4

of

1

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

Final Report To The
U. S. Department of Energy
on
The Evaluation of the Use of
Metal Alloy Fuels in Pressurized
Water Reactors

by

Dr. Dale Lancaster
Nuclear Engineering Program
School of Mechanical Engineering
Georgia Institute of Technology
Atlanta, Georgia 30332-0225

October 26, 1992

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

TABLE OF CONTENTS

	Page
I. Executive Summary	3
II. Introduction	4
III. Fuel Material Evaluation	6
IV. Fuel Pin Design	7
V. Design Optimization	8
VI. Safety Comments	11
VII. Fuel Cycle Possibilities	12
VIII. Conclusions	13
APPENDIX A: Erwin's M.S. Thesis	
APPENDIX B: Choi's Qualifier	
APPENDIX C: Pfeifer's Qualifier	

*Appendices left in because
of context*

I. Executive Summary

The use of metal alloy fuels in a PWR was investigated. It was found that it would be feasible and competitive to design PWRs with metal alloy fuels but that there seemed to be no significant benefits. The new technology would carry with it added economic uncertainty and since no large benefits were found it was determined that metal alloy fuels are not recommended.

Initially, a benefit was found for metal alloy fuels but when the oxide core was equally optimized the benefit faded. On review of the optimization of the current generation of "advanced reactors," it became clear that reactor design optimization has been under emphasized. Current "advanced reactors" are severely constrained. The AP-600 required the use of a fuel design from the 1970's. In order to find the best metal alloy fuel design, core optimization became a central effort. This work is on going.

II. Introduction

There are four key reasons that it is a good time to investigate the use of metal alloy fuels in PWRs. First, the US LMR community has decided that the fuel of choice is a metal alloy with 10% Zr and the rest heavy metals. Second, the AVLIS process uses a metal form, thus fuel cycle components have to be changed for AVLIS. Third, the timing is consistent with the design of the "next generation" of reactors. Finally, pyrochemical reprocessing associated with LMRs may make metal alloy fuels preferable especially in connection with actinide burning.

The fast reactor community has selected a sodium bonded fuel design since it resulted in less stored heat for transients. This less stored heat is due to the high thermal conductivity in the alloy. This benefit would presumably exist in a PWR with metal alloy fuel. The sodium bond will not be used in the PWR due to the energetic reaction of sodium and water. The amount of sodium would not be a significant contributor to the energy release in any nuclear transient; however, sodium bonding will not be used to minimize concerns. The fast reactor community has generated a considerable data base for the swelling of the metal alloy fuel with burnup. Accommodations are made for this swelling by leaving about 30% of the interior cross

sectional area of the pin for the sodium bond. Fairly rapidly the fuel swells to almost a sponge like material which is easily contained by the clad. The PWR design to meet these challenges will be presented in a subsequent section.

Future enrichment facilities in the US (after the Louisiana project) will probably use Atomic Vapor Laser Isotopic Separation (AVLIS) techniques. This process vaporizes metallic uranium. Since the product of the enriching process is no longer UF_6 , modifications will have to be made at the fuel fabrication plants or a conversion process added on to the end of AVLIS. Before these modifications are made it is logical that the fuel form be reevaluated. There are possible savings from never leaving the metal form.

A good deal of the effort toward the next generation of reactors has been in finding passive safety features. The metal alloy fuel holds out the potential for less stored heat. The metal alloys also tend to have a higher isotopic density and hence it may be reasonable to have larger coolant channels. These two features suggest an easier transition to natural circulation cooling.

In the last few years there has been extensive work on pyrochemical techniques for reprocessing. Most of this work

has been done for the metal alloy used in EBR-II. If the PWRs were to use metal alloy fuels, it maybe possible to easily extend this work to allow for reprocessing PWR fuel. In fact, if Integral Fast Reactors (IFRs) are to be built it maybe possible to use their on site reprocessing capability. This would eliminate the need for a separate PWR reprocessing facility. Actinide burning would be made much easier.

III. Fuel Material Evaluation

The fuel material that would best match the experimental database was 10% Zr and 90% actinides (ie: U or Pu). It was known at the start of this work that alloying uranium with zirconium would retard its reaction with water. Unfortunately, experiments done in the mid 1950's showed that a complete fuel pin of this alloy would decompose at power conditions in much less than a week even if the fuel failure was a pinhole. The fuel corrosion rate as a function of zirconium content is summarized in Table 1. It was decided that it would be unacceptable to attempt to shut down and retrieve any damaged pin and it was further decided that the complete dissolving of just one fuel pin would produce unacceptable amount of radiation dose during shutdown and refueling. Although fuel pin reliability has been increasing, it was decided that the minimum alloying

Table 1. Corrosion Rates of Fuels (Summary)

Tested Fuel	Temp. (°F)	Sample Shape & Condition	Corrosion Rate (mg/cm ² hr)	Ref.
Pure U	587	Pin /Unclad	6260	BMI-1156
U- 5w% Zr	587	"	3150	"
U-10w% Zr	595	"	2780	"
U-15w% Zr	604	Pin /Unbonded	165	"
U-20w% Zr	610	"	194	"
U-50w% Zr	600	Unclad	0.025	BMI-874
"	600	"	0.1	WAPD-T-416
U oxide	650	Pin/Clad	~0.00135	WAPD-184

content would be set at 50% zirconium. Some work was done on reviewing other alloys and ternary alloys but there seemed to be no benefits over the 50% zirconium case. The MS thesis in Appendix A evaluated zirconium alloys of 50% to 99%.

IV. Fuel Pin Design

Since metal alloy fuels swell with irradiation, a new fuel pin design is required. Since metal alloy fuels melt at a much lower temperature it is not possible to allow for swelling by a gap between the fuel and the clad. In order to accommodate the swelling it was decided to coextrude the fuel with the cladding with a central void for the swelling. One could pour spheres of fuel into the fuel clad, but this option is more complex without any known advantages. Fuels for the LMR program have had 30% of the cross sectional area for swelling, however, one experiment showed acceptable results with only 15% cross sectional area for swelling (Personal Communication with Walters at Nov. 1992 ANS meeting)). Since 50% Zr is the minimum allowed for this work (contrasted to 10% Zr in the LMRs) it was decided that 15% swelling space would be adequate. Figure 1 shows the fuel pin design selected.

Figure 1: Schematic of a Metallic Fuel Pin

V. Design Optimization

Since the metal alloy fuel does not contain the same amount of uranium per pin as the oxide fuel, the fuel pin pitch must be modified to maintain an appropriate hydrogen to uranium ratio. Alternatively, the fuel pin diameter could be modified so the fuel pin pitch could be maintained. Any change of this type would change the thermal-hydraulic nature of the fuel. This forced the project to find an optimization approach that would identify the best design for the various alloys investigated. To the surprise of the project, although numerous new plant designs are under consideration no modern work on core optimization techniques exists. The first Masters thesis performed for the project identified an approach. The second thesis, Appendix A, automated this approach and then utilized it to find optimum designs for more alloys. (Please see appendix A for a description of the approach.) A PhD project is on going, which will improve the model and use state of the art optimization methods. Appendix B is the written qualifier for this student. A paper has been presented at an ANS national meeting on utilizing the optimization technique for an oxide fueled PWR.

The optimization work shows that there is an optimum power density that is less than the maximum achievable. Historic design optimization designed to thermal limits.

This was appropriate due to the fuel cycle expected at the time (recycling as compared to once through) and the higher uncertainties. The optimum power density is slightly lower than that used in AP600, and considerably less than in the current generation of reactors. The dominant factor influencing the optimum power density is fuel cycle considerations. The key variables here are burnup and cycle length. Two year cycle lengths challenge the burnup capability of the current generation fuels and thereby favor lower power densities.

Table 2 shows the characteristics of the best metal alloy fuel design and compares it to AP600. As can be seen by the table there is hardly an economic reason for a change to metal alloy fuels. The following sections address other reasons.

TABLE 2: Optimized Design Comparison

	AP600	50% Zr Metal Alloy
Pin Diameter (mm)	9.5	19.1
Pitch (mm)	12.6	21.9
Length (m)	3.66	4.29
Fuel Pins	38280	17398
Core Volume (l)	24532	39531
Containment Vol. (1×10^6)	50.66	54.31
MTU	67.71	80.07
Initial Enrichment	2.7	2.4
Specific Power (W/gm)	27.81	23.51
Linear Power (kw/m)	13.4	25.2
Cycle Burnup (MWD/MTU)	15621	13209
Batches per Core	2.88	3.41
Pressure Drop (kpa)	90.61	93.99
Pressure Drop (kpa/m)	24.77	21.91
Refuel Enrichment	4.4	4.2
Peak Fuel Temp. (F)	2920	1320
Total Cost of Components That are Design Dependent ($\$ \times 10^6$)	342	323

VI. Safety Comments

The metal alloy design selected has the same DNBR as the oxide core. Most accidents in PWRs are DNB limited and for all these accidents there is no difference between the metal alloy and oxide designs. The removing of oxygen and the addition of zirconium has very little effect on reactivity, so there is no major safety difference due to neutronics. Except for very fast acting transients, the stored heat in the fuel is dominated by the decay heat which would be indifferent to design.

For some transients the thermal margin to melting is important. The metal alloy core starts at a lower temperature but also has a lower melting temperature. In net there is no major difference between the thermal margin between the oxide and metal alloy fuel.

For a LOCA, the stored heat in the fuel is important during the flow transition. The low amount of stored heat in the metal alloy fuel does eliminate the concern for this period of the accident. There are benefits due to this. Oxide fueled PWRs have been able to handle the transition phase of LOCA. For unanticipated accidents that result in

fuel melting there is a concern of steam explosions with the metal alloy fuel due to the superior thermal conductivity.

Although metal alloy fuels do have a lower stored heat, they also have a lower melting point. It has not been possible to show a clear safety advantage for the metal alloy fuels but rather each fuel has different safety concerns.

VII. Fuel Cycle Possibilities

Under the current once through fuel cycle there may be some advantages to metal alloy fuels by matching better with AVLIS but the advantage of making a new mixture (alloying at 50% zirconium) versus a chemical reaction (oxidizing the product) is very small. Further, until AVLIS becomes the standard for fuel enrichment, the new conversion process required from UF_6 would represent a strong disadvantage.

If the IFR wins acceptance, it would be desirable to use PWR fuel as feed. Using metal alloy fuel it may be possible to use the on site reprocessing facility. Unfortunately, the PWR fuel would contain 50% zirconium rather than the 10% expected for the IFR fuel. This is a non-trivial problem for the pyrochemical reprocessing but it would seem to be easier than reprocessing oxide fuel. If

oxide fuel is to be used as feed for the IFR, a separate reprocessing facility would have to be built.

Actinide burning has recently been explored as a way of decreasing concerns with long term waste disposal. If the actinide burning is to be done in an IFR type reactor the above paragraph covers the subject. Actinides can be burned in a PWR. In this case, the metal fuel would allow a thermal IFR type concept.

One of the PhD students initiated work on a thesis involving the metal alloy fuel cycle. The student has since changed his thesis topic but his qualifier is attached as Appendix C.

VIII. Conclusions

Metal alloy fuels can be used in a PWR. The fuel form that would be acceptable would contain 50% zirconium. This is different than that used in the LMR research and as such would require a significant development program. The economic analysis found that there was no large advantage or disadvantage to metal alloy fuels. Thus the development program and initial costs would prohibit actual implementation of metal fuel. From a safety point of view there was no significant benefit from metal alloy fuels.

Although the stored heat was less, the melting temperature was also lower. If the IFR is accepted there is a significant benefit to the metal alloy fuel due to the possible elimination of a reprocessing facility. If PWR actinide burning is accepted the metal alloy fuel has a significant benefit due to pyrochemical reprocessing.

Initially, a benefit was found for metal alloy fuels but when the oxide core was equally optimized the benefit faded. On review of the optimization of the current generation of "advanced reactors," it became clear that reactor design optimization has been under emphasized. Current "advanced reactors" are severely constrained. The AP-600 required the use of a fuel design from the 1970's. In order to find the best metal alloy fuel design core optimization became a central effort. This work is on going.

