n.	No lifetime steps	0	
		Some lifetime steps	1	(Attach sheet C)
	o.	Bypass FAB Doppler fitting	0	
		Save data for FAB fitting	1	(Enter Reactor Height and Rel. Power)
		Perform FAB fitting	2	
	p.	Bypass CNCR Doppler fitting	0	
		Save data for CNCR fitting	1	(Enter Reactor Height and Rel. Power)
		Perform CNCR fitting		

*This option may be bypassed by entering zero.

LEOPARD CODE, SHEET B

Page _____ of _____

Composition Description

3.

Index	Pellet	Clad	Moderator	Extra
777	0.0	0.0	0.0	0.0

Trace Elements

4.

Index	Factor
777	0.0

NOTES

a. For U-235 enter + a/o or - w/o
b. For B-10, enter + ppm to relate to water, - ppm to relate to UO_2
c. For D_2O enter D/H ratio
d. For all else enter a/o UO_2
e. Enter decimals (e.g. 3 a/o → .03)

5.

"Resonance" temperature
Pellet temperature
Clad temperature
Moderator temperature
Buckling
Non-lattice peaking factor (blank → 1.0)

6.

Pellet O.R.
Clad O.R.
Pitch
Power (100% → 1.0) (Used for Doppler fitting only)
Height (Used for Doppler fitting only)
Non-Lattice Fraction

7.

H_2O Pressure
D_2O Pressure
UO_2 Density (g/cc or %)
PuO_2 Density (g/cc or %)
ThO_2 Density (g/cc or %)
+ → Cd Cutoff; - → L-238

INDICES

1	H
2	O
3	Zr-2
4	C
6	Fe
7	Ni
9	Al
11	Cr
15	Mn
18	U-235
19	U-236
20	U-238
21	Pu-239
22	Pu-240
23	Pu-241
24	Pu-242
25	I-135
26	Sm-149
27	Xe-135
28	F.P.
29	B-10
38	D
50	U-233
51	U-234
62	Th-232
65	Pa-233
75	Pr-149
- - -	- - -
97	ThO_2
98	PuO_2
99	UO_2
100	H_2O
101	D_2O
304	SS-304
316	SS-316
348	SS-348

LEOPARD CODE, SHEET C Page ______ of ______

THIS SHEET IS PRESENT IF AND ONLY IF A BURNOUT CALCULATION (OPTION N) IS INDICATED

8.

	Volume in cm^3
	Power in watts
	Loading, grams U (If zero is entered, the code will compute the loading.)

9.

Step No.	Time	Poison
777	0.0	0.0

NOTE:

For time steps

+ → hours

- → MWD/MTU

For poison

+ → ppm boron

- → D_2O/H_2O Ratio

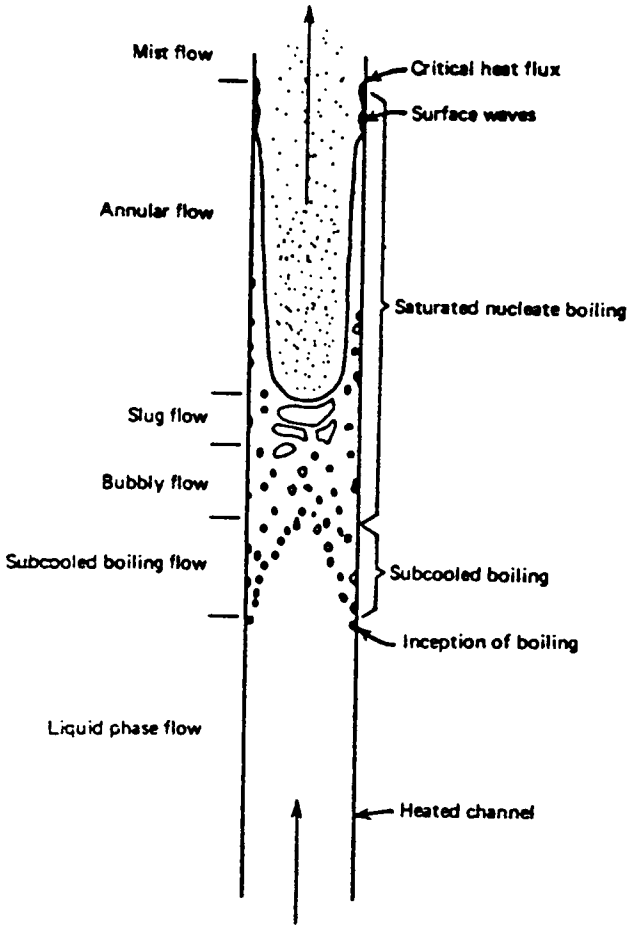


FIGURE C-3 Boiling Channel Flow Patterns

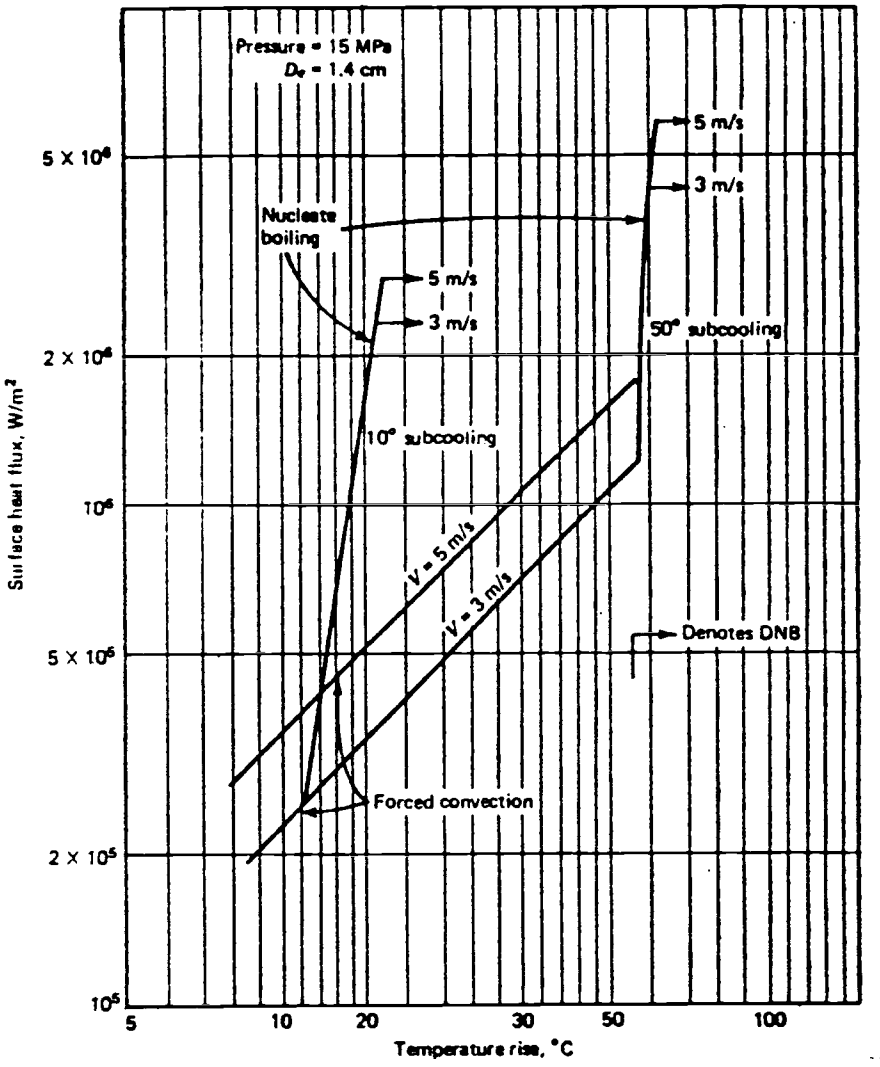


FIGURE C-4 Heat Transfer Mechanisms in PWR's

S.S. LEOPARD

```
TROJAN PWR (W) 20 X 20 SQR. ASSBY: VAR. ENRICHMENT- 3 % U-235
 1  0  1  0  0  0  1  0  0  0  0  0  1  0  0  0  0  0
        99          1.0         0.0         0.0         0.0
         3          0.0         1.0         0.0         0.0
       100          0.0         0.0         1.0         0.0
       777          0.0         0.0         0.0         0.0
        18      -0.0300
       777          0.0
1472.0      1472.0      625.0       600.0       0.000001    1.0
0.16125     0.1870      0.423789    1.0                     0.0
2250.0      0.0         10.2        0.0         0.0
 EOI ENCOUNTERED.
/
```