From an administrative point of view there have been considerable difficulties with this contract. Three faculty were listed with the first proposal. All three have now left Georgia Tech. A fourth faculty was added in the renewal proposal. That faculty member has also left Georgia Tech. In at the time the contract was given to Georgia Tech there were ten faculty in the Nuclear Engineering Program. Seven of those have left Georgia Tech (none retired) and one is actively looking to leave. A stable program would have

aided in the research. The principal investigator, Dale Lancaster, is now at Penn State.

APPENDIX A

**Feasibility and Determination of a
Binary Uranium - Zirconium System for use
In Pressurized Light Water Reactors**

**A Thesis
Presented to the Faculty by
Charles S. Erwin**

**In Partial Fulfillment
of the Requirements for the Degree
Master of Science - Nuclear Engineering**

FG02-89ER12898
**School of Mechanical Engineering
Nuclear Engineering and Health Physics Program
Georgia Institute of Technology
November 1991**

**Feasibility and Determination of a
Binary Uranium - Zirconium System for use
In Pressurized Light Water Reactors**

APPROVED:

Dale B. Lancaster, Chairman

Daniel B. Bullen

S. Mostafa Ghiaasiaan

Date Approved by Chairman _____

For My One and Only One

ACKNOWLEDGEMENTS

The author would like to thank the members of his advisory committee for their support and direction. First, the author thanks Doctor Daniel B. Bullen, whose metallurgical expertise gave the author new insights into binary fuel compositions and corrosion control. Secondly, the author thanks Doctor S. Mostafa Ghiaasiaan, whose schedule included daily intrusions from the writer to find an appropriate DNBR correlation. Last, but absolutely not least, the author thanks his thesis advisor, Doctor Dale B. Lancaster. Doctor Lancaster's efforts in discussing the project, suggesting new ideas, and editing the final manuscript went beyond extraordinary.

The author also bestows acknowledgement to the Department of Energy for providing research funding for metallic fuels in pressurized light water reactors. He also would like to thank his fellow colleagues, H. Pfeifer, K. Choi, and J. Malo, for bringing the author up to speed on the project and for providing insight throughout the research.

Thanks are extended to the Georgia Power Company, Land Engineering Section, for allowing the author to use their equipment and facilities. He especially thanks R.M.

ACKNOWLEDGEMENTS

Kirby and J.B. Thacker for allowing him to continue his education.

The author further wishes to thank the many people who provided moral support and believed in the author's abilities throughout the project. A special thanks to his mother, Jo Yarbrough, his father, Dan Erwin, Allen and Becky Lewis, and the special few he calls 'riends.

In addition, a special show of gratitude and love is extended to his fiancée, Rebecca. Her patience with the author, during the writing of this manuscript, while they planned their wedding, was phenomenal. Without her pushing, prodding, inspiration, and especially her love, this project, and his subsequent degree, may have only been a dream.

TABLE OF CONTENTS

Approvals	ii
Dedication	iii
Acknowledgements	iv
List of Tables	ix
List of Figures	xi
Summary	xiii
Chapter I: Introduction	1
1.1 Historical Perspective on Metal Fuels	1
1.2 Advancements In Metallic Fuels	2
1.3 Why use Metallic Fuels in Light Water Reactors	3
1.3.1 Current Light Water Reactor Technology	3
1.3.2 Enrichment and Fabrication	4
1.3.3 Metallic Fuel Cycle	5
1.3.4 Operational Safety	7
1.4 Objective of Study	8
1.5 Organization of Report	9
Chapter II: Procedures	11
2.1 Reference Design	11
2.1.1 AP-600 Data	11
2.1.2 Thermal Analysis of Reference Core	12
2.2 Design Assumptions	14
2.2.1 Steam Cycle	15

TABLE OF CONTENTS

2.2.2	Neutronics	15
2.2.3	Thermal Operating Margin	16
2.2.4	Fuel Material	16
2.3	Limiting Constraints	17
2.3.1	Fuel Corrosion	17
2.3.2	Geometry	24
2.3.3	Fuel Phase Transition	27
2.4	Method for Metallic Core Analysis	27
2.4.1	Design Analysis Techniques	28
2.4.2	Cost Assumptions for Optimization	32
2.4.3	Discussion of Method	36
Chapter III:	Optimized Design Results	38
3.1	Results of Optimization	38
3.1.1	Designs	38
3.1.2	Costs	45
3.1.3	Cost Comparison	58
3.1.4	Design Comparison	62
3.1.5	Fuel Temperatures	66
Chapter IV:	Sensitivity of Results	73
4.1	Assumption Sensitivity	73
4.1.1	Ore Costs	73
4.1.2	Enrichment Costs	77
4.1.3	Cycle Length	80
4.1.4	Discharge Burnup	84

TABLE OF CONTENTS

Chapter V: Conclusions	88
5.1 Recommendations and Discussion	88
5.1.1 Design Recommendation	88
5.1.2 Discussion of Recommendation	89
5.1.3 Future Work	92
References	93
Appendix A:	97
ITMETAL.BAT listing	97
DICE4 code listing	98
Appendix B:	110
Typical COBRA-IV-PC input file	111
Typical COBRA-IV-PC output file	113
Appendix C:	120
Spreadsheet parameter description and formulae	121

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page No.</u>
1	Accumulated Data for AP-600	13
2	Corrosion Rates of Metallic Fuel Alloys	20
3	Thermal Conductivity as a Function of Alloying Percentage @ 450°C	31
4	Spreadsheet Cost Optimization Parameters	35
5	Design Parameters of Various Pin Diameters for 50 wt. % U - 50 wt. % Zr	39
6	Design Parameters of Various Pin Diameters for 45 wt. % U - 55 wt. % Zr	40
7	Design Parameters of Various Pin Diameters for 40 wt. % U - 60 wt. % Zr	41
8	Design Parameters of Various Pin Diameters for 35 wt. % U - 65 wt. % Zr	42
9	Design Parameters of Various Pin Diameters for 30 wt. % U - 70 wt. % Zr	43
10	Design Parameters of Various Pin Diameters for 25 wt. % U - 75 wt. % Zr	44
11	Comprehensive Cost Listing of Various Pin Diameters for 50 wt. % U - 50 wt. % Zr	47
12	Comprehensive Cost Listing of Various Pin Diameters for 45 wt. % U - 55 wt. % Zr	49
13	Comprehensive Cost Listing of Various Pin Diameters for 40 wt. % U - 60 wt. % Zr	51
14	Comprehensive Cost Listing of Various Pin Diameters for 35 wt. % U - 65 wt. % Zr	53
15	Comprehensive Cost Listing of Various Pin Diameters for 30 wt. % U - 70 wt. % Zr	55

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page No.</u>
16	Comprehensive Cost Listing of Various Pin Diameters for 25 wt. % U - 75 wt. % Zr	57
17	Comprehensive Cost Listing for Optimized Alloying Contents	61
18	Design Parameters for Optimized Alloying Contents	65

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page No.</u>
1	A Fuel Cycle Possible for a Metal Fueled PWR	6
2	Schematic of a Metallic Fuel Pin	18
3	Double-arc-melted U-Zr Alloys Exposed to Water at 600°F	21
4	Corrosion Rates Versus Water Temperature for U-Zr Alloys	22
5	Corrosion Rates Versus Uranium Content for U-Zr Alloys	23
6	Feasible and Unfeasible Fuel Cell Configurations	26
7	Pin Diameter Optimization Flow Chart	37
8	Total Costs Versus Pin Outer Diameter for 50 wt. % U - 50 wt. % Zr	46
9	Total Costs Versus Pin Outer Diameter for 45 wt. % U - 55 wt. % Zr	48
10	Total Costs Versus Pin Outer Diameter for 40 wt. % U - 60 wt. % Zr	50
11	Total Costs Versus Pin Outer Diameter for 35 wt. % U - 65 wt. % Zr	52
12	Total Costs Versus Pin Outer Diameter for 30 wt. % U - 70 wt. % Zr	54
13	Total Costs Versus Pin Outer Diameter for 25 wt. % U - 75 wt. % Zr	56
14	Total Costs Versus Weight Percent Zirconium	60
15	Hot Spot Radial Temperature Profile of Optimized 50 wt. % U - 50 wt. % Zr Alloy	67

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page No.</u>
16	Hot Spot Radial Temperature Profile of Optimized 45 wt. % U - 55 wt. % Zr Alloy	68
17	Hot Spot Radial Temperature Profile of Optimized 40 wt. % U - 60 wt. % Zr Alloy	69
18	Hot Spot Radial Temperature Profile of Optimized 35 wt. % U - 65 wt. % Zr Alloy	70
19	Hot Spot Radial Temperature Profile of Optimized 30 wt. % U - 70 wt. % Zr Alloy	71
20	Hot Spot Radial Temperature Profile of Optimized 25 wt. % U - 75 wt. % Zr Alloy	72
21	Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - \$10/LB U ₃ O ₈	75
22	Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - \$80/LB U ₃ O ₈	76
23	Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - \$40/SWU Enrichment	78
24	Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - \$120/SWU Enrichment	79
25	Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - 18 Month Cycle Length	82
26	Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - 36 Month Cycle Length	83
27	Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - 30 000 MWD/MTU Burnup	86
28	Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - 80 000 MWD/MTU Burnup	87

SUMMARY

In early studies of nuclear fuels for pressurized water reactors, a variety of materials were considered, including metallic fuels. Oxides were eventually chosen and are used as the current commercial nuclear fuel. Recent advancements in metallic fuels at EBR-II and FFTF make it attractive to consider these fuels once again. Other reasons to consider metallic fuels include; the AVLIS enrichment process, the possibility of a coupled fuel cycle with the IFR system, pyroprocessing and actinide recycling, and operational safety features of metallic fuels.

This study examines the feasibility of designing a metal-alloy fueled reactor. The work considers all alloying percentages (1% to 99%) of a binary uranium - zirconium system. Alloys below fifty (50) weight percent zirconium were eliminated due to corrosion rates with heated water. Alloys above seventy-five (75) weight percent zirconium were discarded due to the inability of maintaining a hydrogen to uranium (moderation) ratio in a square lattice.

The optimization of a pin diameter was performed on alloys between fifty (50) and seventy-five (75) weight percent zirconium at five (5) weight percent increments. From the optimized designs of all alloying percentages, it

SUMMARY

is recommended that any further work in the uranium - zirconium binary system for use in pressurized light water reactors be performed on the fifty (50) weight percent uranium - fifty (50) weight percent zirconium alloy. Not only does this design have the lowest total costs, it also exhibits design parameters that most closely resemble current nuclear reactors. A potential problem with this alloy is that the fuel phase transition temperature was exceeded during a thermal analysis of the fuel.

CHAPTER I

INTRODUCTION

1.1 Historical Perspective on Metal Fuels

When commercial light water reactor designs were initially studied, many fuel types were considered. Some of these included carbides, nitrides, metal alloys, and, the current commercial fuel, oxides. There were many reasons that oxides were chosen over the other fuel materials. These include phase changes, pore migration (swelling), material-to-material reactions, and material-to-coolant reactions to name just a few.

There were three primary reasons metallic fuels were rejected as a viable fuel design. First, the metal alloys underwent substantial phase changes and swelling due to the effects of irradiation. [1,2] Second, the effects of corrosion due to the chemical reaction of the uranium metal with the high temperature water coolant were considered to be too great in the event of a cladding breach. The corrosion rate of the metallic fuel, compared with the oxides, were considered to be too high. It was also perceived in the reactor community that metallic fuels could not achieve

high enough burnups necessary for economic operation. [3] However, there has been renewed interest and research concerning metal fuels. This is due to research at EBR-II [4], the Fast Flux Test Facility (FFTF) [5], and developments in the Integral Fast Reactor (IFR) program [6].

1.2 Advancements in Metallic Fuels

The current fuel design employed by the IFR program is a metallic fuel alloy. The fuel is a uranium/zirconium alloy containing 90 percent by weight uranium. Through reactor research, metallic fuels have made several advances.

The drastic phase changes and accompanying swelling due to irradiation effects can be controlled with the proper use of alloying materials. [7] As noted above, the current alloying material is zirconium. The zirconium acts to stabilize the gamma phase of the uranium metal. [8] This stabilization reduces the swelling and limits the volume changes.

There has been limited research into water induced corrosion in metallic fuels. This is due to the use of liquid metal coolants by the fast reactor community. Research performed when metal fuels were first considered for light water reactors (LWRs) indicates that the use of an alloying agent in the uranium helps reduce the corrosion

rates of the fuel in water. [9]

The economic problems associated with low burnup were basically unfounded. Design developments at EBR-II in the 1970's have shown that metallic fuels can achieve high burnups. It has also been shown that metallic fuels perform well in irradiation environments, exhibiting very limited swelling or dimensional changes. [3]

1.3 Why Use Metallic Fuels in Light Water Reactors

This section of the report tells of the current LWR technology and conveys reasons for using metallic fuels. These reasons include enrichment and fabrication, the fuel cycle, and operational safety of a metal-fueled reactor core.

1.3.1 Current Light Water Reactor Technology

The current fuel design of the pressurized light water reactor community is an oxide. Specifically, uranium dioxide (UO_2) is used by the commercial industry. All enrichment of uranium in the United States is done with the gaseous diffusion process. To get the fuel into the final form, uranium is chemically combined with oxygen and the oxide powder is compacted and sintered into pellets. There are a number of areas where a metallic fuel could provide

advantages over an oxide fuel. One area includes the enrichment and fabrication processes. Another area where advantages are possible is in the fuel cycle. A final advantage could result from enhanced safety in the operation of the plant with a metallic fuel instead of an oxide.

1.3.2 Enrichment and Fabrication

The Department of Energy decided to adopt Atomic Vapor Laser Isotope Separation (AVLIS) as the method for future uranium enrichment. Since the feed and product of AVLIS are an alloyed uranium metal, it is attractive to evaluate a metal fueled pressurized water reactor (PWR).

Current enrichment costs for gaseous diffusion techniques are approximately \$100/SWU. [10] AVLIS enrichment costs are expected to achieve a levelized SWU cost below \$40/SWU. [11] It should be noted that if all enrichment were performed by the AVLIS process, an additional \$15/SWU would be necessary to convert to an oxide. [11] For a metallic fuel, current alloying techniques could be employed to obtain a near zero additional processing cost. [11] There would also be safety advantages involved by using a metallic fuel. The enrichment and alloying processes would not involve the use of toxic fluorine, which is employed in the enrichment and chemical conversion of oxide

fuel.

A metallic fuel could also have fabrication advantages. Instead of compacting and sintering UO_2 powder into pellets, the metallic fuel could be cast or coextruded with the cladding. Not only could this be a cost benefit, but it could also be a safety benefit. Casting or coextrusion of the fuel could decrease the amounts of uranium powder in the environment of the fuel fabrication facility.

1.3.3 Metallic Fuel Cycle

Currently, the United States employs a once-through fuel cycle. When and if the United States decides to employ a fuel cycle that includes recycling, an oxide fueled core would most likely be processed using PUREX chemical techniques. With pyroprocessing, utilized in the IFR concept, "direct production of a metal product avoids expensive and cumbersome chemical conversion steps that would result from use of the conventional PUREX solvent extraction process." [3]

Another advantage of metal fueled cores could be realized from actinide recycling. Actinides can be separated with the uranium/plutonium product stream much easier with the aforementioned pyroprocessing than with the PUREX process. [3] Actinide recycling would help reduce the effective lifetime of some of the nuclear wastes from

Figure 1: A Fuel Cycle Possible for a Metal Fueled PWR

millions of years to a few hundred years. [3] This would not only increase engineering confidence factors for a repository, it would also most certainly help the nuclear power industry with public relations. Figure 1 shows a potential metallic fuel LWR/IFR fuel cycle.

These fuel cycle advantages could also create cost benefits. Pyroprocessing would eliminate the need for the PUREX process. Also, the LWR industry could obtain possible fuel credit from the IFR's need for plutonium in their product stream. The actinide recycle of the LWR fuel would also possibly help reduce "long range" engineering and design costs associated with keeping spent fuel for thousands of years in a repository.

1.3.4 Operational Safety

There are two possible operational safety advantages realized with metal fueled cores. First, the fuel density of metallic fuels is greater than current oxide fuels for metal alloying percentages greater than 50 weight percent uranium. This implies the use of larger coolant channels, which would possibly help with natural circulation. This is due to a lower pressure drop per foot in the core. Second, depending on the metal alloying content, the thermal conductivity of a metallic fuel is approximately 5 to 10 times larger than uranium dioxide at reactor operating

temperatures. This translates into a lower radial temperature profile. The lower temperature profile signifies lower centerline and average temperatures in the fuel pin. This represents less stored energy in the fuel pin. "In fact, the peak temperature is less than the 2200°F required for a rapid Zr-H₂O reaction." [12] These operational safety benefits could also lend themselves to cost advantages.

The lower pressure drop per foot, due to the larger coolant channels, produces a cost benefit from having to use less pumping power. As a result of the stored energy being less in the fuel pins, it may be possible to reduce the size of some safety systems. It may also be possible to eliminate these safety systems. The reduction or elimination of any system would be a cost benefit.

1.4 Objective of Study

Fuel pin optimization techniques for a uranium/zirconium alloy-fueled reactor were previously demonstrated. [13] This previous study showed how optimization techniques can be applied to a 90 weight percent uranium - 10 weight percent zirconium fuel to find the optimum fuel pin diameter and pitch.

The current study has been performed to find an

appropriate design for a uranium - zirconium alloy-fueled PWR. This study considers all fuel alloying contents (1 wt. percent to 99 wt. percent). Any alloying contents that are unfeasible are eliminated from the study. From the remaining possibilities, an optimum design is found for each alloying content at five weight percent increments. From these choices, the "best" overall design is selected as the prime candidate for a metal fueled PWR.

1.5 Organization of Report

Chapter I of this report has introduced the reader to metal fuels. It has presented the historical perspective of metal fuels. Chapter I has also discussed advances with metal fuels and why it is advantageous to consider metallic fuels for use in LWRs.

Chapter II outlines the procedures that were taken to perform this study. This chapter discusses the reference design, assumptions necessary to make the task amenable, limiting constraints of a metal fueled LWR, and reviews the method of attack for metallic fuel analysis.

Chapter III introduces the results of the optimization process. It gives detailed breakdowns of the costs used in the optimization process. Chapter III establishes and discusses design criteria and parameters. Chapter III also shows radial temperature profiles for the "hot spots"

in the metallic fuel. These are used to further limit the feasible designs.

Chapter IV shows how sensitive the results are to some key cost parameters. The parameters considered in this study are ore and enrichment costs, cycle length, and discharge burnup.

Chapter V is the conclusions portion of this study. A recommendation for the appropriate design of a metal fueled LWR is given. The chapter discusses this recommendation in regards to the strengths and weaknesses of the study. This chapter also discusses future dimensions of the project to be considered.

Appendix A gives a listing of ITMETAL.BAT and a detailed description of the program, DICE4. These programs were developed in an effort to automate the optimization process. Appendix B shows typical input and output files for the COBRA-IV-PC program. Finally, Appendix C lists the spreadsheet parameters and formulae for the costs of AP-600 and a typical optimization of a metal-fueled core.

CHAPTER II

PROCEDURES

2.1 Reference Design

The purpose of this study is to find a feasible design for a metal-fueled PWR. For comparison purposes, a PWR blueprint is necessary as a reference design. The Department of Energy has chosen the Westinghouse AP-600 advanced PWR as a candidate design for light water reactors. For this reason, the AP-600 [14] reactor will be used as a guideline for the design of the metal-fueled reactor.

2.1.1 AP-600 Data

Although the AP-600 design is being constantly revised, the latest available design data [15] were utilized for this investigation. Some important parameters necessary to complete the study were; fuel type, fuel assembly type, fuel rod and fuel assembly dimensions, number of assemblies, and reactor power. It was also necessary to obtain some important coolant parameters. These include; system pressure, core flow rate, core by-pass percentage,

and the core inlet and outlet temperatures.

Once it was known that the fuel design was a standard Westinghouse UO_2 17 x 17 assembly, the necessary dimensions were identified [16] and calculations essential to the scope of the reference design were completed. These calculations included the hydrogen to uranium (H/U) ratio, the height to diameter (H/D) ratio, and the core lattice (or non-lattice) fraction.

In addition to these calculations, it was also necessary to calculate parameters to perform a thermal analysis. These calculations included the channel flow area, hydraulic diameter, channel mass flux, and average heat flux.

The formulae for the above calculations are not presented in this thesis. They can be found in most books concerning nuclear engineering design. Table 1 furnishes a summary of the aforementioned calculations.

2.1.2 Thermal Analysis of Reference Core

Utilizing the data in Table 1, the next step of the process involved performing a thermal analysis of the reference core. The thermal evaluation of the reference core is necessary for comparing the AP-600 core to the metal-fueled core. The thermal analysis comparison of the metal core to the AP-600 core will be considered in Section 2.2.3 (Thermal Operating Margin).

Table 1: Accumulated Data for AP-600

GIVEN PARAMETERS:	DATA:
Reactor Power (MWth)	1933
System Pressure (psia)	2250
Fuel Type	UO ₂
Fuel Assembly Type	Wshe 17x17 Std
Number of Assemblies	145
Number of Fuel Rods per Assembly	264
Assembly Width (inches)	8.434
Active Length (inches)	144
Rod Pitch (inches)	0.496
Rod Diameter (inches)	0.374
Clad Material	Zircaloy-4
Clad Thickness (inches)	0.0225
Diametrical Gap (inches)	0.0065
Fuel Pellet Diameter (inches)	0.3225
Core Flow Rate (pounds/hour)	7.360E+07
Core Bypass (percent)	7.5
Core Inlet Temperature (Deg.F)	529.1
Core Exit Temperature (Deg.F)	598.7
CALCULATED PARAMETERS:	DATA:
Height-to-diameter Ratio	1.252
Hydrogen-to-uranium Ratio	4.802
Core Non-lattice Fraction	0.0939
Channel Flow Area (square inches)	0.1362
Hydraulic Diameter (inches)	0.4635
Channel Mass Flux (pounds/hour-square foot)	1.880E+06
Average Heat Flux (BTU/hour-square foot)	1.429E+05
Minimum Departure from Nucleate Boiling Ratio	3.26

The thermal analysis of the AP-600 design was performed using the PC based computer code, COSRA-IV-PC [17]. The data from Table 1, along with a typical limiting axial heat flux profile [13] and a Westinghouse fuel thermal conductivity correlation [18] were input parameters employed in the program. The W-3 correlation [19] option was chosen for the critical heat flux calculations.

The results of the thermal analysis yielded a minimum departure from nucleate boiling ratio (MDNBR) of 3.26. Other results available from the thermal evaluation include; core pressure drop, axial and radial temperature profiles, and channel flow velocities. It should be noted that the thermal analysis is performed at steady-state conditions.

2.2 Design Assumptions

To accurately compare a metal-fueled reactor to the Westinghouse AP-600 reactor, a number of assumptions are made concerning parameters that should not be affected by the fuel design. These assumptions were first reported by Marsh [13]. The author agrees with these assumptions, with minor exceptions, and briefly reviews them below as a courtesy to the reader.

2.2.1 Steam Cycle

The secondary loop of any metal-fueled design will be the same as the AP-600 design. This requires the same reactor power. It also requires that the core inlet and outlet temperatures remain the same to provide the identical primary coolant temperatures in the steam generators.

2.2.2 Neutronics

Since oxygen and zirconium have relatively small cross-sections in PWRs compared to the uranium, neither element significantly contributes to the neutronic behavior of the reactor. Therefore, the H/U ratio for all designs will be the same as in the AP-600 core (4.802). Maintaining this value, also ensures a similar moderator temperature coefficient for both designs.

Another contributor to the neutronic behavior of the reactor is the H/D ratio. This study maintains the same H/D ratio (1.252) in the metal core as with AP-600.

A final assumption, necessary to preserve similar neutronics, involves the non-lattice fraction in the core. As noted in the two cases above, the non-lattice fractions are also fixed at the same value (0.0939).

2.2.3 Thermal Operating Margin

The Westinghouse AP-600 reactor is an advanced PWR design. For the metal-fueled reactor to also be considered an advanced design, it must exhibit the same thermal operating margin. To ensure the same safety margin with regard to the critical heat flux, the MDNBR for the two reactor cores is held the same.

2.2.4 Fuel Material

The fuel material is an alloy of uranium and zirconium. The alloying percentages vary from one (1) weight percent uranium to ninety-nine (99) weight percent uranium.