TRAIN. LEOPARD

```
TROJAN PWR 17 X 17 SQR. - U-235 = 3 % - 50.000 MWD/MTU
 1  0  1  0  0  1  1  0  0  0  0  0  1  1  0  0  0  1  1
        99          1.0         0.0         0.0         0.0
         3          0.0         1.0         0.0         0.0
       100          0.0         0.0         1.0         0.0
       777          0.0         0.0         0.0         0.0
        18      -0.0300
       777          0.0
1472.0      1472.0      625.0       600.0       0.000001    1.08
0.16125     0.1870      0.496000    1.0         144.0       0.090
2250.0      0.0         0.95        0.0         00.0        0.0
1.0         104.0       0.0         0.0         1.0
          1 -10.0
          2 -20.0
          3 -70.0
          4 -400.0
          5 -500.0
          6 -1000.0
          7 -1000.0
          8 -2000.0
          9 -2500.0
         10 -2500.0
         11 -5000.0
         12 -5000.0
         13 -5000.0
         14 -5000.0
         15 -5000.0
         16 -5000.0
         17 -5000.0
         18 -5000.0
        777 0.0
 EOI ENCOUNTERED.
/
```

COBRA

```
1500
   1   1        TROJAN PWR 1 ROD -20120 SQR LATT- PUMP COAST-DN 50% - CHUCK K.
   1  18    0    0
 14.7211.9   .01672    26.800    180.15    1150.50    .75234    .39059   .003792
 100.327.9   .01774    4.4340    298.61    1187.80    .51171    .39247   .002990
500.0467.1   .01975    .92830    449.50    1205.30    .27771    .36513   .002026
1000.544.8   .02159    .44590    542.40    1192.40    .23413    .33125   .001489
1250.572.6   .02250    .34540    578.60    1181.60    .22044    .31512   .001296
1500.596.4   .02346    .27690    611.50    1168.70    .20774    .29977   .001131
1750.617.3   .02450    .25680    642.30    1153.20    .19890    .28762   .000986
2000.636.0   .02565    .18813    671.90    1136.30    .19006    .27547   .000857
2100.642.9   .02616    .17491    683.60    1128.50    .18657    .27061   .000809
2200.649.6   .02670    .16270    695.30    1120.30    .18310    .26597   .000763
2300.656.1   .02728    .15133    707.00    1111.40    .18003    .26198   .000718
2400.662.3   .02791    .14067    718.80    1101.80    .17697    .25800   .000675
2500.668.3   .02860    .13059    730.90    1091.40    .17389    .25401   .000633
2750.682.5   .03077    .10717    763.00    1060.40    .16622    .24405   .000536
2850.687.8   .03192    .09811    777.30    1045.00    .16315    .24007   .000499
3000.695.5   .03431    .08404    802.50    1015.50    .10029    .11920   .000445
3100.700.5   .03701    .07322    825.40     986.80    .09351    .11881   .000411
3200.705.3   .04805    .05444    890.60     919.90    .10164    .13405   .000378
   2   0    0   0    1   1
 .079 -.25
  16. -.50
   3  11
  0.0 .150  0.1 .868  0.21.248  0.31.375  0.41.395  0.51.315
  0.61.201  0.71.040  0.80.840  0.90.525  1.00.125
   4   2   2
0   1.06971.1751.175    2.0498
0   2.06971.1751.175    1.0498
   7   2   0   8   1   3   0
0.020    10.180    10.320    10.450    10.650    10.720    1
0.860    10.990    1
   1 0.30
   2 0.30
   8   1   1   2   1   2   1   1
0   1.3740 1.00   1 1.00   2 0.00
3.939.0590650.4.32259.2500.078409.5.02251200.
      0.0 -4.5790E-4 7.6231E-8
   9   0   1   0   0   0
 144. 35.
  12  35
  10   2   0   0
0.079 -.250
  11   1   1   0   0  10   0   0
    2250.     552.5    3.6590   .137279
0.00  1.00 1.50 0.95 2.50 0.90 5.10 0.85 8.10 0.80 11.5 0.75
15.1  0.70 20.1 0.65 25.1 0.50 35.0 0.50
  12   3   0   0   7   0   0   0   0

 EOI ENCOUNTERED.
/
```

FIGURE C-5 COBRA-IV & LEOPARD Input Decks