The fuel is coextruded to a zircaloy clad rod of typical thickness (0.0225 in.). The fuel is assumed to swell fifteen (15) percent (cross-sectional area) in the radial direction and the axial swelling of the fuel is neglected due to the coextrusion of the fuel with the cladding. This assumption is reasonable based upon recent (since 1985) experiments performed at EBR-II and FFTF. In these experiments, high fuel-smeared density (85 to 90%) analyses were performed on a range of fuel compositions at operating temperatures and conditions [20]. The results of these experiments indicate that a fifteen (15) percent cross-sectional void area in the fuel pin center is suffi-

cient to allow for swelling of the metal alloy. A schematic of the metal alloy fuel pin is shown in Figure 2.

2.3 Limiting Constraints

When a design task is undertaken, the limiting factors of the study must be considered. The investigation of metallic fuel alloys also has limiting constraints. The restrictions that apply to metallic fuels are fuel corrosion, geometrical core configurations, and fuel phase transitions.

2.3.1 Fuel Corrosion

The previous study of metal fuel pin optimization [13] used a fuel alloy of ten (10) weight percent zirconium and ninety (90) weight percent uranium. This choice was made based on the IFR fuel cycle assuming that there are no fuel clad failures in the PWR design. The IFR concept uses liquid sodium as a coolant and corrosion is not an issue. The PWR concept uses water as a coolant. Corrosion of a failed fuel pin by the heated water is a major concern with light water reactors.

After reviewing the literature, it was found that the fuel would be dissolved into the primary loop if a cladding breach occurred for alloys containing forty (40) weight

Figure 2: Schematic of a Metallic Fuel Pin

percent or less zirconium in 600°F water [21,22,23]. This is due to the corrosion rates of the metal fuels being approximately six orders of magnitude (10^6) greater than the corrosion rate of uranium dioxide (≈ 0.00135 mg/cm²/hr) [24]. Several fuel pin failures (not uncommon in PWR performance) would create a maintenance problem due to a large radionuclide inventory in the primary coolant. For low zirconium percentage alloys, it has been shown that corrosion rates are reduced by alloying a third compound with the binary system [25]. Although a reduction is seen, the corrosion rates are still higher than the fifty (50) weight percent uranium - fifty (50) weight percent zirconium alloy. Increasing the zirconium content in the fuel shows a marked increase in the corrosion resistance of the metal alloys [21,26]. Table 2 summarizes the corrosion rates for various metal alloys containing up to fifty (50) weight percent zirconium. Figure 3 through Figure 5 also show corrosion rates for various alloying contents and water conditions. [27]

Upon reviewing the data, only zirconium alloying percentages greater than fifty (50) weight percent can be considered as possible designs for a metal-fueled reactor due to the corrosion limitations. While this poses a serious barrier to allowable candidate alloys, the study must also consider other limiting constraints to the feasi-

Table 2: Corrosion Rates of Metallic Fuel Alloys

Tested Fuel	Temperature (Deg.F)	Sample Shape & Condition	Corrosion Rate (mg/cm**2-hr)
Pure U	587	Pin/Unciad	6260
U - 5 wt% Zr	587	Pin/Unciad	3150
U - 10 wt% Zr	595	Pin/Unciad	2780
U - 15 wt% Zr	604	Pin/Unbonded	165
U - 20 wt% Zr	610	Pin/Unbonded	194
U - 50 wt% Zr	600	Unciad	0.025-0.1
U oxide	650	Pin/Clad	0.00135

Figure 3: double-arc-welded U-Zr Alloys Exposed to Water at 600°F

Figure 4: Corrosion Rates Versus Water Temperature for U-Zr Alloys

Figure 5: Corrosion Rates Versus Uranium Content for U-Zr Alloys

bility of a metallic fuel core design.

2.3.2 Geometry

The hydrogen to uranium (H/U) ratio is the relation of hydrogen atoms to uranium atoms in a typical fuel cell in the core of the reactor. This ratio helps determine the neutronic behavior of the reactor, and establishes a moderator temperature coefficient. As was discussed earlier, the H/U ratio for this study is fixed at the AP-600 value of 4.802.

Figure 6 - part A shows a typical square lattice fuel cell and an equivalent fuel cell. The hatched area represents the water that flows through the channel.

Since the H/U ratio is fixed, it is possible for the pin diameter to exceed the pitch of the lattice at low uranium alloy concentrations. This is graphically shown in part B of Figure 6. Although this is mathematically possible, it is not feasible for the fuel pins to overlap in the fuel cell. The equation for the pitch of the metal-fueled core is given by:

$$P = [((HU * FUELA * DU * MH) / (2 * DH * MU)) + ((\pi * PIND^2) / 4)]^{1/2} \quad /1/$$

where,

P = Pitch
HU = Hydrogen to uranium ratio
DU = Density of uranium
MH = Molecular weight of water

MU = Molecular weight of uranium
DH = Density of water
PIND = Pin diameter
FUELA = Area of fuel in pin

When calculating the pitch for the higher zirconium content alloys, the pin diameters become larger than the pitch at a constant H/U ratio. Due to safety concerns, the fuel pins must maintain some slight distance between them so that the structure can be mechanically supported and not create "hot-spots" on the fuel pins.

Therefore, a lower limit on the amount of the uranium alloy content can be determined. Solving equation /1/, the alloying content at which the fuel pins touched was calculated to be twenty-two (22) weight percent uranium. To be consistent with the limits on corrosion, the geometrical restriction is given in terms of the zirconium weight percentage. Due to the aforementioned safety concerns, instead of the constraining limit being seventy-eight (78) weight percent zirconium, the limit was set at seventy-five (75) weight percent. This represents a conservative limit due to the margin of safety maintained. Although seventy-six (76) and seventy-seven (77) weight percents are feasible designs, extra space between fuel pins is preserved at seventy-five (75) weight percent zirconium.

Figure 6: Feasible and Unfeasible Fuel Cell Configurations

2.3.3 Fuel Phase Transition

Due to the thermodynamics of the binary uranium - zirconium structure, there exists a phase transition for the alloy at approximately 1157°F (610°C). [28] The phase transition of the alloy occurs at a slightly lower temperature for higher zirconium weight percentages. This phase change is at approximately 1103°F (595°C). [28]

An evaluation was performed at Westinghouse Atomic Power Division on a range a uranium - zirconium contents (40 weight percent to 90 weight percent). From the study, the range of alloys had dimensional stability acceptable for engineering purpose if the previously mentioned phase transition temperatures were not exceeded. [29]

A thermal analysis of the fuel must be performed before the temperature distribution in the fuel pins can be evaluated. Therefore, it is only possible to eliminate alloying systems by the previous two mechanisms until the thermal considerations can be taken into account.

2.4 Method for Metallic Core Analysis

At this time, the procedures of this study will be reviewed. A reference design for a metal-fueled reactor was established. The design chosen was the advanced Westinghouse AP-600 light water reactor. Assumptions were

given so that the study was comparable. These assumptions included the steam cycle, neutronics, thermal operating margin, and fuel material. Limiting constraints to the design were given for corrosion processes and geometrical restrictions. Therefore, only binary systems between fifty (50) weight percent zirconium and seventy-five (75) weight percent zirconium can be considered as feasible designs. Another limiting factor to the fuel composition in feasible designs is the phase transition temperature of the fuel alloys. This will be considered after a thermal analysis of each binary system can be performed. The results of this study are presented in the following chapter. The method of the fuel composition analysis is given below.

2.4.1 Design Analysis Techniques

The window of feasible alloying contents is now set for the study. To proceed with the investigation, each alloying content (at 5 weight percent increments) will be optimized with regard to cost. But, before the cost optimization can be performed, it is necessary to establish the dimensions and parameters of the metallic core.

For each alloying content, an arbitrary pin diameter is chosen. Utilizing the hypotheses of section 2.2 (Design Assumptions), it is possible to calculate the dimensions of the core. It is also plausible to calculate parameters

necessary to perform a thermal analysis. The process of calculating these dimensions and parameters was automated with a Fortran program, DICE4.

DICE4 calculates the DNBR at the top of the core with the W-3 critical heat flux correlation. The number of assemblies (number of pins) is iterated upon until the DICE4 DNBR matches a target DNBR, "DNBRt". Once the calculated DNBR matches DNBRt, DICE4 computes and outputs design and thermal parameters as the necessary input "cards" for COBRA-IV-PC, a thermal-hydraulics code. Since DICE4 does a crude thermal analysis, it is necessary to perform the thermal analysis on the metallic core with COBRA-IV-PC. A descriptive listing of the DICE4 program is given in Appendix A.

The thermal analysis of the metallic core is similar to that for AP-600. The same limiting axial heat flux profile and radial peaking factor are used for the thermal evaluation.

A fuel thermal conductivity correlation for the range of alloying contents under consideration could not be found. Therefore, a temperature independent fuel thermal conductivity was chosen for each alloy. [28] The value was chosen based on the assumption that the average operating fuel temperature was 450°C. This value was then reduced by fifteen (15) percent to account for the homogenization of

the fuel into the central void. The values for the fuel thermal conductivity used in this study are shown in Table 3. These values are conservative because the thermal conductivity of the metallic fuel increases with increasing operating temperature. [28] It will be shown in Section 3.1.5 (Fuel Temperatures) that the actual operating temperatures are higher than originally predicted. It is then necessary to run COBRA-IV-PC to obtain a more accurate DNBR for the metallic core. The value for the DNBR found by COBRA-IV-PC is denoted by "DNBRc".

Once a value for the "DNBRc" is found for the metallic core, it is compared to the DNBR value, "DNBRr", found for AP-600. If "DNBRc" does not match "DNBRr", the target departure from nucleate boiling ratio, "DNBRt" is modified and DICE4 is re-run. Changing "DNBRt", modifies the number of pins and rod length in DICE4 which, in turn, adjusts the DNBR values obtained from COBRA-IV-PC. This process is repeated until the DNBR value for the metallic core matches the DNBR value for AP-600. Typical input and output files for COBRA-IV-PC are given in Appendix B.

The same critical heat flux option for AP-600 is used for the metal-fueled cores in COBRA-IV-PC to match the DNBR. It will be shown in Section 3.1.1 (Designs) that the cores for the metallic fuel alloys will extend beyond the range (12"-144") of the W-3 correlation. It was, there-

Table 3: Thermal Conductivity as a Function of Alloying Percentage @ 450 Degrees Celcius

Fuel Alloying Percentage	Fuel Thermal Conductivity [28] (Btu/hr-ft-deg.F)	Adjusted Fuel Thermal Conductivity (Btu/hr-ft-deg.F)
U - 50 wt% Zr	10.90	9.27
U - 55 wt% Zr	10.40	8.84
U - 60 wt% Zr	10.00	8.5
U - 65 wt% Zr	10.25	8.71
U - 70 wt% Zr	10.50	8.93
U - 75 wt% Zr	10.90	9.27

fore, necessary to check the validity of the critical heat flux correlation with an alternative method. The alternative method chosen was the table look-up technique. [29] The tables for this method are statistical averages of experimental data for critical heat flux. The values tabulated are presented for a wide variety of conditions (pressure, mass flow, and local quality). This method is more general than empirical correlations, which have a limited range of application. Although the tables did not exactly predict the same DNBR as the W-3 correlation, the relative DNBR values for the metallic cores were within a five (5) percent window surrounding the value for AP-600. Since a correlation for longer, metal-fueled cores does not exist, it is assumed that this small variance in the DNBR is negligible to the study.

Once the DNBRs are matched, a new pin diameter is chosen and the process is repeated until a range of possible designs are formed. It is now necessary to perform the cost optimization for these arrangements to find the "best" design.

2.4.2 Cost Assumptions for Optimization

Cost assumptions were initially reported in a previous study for metallic fuel pin optimization. [13] Table 4 illustrates these cost parameters. The shaded regions in

the table show were differences between the two studies exist. Although there are several shaded parameters, there were essentially only three exceptions. Two of the parameters, core mass flow rate and reactor power, were changed based on the latest available data (see Table 1) for AP-600. The zirconium weight percentage obviously changes with each alloy considered.

In the original study, the enrichment technique for the metallic fuel was based on the uranium oxide fuel cycle, the gaseous diffusion process. The cost of enrichment was given as \$109/SWU. As reported in section 1.3.2 (Enrichment and Fabrication), the levelized SWU cost for the AVLIS process is below \$40/SWU. Therefore, \$40/SWU is chosen as the enrichment cost for the analysis. In addition to the \$40/SWU enrichment cost, it is assumed that the cost to re-alloy the AVLIS product will be the same as the cost to convert the product to an oxide, \$15/SWU. Therefore, both AP-600 and the metal-fueled cores will have a total enrichment cost of \$55/SWU.

Also, the original research considered changes in the vessel size were linearly proportional to changes in the containment size. This study assumes that the ratio of containment volume to the volume of water in the vessel is the limiting constraint. Therefore, for varying vessel sizes, the ratio of water in the vessel to the water in a

reference vessel [30] is multiplied by the containment size of the reference to obtain the metallic core containment size. This is shown mathematically as the ratio below:

$$\frac{\text{Ref. Vessel Water Vol.}}{\text{Dgn. Vessel Water Vol.}} = \frac{\text{Ref. Containment Size}}{\text{Dgn. Containment Size}} \quad /2/$$

The cycle length was also changed from eighteen (18) months to twenty-four (24) months. This change was made at the author's discretion for reasons that will become apparent in Chapter IV (Sensitivity of Results).

Due to uncertainties in the fabrication costs for the metallic fuel pins, Marsh presented techniques that found the minimum and maximum costs, along with a range of median values for fuel fabrication. [13] The maximum fabrication cost was the \$340/fuel pin value of AP-600. The minimum fabrication cost was just the cost of the zirconium in the core. The median values were simply the minimum value plus \$50/fuel pin increments up to \$200/rod. These uncertainties still exist at the time of this study, but the minimum and maximum fabrication costs are considered unrealistic by the author. The fabrication cost chosen, is the cost for the zirconium in the core plus \$150 per fuel rod.

Table 4: Spreadsheet Cost Optimization Parameters

Constants:	Values:	Units / Comments:
Interest rate	0.0435	(assumes no inflation Page 19 of DOE/NE-0195)
Analysis period	30	yr
$P/A_{i,n}$	16.58028976141	(discrete)
Vessel cost	9213	basis is active core vol, 1987 numbers not escalated or moved to startup
Core mass flow rate	20444.44	lb/sec
Pump capital cost/lp	150.7	\$/lp
Water density at 540	45.9	lb/ft ³
Electric cost for pumps	0.04	\$/kwhr
Fab/pin	Min. + \$150/prod	\$/pin basis \$220/kg fab
Clad thickness	0.0225	inches
Zr weight fraction	0.5	
Zr cost/lb	45	\$/lb
Reactor power	1933	MW thermal
Fraction of power in pins	0.974	
Cycle length	24	months
Capacity factor	0.75	
Tails enrichment	0.002	in weight fraction
Value function for tails	6.19	
Value function for feed	4.87	
Cost of SWU	55	\$/kgSWU
Fabrication lead time	1	yr
Enrichment lead time	2	yr
Ore lead time	3	yr
Fab time value adj	1.0435	
Enrich time value adj	1.08889225	
Ore time value adj	1.136259062875	
U3O8 cost	25	\$/lb
Discharge burnup	45000	MWD/MTU
Refueling periods	14	
Int. over refueling period	0.08889225	
Present worth refueling	7.834915108371	
Containment cost	25.285	\$/ft ³

* Shaded regions represent values that differ from Marsh [13]

2.4.3 Discussion of Method

With the fuel pin design parameters and the cost assumptions complete, the optimization can be performed. The above parameters are input into a spreadsheet for easy calculation and a cost is established for a pin diameter at a given alloying percentage. Appendix C shows a descriptive listing and the formulae used for calculating the total costs of AP-600 and a metal-fueled design.

After several pin diameters have been examined, a minimum cost can be determined for that alloying content. The entire process presented in these last three sections is then repeated for every alloying content in the window of feasible alloys.

The process of finding the design parameters and the total costs for each pin diameter of an alloying percentage is shown in Figure 7. The figure gives a flow chart for the methods outlined above. This includes the DICE4 program, the COBRA-IV-PC analysis, and the spreadsheet steps for finding total costs.

Figure 7: Pin Diameter Optimization Flow Chart

CHAPTER III

OPTIMIZED DESIGN PARAMETERS

3.1 Results of Optimization

The outcomes of the optimization process for each alloy are presented in the next five sections. The first section shows the designs of all the alloying percentages. The second section gives the costs for the aforementioned designs. This is shown graphically and in a tabular framework. The third section compares and discusses the costs for the optimized designs. In the fourth section, the design parameters resulting from the optimization process are compared. The fifth section shows and discusses the temperature profiles as predicted in the thermal analyses.

3.1.1 Designs

This part of the research presents the design parameters for each alloying content. Tables for each alloy are given on the following pages. These tables show pertinent design parameters of each pin outer diameter for every alloy considered. The designs highlighted with double lines are the ones that show the lowest total costs.

Table 5: Design Parameters of Various Pin Diameters for 50 wt. % U - 50 wt. % Zr

	AP-600				Min. for 50 U - 50 Zr		
Pin dia (mm)	9.5	15.2	16.5	17.8	19.1	20.3	21.6
Pitch (mm)	12.6	17.4	18.9	20.4	21.9	23.4	24.9
Length (m)	3.66	3.48	3.77	4.03	4.29	4.53	4.76
Number of fuel pins	38280	18124	18000	17731	17398	17010	16598
Core Volume (l)	24532	21107	26759	32884	39531	46606	54116
Containment Volume (l)	5.066E+07	4.298E+07	4.644E+07	5.021E+07	5.431E+07	5.870E+07	6.337E+07
MTU Core Total	67.71	41.94	53.57	66.25	80.07	94.85	110.59
Initial Enrichment (%)	2.7	3.7	3.1	2.7	2.4	2.3	2.1
Power W/(gram-U)	27.81	44.89	35.15	28.42	23.51	19.85	17.02
Linear Power (kW/m)	13.4	29.9	27.8	26.3	25.2	24.4	23.8
Cycle Burnup (MWD/MTU)	15621	25221	19746	15966	13209	11152	9564
No. Batches Per Core	2.881	1.784	2.279	2.818	3.407	4.035	4.705
Pressure Drop (kpa)	90.61	178.70	137.25	111.19	93.99	82.28	74.27
Pressure Drop (kPa/m)	24.77	51.36	36.44	27.57	21.91	18.16	15.60
No. of Rods Refueled	13288	10158	7898	6291	5107	4215	3528
Refuel Enrichment (%)	4.4	4.9	4.6	4.4	4.2	4.1	4.0
Peak Fuel Temp. (deg.F)	2920	1467	1401	1355	1320	1294	1275

Table 6: Design Parameters of Various Pin Diameters for 45 wt. % U - 55 wt. % Zr

	AP-600				Min. for 45 U - 55 Zr		
Pin dia (mm)	9.5	17.8	19.1	20.3	21.6	22.9	24.1
Pitch (mm)	12.6	19.8	21.3	22.7	24.2	25.6	27.1
Length (m)	3.66	3.62	3.87	4.11	4.34	4.56	4.77
Number of fuel pins	38280	15144	15038	14854	14634	14352	14064
Core Volume (l)	24532	23765	29072	34788	40967	47394	54241
Containment Volume (l)	5.066E+07	4.349E+07	4.647E+07	4.970E+07	5.321E+07	5.686E+07	6.077E+07
MTU Core Total	67.71	43.60	53.64	64.51	76.31	88.62	101.78
Initial Enrichment (%)	2.7	3.6	3.1	2.7	2.5	2.3	2.2
Power W/(gram-U)	27.81	43.18	35.10	29.18	24.67	21.24	18.50
Linear Power (kW/m)	13.4	34.3	32.3	30.8	29.6	28.8	28.1
Cycle Burnup (MWD/MTU)	15621	24260	19718	16395	13861	11935	10393
No. Batches Per Core	2.081	1.855	2.202	2.745	3.246	3.770	4.330
Pressure Drop (kPa)	90.61	180.10	143.91	119.61	102.62	90.72	81.87
Pressure Drop (kPa/m)	24.77	49.75	37.17	29.10	23.64	19.91	17.18
No. of Nodes Refueled	13208	8164	6589	5412	4508	3807	3248
Refuel Enrichment (%)	4.4	4.8	4.6	4.4	4.3	4.2	4.1
Peak Fuel Temp. (deg.F)	2920	1600	1536	1489	1452	1424	1402

Table 7: Design Parameters of Various Pin Diameters for 40 wt. % U - 60 wt. % Zr

	AP-600			Min for 40 U - 60 Zr			
Pin dia (mm)	9.5	22.9	24.1	25.4	26.7	27.9	29.2
Pitch (mm)	12.6	24.9	26.3	27.7	29.1	30.5	31.9
Length (m)	3.66	4.08	4.29	4.49	4.68	4.87	5.04
Number of fuel pins	38280	12240	12098	11939	11755	11554	11343
Core Volume (l)	24532	34145	39569	45363	51416	57729	64286
Containment Volume (l)	5.066E+07	4.757E+07	5.035E+07	5.332E+07	5.644E+07	5.969E+07	6.309E+07
MTU Core Total	67.71	57.33	66.68	76.70	87.19	98.16	109.57
Initial Enrichment (%)	2.7	2.9	2.7	2.5	2.4	2.2	2.2
Power W/(gram-U)	27.81	32.84	28.23	24.55	21.59	19.18	17.18
Linear Power (kW/m)	13.4	37.7	36.3	35.1	34.2	33.5	32.9
Cycle Burnup (MWD/MTU)	15621	18450	15862	13791	12132	10776	9653
No. Batches Per Core	2.881	2.439	2.837	3.263	3.709	4.176	4.662
Pressure Drop (kPa)	90.61	142.53	122.75	108.07	97.03	88.70	82.27
Pressure Drop (kPa/m)	24.77	34.89	28.61	24.07	20.73	18.23	16.31
No. of Rods Refueled	13288	5018	4264	3659	3169	2767	2433
Refuel Enrichment (%)	4.4	4.5	4.4	4.3	4.2	4.1	4.0
Peak Fuel Temp. (deg.F)	2920	1691	1647	1612	1583	1560	1542

Table 8: Design Parameters of Various Pin Diameters for 35 wt. % U - 65 wt. % Zr

	AP-600				Min. for 35 U - 65 Zr		
Pin dia (mm)	9.5	25.4	26.7	27.9	29.2	30.5	31.8
Pitch (mm)	12.6	26.9	28.3	29.7	31.0	32.4	33.8
Length (m)	3.66	3.98	4.17	4.35	4.53	4.70	4.87
Number of fuel pins	38280	9923	9862	9786	9688	9572	9445
Core Volume (l)	24532	31621	36344	41378	46653	52137	57846
Containment Volume (l)	5.066E+07	4.466E+07	4.681E+07	4.910E+07	5.154E+07	5.406E+07	5.669E+07
MTU Core Total	67.71	47.51	54.78	62.54	70.69	79.19	88.05
Initial Enrichment (%)	2.7	3.3	3.0	2.8	2.6	2.5	2.3
Power W/(gram-U)	27.81	39.63	34.37	30.11	26.63	23.77	21.38
Linear Power (kW/m)	13.4	47.7	45.8	44.2	42.9	41.8	40.9
Cycle Burnup (MWD/MTU)	15621	22264	19311	16913	14962	13357	12013
No. Batches Per Core	2.881	2.021	2.330	2.661	3.008	3.369	3.746
Pressure Drop (kpa)	90.61	185.60	159.41	139.24	123.64	111.84	102.31
Pressure Drop (kPa/m)	24.77	46.62	38.22	31.97	27.28	23.78	21.01
No. of Rods Refueled	13288	4909	4232	3678	3221	2841	2521
Refuel Enrichment (%)	4.4	4.7	4.6	4.4	4.3	4.2	4.2
Peak Fuel Temp. (deg.F)	2920	1919	1863	1817	1778	1747	1721

Table 9: Design Parameters of Various Pin Diameters for 30 wt. % U - 70 wt. % Zr

	AP-600				Min. for 30 U - 70 Zr		
Pin dia (mm)	9.5	31.8	33.0	34.3	35.6	36.8	38.1
Pitch (mm)	12.6	32.8	34.2	35.5	36.8	38.1	39.5
Length (m)	3.66	4.27	4.43	4.58	4.74	4.88	5.03
Number of fuel pins	38280	7666	7634	7579	7515	7444	7373
Core Volume (l)	24532	38898	43534	48274	53217	58339	63714
Containment Volume (l)	5.066E+07	4.582E+07	4.769E+07	4.958E+07	5.156E+07	5.363E+07	5.581E+07
MTU Core Total	67.71	51.47	57.72	64.13	70.83	77.78	85.08
Initial Enrichment (%)	2.7	3.2	2.9	2.8	2.6	2.5	2.4
Power W/(gram-U)	27.81	36.58	32.62	29.36	26.58	24.21	22.13
Linear Power (kW/m)	13.4	57.6	55.7	54.2	52.9	51.8	50.8
Cycle Burnup (MWD/MTU)	15621	20551	18324	16492	14934	13599	12433
No. Batches Per Core	2.881	2.190	2.456	2.729	3.013	3.309	3.619
Pressure Drop (kpa)	90.61	184.47	163.44	146.93	133.51	122.55	113.41
Pressure Drop (kPa/m)	24.77	43.24	36.90	32.05	28.19	25.10	22.55
No. of Rods Refueled	13288	3501	3109	2778	2494	2250	2037
Refuel Enrichment (%)	4.4	4.6	4.5	4.4	4.3	4.3	4.2
Peak Fuel Temp. (deg.F)	2920	2115	2063	2022	1986	1955	1927

Table 10: Design Parameters of Various Pin Diameters for 25 wt. % U - 75 wt. % Zr

	AP-600				Min. for 25 U - 75 Zr		
Pin dia (mm)	9.5	39.4	40.6	41.9	43.2	44.5	45.7
Pitch (mm)	12.6	39.7	41.0	42.3	43.6	44.8	46.1
Length (m)	3.66	4.45	4.59	4.72	4.86	4.98	5.11
Number of fuel pins	38280	5710	5699	5670	5644	5607	5570
Core Volume (l)	24532	44196	48508	52825	57403	62038	66876
Containment Volume (l)	5.066E+07	4.542E+07	4.689E+07	4.837E+07	4.995E+07	5.154E+07	5.321E+07
MTU Core Total	67.71	50.10	55.08	60.05	65.33	70.69	76.29
Initial Enrichment (%)	2.7	3.2	3.0	2.9	2.7	2.6	2.5
Power W/(gram-U)	27.81	37.58	34.19	31.38	28.82	26.63	24.68
Linear Power (kW/m)	13.4	74.1	71.9	70.3	68.7	67.4	66.1
Cycle Burnup (MWD/MTU)	15621	21114	19209	17616	16190	14963	13864
No. Batches Per Core	2.881	2.131	2.343	2.555	2.779	3.007	3.246
Pressure Drop (kpa)	90.61	209.11	188.83	172.41	158.31	146.69	136.58
Pressure Drop (kPa/m)	24.77	46.98	41.12	36.50	32.59	29.43	26.72
No. of Rods Refueled	13288	2679	2433	2220	2030	1864	1716
Refuel Enrichment (%)	4.4	4.7	4.6	4.5	4.4	4.3	4.3
Peak Fuel Temp. (deg.F)	2920	2437	2382	2339	2297	2263	2231

3.1.2 Costs

Given below are figures showing the total costs for each zirconium percent alloy as a function of pin outer diameter. This is the basis for the optimization process. In addition to each figure, a table for each binary system is also presented. Each table gives a comprehensive listing of the total costs. It should be noted that total costs are not total plant costs. The expenses shown are only the ones associated with changing design parameters due to the use of varying fuel alloys.

Figure 8: Total Costs Versus Pin Outer Diameter for 50 wt. % U - 50 wt. % Zr

Table 11: Comprehensive Cost Listing of Various Pin Diameters for 50 wt. % U - 50 wt. % Zr

	AP-600				Min. for 50 U - 50 Zr		
Pin dia (mm)	9.5	15.2	16.5	17.8	19.1	20.3	21.6
Miscellaneous Costs:							
Containment & Vessel (\$)	5.32E+07	4.52E+07	5.02E+07	5.55E+07	6.14E+07	6.76E+07	7.42E+07
Pump Capital (\$)	2.31E+05	4.55E+05	3.50E+05	2.83E+05	2.40E+05	2.10E+05	1.89E+05
Pump Operating (\$)	6.64E+06	1.31E+07	1.01E+07	8.15E+06	6.89E+06	6.03E+06	5.44E+06
Initial Core Costs:							
Fab. Costs (\$)	1.36E+07	6.14E+06	6.91E+06	7.70E+06	8.55E+06	9.43E+06	1.03E+07
Enrichment (\$)	1.46E+07	1.47E+07	1.44E+07	1.45E+07	1.50E+07	1.56E+07	1.65E+07
Ore (\$)	2.41E+07	2.11E+07	2.23E+07	2.39E+07	2.60E+07	2.84E+07	3.10E+07
Refueling Costs (Levelized):							
Fab. Costs (\$)	4.71E+06	3.44E+06	3.03E+06	2.73E+06	2.51E+06	2.34E+06	2.20E+06
Enrichment (\$)	8.15E+07	9.51E+07	8.73E+07	8.19E+07	7.81E+07	7.52E+07	7.30E+07
Ore (\$)	1.11E+08	1.25E+08	1.17E+08	1.11E+08	1.07E+08	1.04E+08	1.02E+08
Total Costs (\$)	3.42E+08	3.48E+08	3.32E+08	3.25E+08	3.23E+08	3.25E+08	3.30E+08

Figure 9: Total Costs Versus Pin Outer Diameter for 45 wt. % U - 55 wt. % Zr

Table 12: Comprehensive Cost Listing of Various Pin Diameters for 45 wt. % U - 55 wt. % Zr

	AP-600				Min. for 45 U - 55 Zr		
Pin dia (mm)	9.5	17.8	19.1	20.3	21.6	22.9	24.1
Miscellaneous Costs:							
Containment & Vessel (\$)	5.32E+07	4.66E+07	5.10E+07	5.57E+07	6.08E+07	6.62E+07	7.19E+07
Pump Capital (\$)	2.31E+05	4.59E+05	3.67E+05	3.05E+05	2.62E+05	2.31E+05	2.09E+05
Pump Operating (\$)	6.64E+06	1.32E+07	1.05E+07	8.77E+06	7.52E+06	6.65E+06	6.00E+06
Initial Core Costs:							
Fab. Costs (\$)	1.36E+07	6.00E+06	6.72E+06	7.47E+06	8.26E+06	9.07E+06	9.92E+06
Enrichment (\$)	1.46E+07	1.46E+07	1.44E+07	1.45E+07	1.48E+07	1.53E+07	1.60E+07
Ore (\$)	2.41E+07	2.12E+07	2.23E+07	2.37E+07	2.54E+07	2.74E+07	2.95E+07
Fueling Costs (Levelized):							
Fab. Costs (\$)	4.71E+06	3.24E+06	2.94E+06	2.72E+06	2.54E+06	2.40E+06	2.29E+06
Enrichment (\$)	8.15E+07	9.37E+07	8.73E+07	8.26E+07	7.90E+07	7.63E+07	7.41E+07
Ore (\$)	1.11E+08	1.24E+08	1.17E+08	1.12E+08	1.08E+08	1.05E+08	1.03E+08
Total Costs (\$)	3.42E+08	3.45E+08	3.32E+08	3.26E+08	3.24E+08	3.25E+08	3.29E+08

Figure 10: Total Costs Versus Pin Outer Diameter for 40 wt. % U - 60 wt. % Zr

Table 13: Comprehensive Cost Listing of Various Pin Diameters for 40 wt. % U - 60 wt. % Zr

	AP-600			Min. for 40 U - 60 Zr			
Pin dia (mm)	9.5	22.9	24.1	25.4	26.7	27.9	29.2
Miscellaneous Costs:							
Containment & Vessel (\$)	5.32E+07	5.36E+07	5.78E+07	6.24E+07	6.71E+07	7.21E+07	7.73E+07
Pump Capital (\$)	2.31E+05	3.63E+05	3.13E+05	2.75E+05	2.47E+05	2.26E+05	2.10E+05
Pump Operating (\$)	6.64E+06	1.04E+07	9.00E+06	7.92E+06	7.11E+06	6.50E+06	6.03E+06
Initial Core Costs:							
Fab. Costs (\$)	1.36E+07	6.83E+06	7.52E+06	8.25E+06	9.00E+06	9.78E+06	1.06E+07
Enrichment (\$)	1.46E+07	1.44E+07	1.45E+07	1.48E+07	1.53E+07	1.58E+07	1.64E+07
Ore (\$)	2.41E+07	2.27E+07	2.40E+07	2.55E+07	2.71E+07	2.89E+07	3.09E+07
Refueling Costs (Levelized):							
Fab. Costs (\$)	4.71E+06	2.80E+06	2.65E+06	2.53E+06	2.43E+06	2.34E+06	2.27E+06
Enrichment (\$)	8.15E+07	8.55E+07	8.18E+07	7.89E+07	7.66E+07	7.47E+07	7.31E+07
Ore (\$)	1.11E+08	1.15E+08	1.11E+08	1.08E+08	1.06E+08	1.04E+08	1.02E+08
Total Costs (\$)	3.42E+08	3.31E+08	3.27E+08	3.26E+08	3.27E+08	3.30E+08	3.34E+08

Figure 11: Total Costs Versus Pin Outer Diameter for 35 wt. % U - 65 wt. % Zr

Table 14: Comprehensive Cost Listing of Various Pin Diameters for 35 wt. % U - 65 wt. % Zr

	AP-600				Min. for 35 U - 65 Zr		
Pin dia (mm)	9.5	25.4	26.7	27.9	29.2	30.5	31.8
Miscellaneous Costs:							
Containment & Vessel (\$)	5.32E+07	5.02E+07	5.36E+07	5.73E+07	6.12E+07	6.52E+07	6.94E+07
Pump Capital (\$)	2.31E+05	4.73E+05	4.06E+05	3.55E+05	3.15E+05	2.85E+05	2.61E+05
Pump Operating (\$)	6.64E+06	1.36E+07	1.17E+07	1.02E+07	9.06E+06	8.20E+06	7.50E+06
Initial Core Costs:							
Fab. Median (\$)	1.36E+07	5.94E+06	6.54E+06	7.16E+06	7.80E+06	8.47E+06	9.15E+06
Enrichment (\$)	1.46E+07	1.44E+07	1.44E+07	1.44E+07	1.46E+07	1.49E+07	1.53E+07
Ore (\$)	2.41E+07	2.16E+07	2.24E+07	2.34E+07	2.46E+07	2.59E+07	2.73E+07
Refueling Costs (Levelized):							
Fab. Median (\$)	4.71E+06	2.94E+06	2.80E+06	2.69E+06	2.59E+06	2.51E+06	2.44E+06
Enrichment (\$)	8.15E+07	9.09E+07	8.67E+07	8.33E+07	8.05E+07	7.83E+07	7.64E+07
Ore (\$)	1.11E+08	1.21E+08	1.16E+08	1.13E+08	1.10E+08	1.07E+08	1.05E+08
Total Costs:	3.42E+08	3.41E+08	3.34E+08	3.30E+08	3.28E+08	3.28E+08	3.30E+08

Figure 12: Total Costs Versus Pin Outer Diameter for 30 wt. % U - 70 wt. % Zr

Table 15: Comprehensive Cost Listing of Various Pin Diameters for 30 wt. % U - 70 wt. % Zr

	AP-600				Min. for 30 U - 70 Zr		
Pin dia (mm)	9.5	31.8	33.0	34.3	35.6	36.8	38.1
Miscellaneous Costs:							
Containment & Vessel (\$)	5.32E+07	5.36E+07	5.67E+07	6.00E+07	6.34E+07	6.69E+07	7.06E+07
Pump Capital (\$)	2.31E+05	4.70E+05	4.17E+05	3.75E+05	3.40E+05	3.12E+05	2.89E+05
Pump Operating (\$)	6.64E+06	1.35E+07	1.20E+07	1.08E+07	9.79E+06	8.98E+06	8.31E+06
Initial Core Costs:							
Fab. Costs (\$)	1.36E+07	6.17E+06	6.71E+06	7.26E+06	7.82E+06	8.41E+06	9.01E+06
Enrichment (\$)	1.46E+07	1.44E+07	1.44E+07	1.45E+07	1.46E+07	1.49E+07	1.52E+07
Ore (\$)	2.41E+07	2.20E+07	2.28E+07	2.36E+07	2.46E+07	2.56E+07	2.68E+07
Refueling Costs (Levelized):							
Fab. Costs (\$)	4.71E+06	2.02E+06	2.73E+06	2.66E+06	2.60E+06	2.54E+06	2.49E+06
Enrichment (\$)	8.15E+07	8.84E+07	8.53E+07	8.27E+07	8.05E+07	7.86E+07	7.70E+07
Ore (\$)	1.11E+08	1.18E+08	1.15E+08	1.12E+08	1.10E+08	1.08E+08	1.06E+08
Total Costs (\$)	3.42E+08	3.39E+08	3.35E+08	3.32E+08	3.31E+08	3.31E+08	3.33E+08

Figure 13: Total Costs Versus Pin Outer Diameter for 25 wt. % U - 75 wt. % Zr

Table 16: Comprehensive Cost Listing of Various Pin Diameters for 25 wt. % U - 75 wt. % Zr

	AP-600				Min. for 25 U - 75 Zr		
Pin dia (mm)	9.5	39.4	40.6	41.9	43.2	44.5	45.7
Miscellaneous Costs:							
Containment & Vessel (\$)	5.32E+07	5.49E+07	5.77E+07	6.04E+07	6.33E+07	6.62E+07	6.93E+07
Pump Capital (\$)	2.31E+05	5.33E+05	4.81E+05	4.39E+05	4.04E+05	3.74E+05	3.48E+05
Pump Operating (\$)	6.64E+06	1.53E+07	1.38E+07	1.26E+07	1.16E+07	1.08E+07	1.00E+07
Initial Core Costs:							
Fab. Costs (\$)	1.36E+07	5.98E+08	6.44E+06	6.90E+06	7.38E+06	7.86E+06	8.36E+06
Enrichment (\$)	1.46E+07	1.44E+07	1.44E+07	1.44E+07	1.45E+07	1.46E+07	1.48E+07
Ore (\$)	2.41E+07	2.19E+07	2.24E+07	2.31E+07	2.38E+07	2.46E+07	2.54E+07
Refueling Costs (Levelized):							
Fab. Costs (\$)	4.71E+06	2.81E+06	2.75E+06	2.70E+06	2.65E+06	2.61E+06	2.58E+06
Enrichment (\$)	8.15E+07	8.92E+07	8.65E+07	8.43E+07	8.23E+07	8.05E+07	7.90E+07
Ore (\$)	1.11E+08	1.19E+08	1.16E+08	1.14E+08	1.12E+08	1.10E+08	1.08E+08
Total Costs (\$)	3.42E+08	3.43E+08	3.39E+08	3.37E+08	3.36E+08	3.35E+08	3.36E+08

3.1.3 Cost Comparison

In order to compare the designs of the varying alloys, all the optimized designs will be discussed in relation to each other. Figure 14 shows how the total cost of a design varies as a function of weight percent zirconium. Also given below, in Table 17, are a complete listing of the costs shown in Figure 14.

From the data presented in Figure 14, the most cost efficient alloying percentage is the fifty (50) weight percent uranium - fifty (50) weight percent zirconium alloy. The general trend of the costs are to increase with increasing zirconium content. Although the costs increase the total costs for the largest zirconium alloying content considered is below the total costs for AP-600.

From Table 17, more detail can be inferred about the trends of the costs. The initial core costs decrease, but the variations are small. The pump costs increase with increasing zirconium content, and contribute approximately ten (10) to thirty (30) percent towards the general increasing trend. The largest contributor to the increasing costs emanates from the levelized refueling costs. Although the refueling enrichment percentage increases only slightly, the ore and enrichment costs for refueling account for approximately seventy (70) percent of the difference between the total cost for the seventy-five (75)

weight percent zirconium design and the total cost for the fifty (50) weight percent zirconium design.

It will be shown in Section 3.1.4 (Design Comparison), that the fifty (50) weight percent zirconium design has the highest number of batches of all optimized designs. This design, therefore, requires the lowest initial and refueling enrichments. This is illustrated by the batch equation used in the spreadsheet optimization [31];

$$\text{Enrichment} = ((\text{Discharge Burnup}/A) * (B+1)/(2*B)) + C \quad /3/$$

where,

A = 9000

B = Number of batches

C = 1.0 for refueling or 1.5 for initial core

The values for A and C are constants for a typical PWR. For lower enrichments, the feed to product ratio is less, and therefore, the amount of ore needed is less. This all drives the costs to a minimum for the fifty (50) weight percent design.

Figure 14: Total Costs Versus Weight Percent Zirconium

Table 17: Comprehensive Cost Listing for Optimized Alloying Contents

	AP-600	Min. for 50 U - 30 Zr	Min. for 45 U - 55 Zr	Min. for 40 U - 60 Zr	Min. for 35 U - 65 Zr	Min. for 30 U - 70 Zr	Min. for 25 U - 75 Zr
Pin dia (mm)	9.5	19.1	21.0	25.4	29.2	35.0	43.2
Miscellaneous Costs:							
Containment & Vessel (\$)	5.32E+07	6.14E+07	6.09E+07	6.24E+07	6.12E+07	6.34E+07	6.33E+07
Pump Capital (\$)	2.31E+05	2.40E+05	2.82E+05	2.75E+05	3.15E+05	3.40E+05	4.04E+05
Pump Operating (\$)	6.64E+06	6.89E+06	7.52E+06	7.92E+06	9.06E+06	9.79E+06	1.16E+07
Initial Core Costs:							
Fab. Median (\$)	1.36E+07	6.55E+06	8.26E+06	8.25E+06	7.80E+06	7.82E+06	7.39E+06
Enrichment (\$)	1.46E+07	1.50E+07	1.48E+07	1.48E+07	1.46E+07	1.46E+07	1.45E+07
Ops (\$)	2.41E+07	2.60E+07	2.54E+07	2.55E+07	2.46E+07	2.48E+07	2.38E+07
Refueling Costs (Levelized):							
Fab. Median (\$)	4.71E+06	2.51E+06	2.54E+06	2.53E+06	2.59E+06	2.60E+06	2.65E+06
Enrichment (\$)	8.15E+07	7.81E+07	7.90E+07	7.89E+07	8.05E+07	8.05E+07	8.23E+07
Ops (\$)	1.11E+08	1.07E+08	1.08E+08	1.08E+08	1.10E+08	1.10E+08	1.12E+08
Total Costs:	3.42E+08	3.23E+08	3.24E+08	3.26E+08	3.26E+08	3.31E+08	3.38E+08

3.1.4 Design Comparison

All optimized (highlighted) designs are shown in Table 18. The trends with the design parameters as a function of alloying percentage are given below.

Since the optimum pin diameters increase with increasing zirconium percentages, it is only natural that the pitch must also increase. The active length of the core also increases. This is due to maintaining a constant MDNBR, as previously reported by Lancaster, et. al. [12] The core volume increases with increasing pin outer diameter. This is because, as the pin diameters become larger, the amount of metal-alloy increases the volume of the core. It is also due to the aforementioned pin length increasing. The containment volume, however, remains moderately constant (decreasing slightly). This is due to the amount of water in the core decreasing. The water decreases because the H/U ratio is fixed, and as can be seen from Table 18, the metric tonnes of uranium (MTU) in the core decreases. The decrease in MTU also accounts for the power density (Watts/gram-U) increasing. The pressure drop increases because the flow area per channel decreases. The refueling enrichment increases very slightly. Finally, the peak fuel temperatures increase with increasing zirconium composition.

It should be noticed by the reader, that the power

densities for the metallic cores vary only slightly from the fifty (50) weight percent zirconium design to the seventy-five (75) weight percent design. This implies that there exists an optimum power density for the optimized metallic cores. Therefore, the optimum number of batches fluctuates around three (3). The following equation shows how the number of batches [31] correlates to the power density.

$$\text{Batches} = \text{Discharge Burnup} / (\text{Cycle Length} * \text{Power Density}) \quad /4/$$

The power density, however, is simply the reactor power divided by the MTU. Therefore:

$$\text{Batches} = (\text{Burnup} * \text{MTU}) / (\text{Cycle Length} * \text{Reactor Power}) \quad /5/$$

The discharge burnup, cycle length, and reactor power are fixed quantities in this study. Therefore, the only variables to equation /5/ are the number of batches and the MTU. The MTU would remain essentially constant in each design if the pump costs were neglected in the cost optimization, and therefore, the number of batches would also be constant. At the higher weight percent zirconium alloys, the smaller channel flow areas increase the pressure drop. This increases the pumping costs. The pumping costs become a major contributor to the overall optimization, and drive the higher zirconium content alloys to a lower optimum pin

2 of 4

diameter. These lower pin diameters (at a constant alloying percentage) lowers the MTU needed in the core, which increases the power density, and ultimately decreases the number of batches for the high zirconium alloys.

Table 18: Design Parameters for Optimized Alloying Contents

	AP-600	Min. for 50 U - 50 Zr	Min. for 45 U - 55 Zr	Min. for 40 U - 60 Zr	Min. for 35 U - 65 Zr	Min. for 30 U - 70 Zr	Min. for 25 U - 75 Zr
Pin dia (mm)	9.5	19.1	21.8	25.4	29.2	35.6	43.2
Pitch (mm)	12.8	21.9	24.2	27.7	31.0	36.8	43.6
Length (m)	3.66	4.29	4.34	4.49	4.53	4.74	4.86
Number of fuel pins	36280	17398	14634	11939	9688	7515	5644
Core Volume (l)	24532	39531	40967	45363	46653	53217	57403
Containment Volume (l)	5.066E+07	5.431E+07	5.321E+07	5.332E+07	5.154E+07	5.156E+07	4.995E+07
MTU Core Total	67.71	80.07	76.31	78.70	70.69	70.83	65.33
Initial Enrichment (%)	2.7	2.4	2.5	2.5	2.6	2.6	2.7
Power W/(gram-U)	27.81	23.51	24.67	24.55	26.63	26.58	26.82
Linear Power (kW/m)	13.4	25.2	29.8	35.1	42.9	52.9	68.7
Cycle Burnup (MWD/MTU)	15621	13209	13061	13791	14962	14934	16190
No. Batches Per Core	2.681	3.407	3.246	3.263	3.008	3.013	2.779
Pressure Drop (kpa)	90.81	93.99	102.62	108.07	123.64	133.51	158.31
Pressure Drop (kPa/m)	24.77	21.91	23.64	24.07	27.28	28.19	32.59
No. of Rods Refueled	13288	5107	4508	3659	3221	2494	2030
Refuel Enrichment (%)	4.4	4.2	4.3	4.3	4.3	4.3	4.4
Peak Fuel Temp. (deg.F)	2920	1320	1452	1612	1778	1988	2297

3.1.5 Fuel Temperatures

This section gives the radial temperature profiles for the metallic fuel pins. It should be noted that the temperature profiles given are for the hottest axial fuel location in the hottest radial pin. This is due to the limiting axial heat flux profile and radial peaking factor used in COBRA-IV-PC for the thermal analyses. Only the profiles of each alloying content's optimized design are shown in Figures 15 through 20. Also shown in the figures is the fuel phase transition temperature for the alloy.

It can be seen from the aforementioned figures, that for the limiting cases presented, the fuel phase transition temperature is exceeded by all designs. The amount of fuel that exceeds the transition temperature increases with increasing zirconium alloying percentage. Approximately sixty-one (61) percent of the fuel (in a radially cut cross-section) exceeds the temperature for the fifty (50) weight percent uranium - fifty (50) weight percent zirconium design. At the other extreme, approximately eighty-eight (88) percent of the fuel exceeds the phase transition temperature for the twenty-five (25) weight percent uranium - seventy-five (75) weight percent zirconium design. The detrimental effects of exceeding the transition temperature are primarily due to possible dimensional instability. [32]

HOT SPOT RADIAL TEMPERATURE PROFILE
Optimized 50 wt. % U - 50 wt. % Zr

Figure 15: Hot Spot Radial Temperature Profile of Optimized 50 wt. % U - 50 wt. % Zr Alloy

Figure 16: Hot Spot Radial Temperature Profile of Optimized 45 wt. % U - 55 wt. % Zr Alloy

Figure 17: Hot Spot Radial Temperature Profile of Optimized 40 wt. % U - 60 wt. % Zr Alloy

HOT SPOT RADIAL TEMPERATURE PROFILE Optimized 35 wt. % U - 65 wt. % Zr

Figure 18: Hot Spot Radial Temperature Profile of Optimized 35 wt. % U - 65 wt. % Zr Alloy

Figure 19: Hot Spot Radial Temperature Profile of Optimized 30 wt. % U - 70 wt. % Zr Alloy

Figure 20: Hot Spot Radial Temperature Profile of Optimized 25 wt. % U - 75 wt. % Zr Alloy

CHAPTER IV

Sensitivity of Results

4.1 Assumption Sensitivity

This portion of the study concentrates on how sensitive a design is to certain key assumptions. The design that will be considered is the fifty (50) weight percent uranium - fifty (50) weight percent zirconium alloy-fueled core. The assumptions studied include ore costs, enrichment costs, cycle length, and discharge burnup. All other parameters will be maintained at their original values except the parameter being considered.

4.1.1 Ore Costs

Ore costs for refueling the core over the life of the plant are the largest contributor to the total costs. Therefore, the sensitivity analysis will begin by examining how the ore (yellow cake) cost effects the optimization study.

The first condition considers what occurs when the U_3O_8 cost is \$10/LB. Although the total costs for the design decrease significantly, the optimum pin diameter

remains the same. In fact, the pin diameter would not decrease unless the cost of yellow cake fell below \$4/LB. The same situation (in reverse) holds true for higher ore costs. At \$80/LB for U_3O_8 , the total costs increase, but the pin diameter does not increase. The cost for yellow cake would have to be \$104/LB or higher before the pin diameter would be effected.

Figure 21 shows the sensitivity of the design for the \$10/LB U_3O_8 circumstance. Figure 22 displays the results for \$80/LB of yellow cake.

Figure 21: Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - \$10/LB U_3O_8

TOTAL COST vs PIN OUTER DIAMETER 50 w/o U-50 w/o Zr-\$80/LB Yellow Cake

Figure 22: Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - \$80/LB U₃O₈

4.1.2 Enrichment Costs

From 1978 to 1986, the cost of enrichment for the AVLIS process was approximately \$25/SWU (levelized to 1986\$) at the Lawrence Livermore National Laboratory. [11] If these costs could be maintained on a larger scale through technological advancement, then the total cost for enrichment would be \$40/SWU. This would include the previously mentioned \$15/SWU for the re-alloying of the AVLIS product. At \$40/SWU, the optimum design of a metal-fueled core would not be altered, but lower total costs would result. In a worst case scenario, AVLIS would not become commercialized. An enrichment cost of \$120/SWU (approximate gaseous diffusion enrichment cost) was assumed for this analysis. Utilizing these assumptions, the optimum pin diameter for the metal-fueled core would shift to the next highest value considered. These results are shown in Figure 23 and Figure 24 for the two cases presented.

The actual enrichment costs that make the pin diameters move to lower and higher values are \$27/SWU and \$108/SWU, respectively. It is believed that the cost per SWU for the AVLIS process will be maintained within this window. Therefore, while enrichment costs will have an effect on total costs, it should not effect the optimum pin diameter of the chosen design.

Figure 23: Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - \$40/SWU Enrichment

TOTAL COST vs PIN OUTER DIAMETER 50 w/o U-50 w/o Zr-\$120/SWU Enrichment

Figure 24: Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - \$120/SWU Enrichment

4.1.3 Cycle Length

This study used a cycle length of twenty-four (24) months in the analysis of the optimum pin diameter. If the current PWR cycle length, eighteen (18) months is considered, then the optimum pin diameter decreases to the next lower value with lower associated costs. If the cycle length is increased to thirty-six (36) months, the pin diameter increases with higher affiliated costs. The window for the decrease and increase of optimum pin diameters is twenty-one (21) months and thirty-five (35) months, respectively. These results are shown in Figure 25 and Figure 26.

Since all other parameters are held constant, we see the above results due to the batch effect. For decreasing cycle length, the cycle burnup decreases, the number of batches increases, and the initial and refuel enrichments decrease. The opposite holds true for greater cycle lengths. The reason for the decrease in pin diameter at the lower cycle length is due to the difference in initial enrichment costs dominating the total costs. These cost are far less for the lower pin diameter. For the greater cycle length, the pin diameter increases because the refueling costs are much less than the original optimum.

It should be noted that this analysis does not consider the engineering and plant down-time costs associated

- With a refueling outage. These can be on the order of a million (10^6) dollars per day in costs and lost revenues to a utility.

Figure 25: Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - 18 Month Cycle Length

Figure 26: Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - 36 Month Cycle Length

4.1.4 Discharge Burnup

It is generally known that increasing the discharge burnup of the fuel is economically advantageous. Burnup also effects the optimum pin diameter for the design considered.

At a lower burnup value, the total costs increase and the optimum pin diameter also increases. This is graphically shown for thirty-thousand (30 000) megawatt-days per metric tonne uranium (MWD/MTU) in Figure 27. For higher discharge burnups, the total costs decrease and the pin diameter of the design also decreases. This is shown for eighty-thousand (80 000) MWD/MTU in Figure 28. To shift the optimum pin diameter up or down one value on the graph, the discharge burnup must be thirty-two-thousand (32 000) MWD/MTU or fifty-thousand (50 000) MWD/MTU, respectively.

The discharge burnup of the fuel is, by far, the most important parameter in terms of the coupled effects on total costs and pin diameter. This is due to the previously mentioned batch effect. For the lower discharge burnup, the number of batches goes down. The refueling enrichment goes down, but the initial enrichment increases. Since the number of batches decreases, more rods need to be replaced during an outage. It is for this reason that the optimum pin diameter increases. At the larger pin diameter, fewer pins can be replaced to refuel the same amount of uranium.

The refueling costs for the higher pin diameter are less than the refueling costs for the original optimum, and dominate the total costs. When the discharge burnup was increased, the opposite effects were discovered. The optimum pin diameter decreased due to the effects on cost generated by the initial core costs. These costs were less than the prior optimum pin diameter, and dominated the overall costs.

Figure 27: Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - 30 000 MWD/MTU Burnup

Figure 28: Total Costs Versus Pin Outer Diameter for 50 wt% U - 50 wt% Zr - 80 000 MWD/MTU Burnup

CHAPTER V

Conclusions

5.1 Recommendations and Discussion

This portion of the study recommends a metallic fuel core design for consideration in pressurized light water reactors. A discussion of the recommended design follows. Also discussed are the strengths and concerns of the study in relation to the design. Finally, suggestions for future work on the subject of metallic fuels and core optimization techniques are given.

5.1.1 Design Recommendation

Upon examination of the results presented in Chapter III, it is recommended that the most feasible design for a metal-alloyed reactor is the fifty (50) weight percent uranium - fifty (50) weight percent zirconium alloy. Since economics is the "bottom-line" of our society, costs for the design will be discussed first. Thereafter, an inter-location on the design parameters will follow.

Reviewing Table 17, reveals that the above alloy is the least expensive over a thirty (30) year life of plant

analysis period. The savings that could be realized from a metallic fueled plant over the AP-600 design is approximately \$19 million dollars over a levelized thirty (30) year period. This value, however, does not represent a substantial enough savings to warrant changing PWR fuels.

The design parameters associated with choosing the aforementioned alloy were presented in Table 18. After evaluating the optimized designs, the recommended design demonstrates the smallest pin diameter, the smallest pitch, the shortest length, and the least core volume. Also exhibited by the recommended design are the lowest pressure drop and pressure drop per metre, and minimum initial and refueling enrichment percentages. Of all the optimized designs, the fifty (50) weight percent uranium - fifty (50) weight percent zirconium alloy displays parameters that most closely resemble current nuclear reactor cores.

5.1.2 Discussion of Recommendation

It was shown in Section 3.1.3 (Cost Comparison) that the difference in total costs only differ by approximately thirteen (13) million dollars. This figure is rather insignificant when compared to the total costs for the entire plant. Therefore, cost is not the motive for recommending the aforementioned alloy. The two main concerns of the metal fuel are corrosion and the fuel phase temperature

transition. Due to corrosion concerns, high zirconium content alloys are favored in the event of fuel pin failure. For temperature transition reasons, lower zirconium contents are preferential. It is believed by the author that the disadvantages associated with the fuel transition temperature offsets the corrosion advantages exhibited by higher zirconium content alloys. Although the recommended design exceeds the phase transition temperature, it is the most favorable in these regards.

It was recommended in the previous section that any further work for this binary system be for the fifty (50) weight percent uranium - fifty (50) weight percent zirconium alloy. There are strengths and weaknesses to this recommendation.

One strength of this study is that all possible binary alloying systems were considered for a feasible design. This leaves little doubt to which alloy is considered the optimum. The corrosion analysis of the binary systems is reasonably detailed, and the author feels confident about decisions regarding this subject. Also, the optimization techniques first presented by Marsh [13] were remodeled to reflect new available data and new assumptions. Albeit, there are weaknesses in the study also.

The author is aware of two primary uncertainties in the study. These uncertainties include the thermal analy-

sis, and the optimization of the reference design.

One complication associated with the thermal analysis of the metal-alloyed core is the determination of the critical heat flux (CHF). From Section 3.1.1 (Designs), it was shown that the optimized designs generated cores longer than twelve (12) feet. The CHF option used in COBRA-IV-PC was the W-3 correlation. Since the core lengths exceed the range of W-3, the values obtained from this correlation were checked with table look-up methods. The values for MDNBR found by the tables varied slightly from the values discovered with W-3. Although table look-up techniques reduce the error for predicted CHF values [29], it is very cumbersome to match MDNBRs with this method. A suitable CHF correlation for cores longer than twelve (12) feet would help reduce the uncertainty of the optimization in regards to matching the MDNBR.

Another concern with the thermal analysis is the thermal conductivity of the fuel. It was shown in Section 3.1.5 (Fuel Temperatures), that the fuel phase transition temperature was exceeded by the metallic alloys. Although the recommended design is the optimum in this regard, it also transgresses the phase change temperature for the limiting case presented. The author perceives that this is due to the lack of an appropriate thermal conductivity correlation in the thermal analysis of the fuel. The recom-

mended design should not be discarded until a suitable correlation can be formulated, and auxiliary thermal evaluations of the fuel can be performed.

Another area of concern is in regard to the reference design. There exists the possibility that the AP-600 design is not optimized. It was reported by Lancaster and Pfeifer that the optimization of the AP-600 design should be examined further. [33] It is uncertain how further optimization of AP-600 would affect the optimization of the metallic core at the time of this study.

5.1.3 Future Work

In the last section, some suggestions for improving the study were expressed. These recommendations encompass; a CHF correlation for longer cores, a thermal conductivity correlation for the entire binary uranium - zirconium system, and more irradiation studies for the binary system. In addition to these suggestions, the author recommends that simultaneous optimization techniques be developed. One such technique may include the balancing of costs and operating conditions for a reactor design.

REFERENCES

1. Butcher, B.R. and Holden, A.N., "The Kinematics and Crystallography of the Transformation Between the Primary Phases of Uranium and its Dilute Alloys", Progress in Nuclear Energy, Vol. 2, Pergamon Press, 1959
2. Olander, D.R., Fundamental Aspects of Nuclear Reactor Fuel Elements, Published by Technical Information Center, Office of Public Affairs, Energy Research and Development Administration, Oak Ridge, TN, 1976, p. 463
3. Till, C.E. and Chang, Y.I., "Evolution of the Liquid Metal Reactor: The Integral Fast Reactor (IFR) Concept", Proceedings of the American Power Conference, 1989
4. Walters, L.C., Seidel, B.R., and Kittel, J.H., "Performance of Metallic Fuels and Blankets in Liquid-Metal Fast Breeders", Nuclear Technology, Vol. 65, Dec. 1983
5. Seidel, B.R. and Walters, L.C., "Progress in the Development of Metallic Fuel in Fast Reactors", Transactions of the American Nuclear Society, 56, June 1988, p. 68
6. Burris, L., et. al., "Proposed Pyrometallurgical Process for Rapid Recycle of Discharged Fuel Materials from the Integral Fast Reactor", Proceedings of the Fuel Reprocessing and Waste Management Meeting, Jackson, NY, August 26-29, 1984
7. Till, C.E., "Looking to the Future with the Integral Fast Reactor", Nuclear Engineering International, Vol. 30, Feb. 1985, p. 21
8. Weisman, J., Elements of Nuclear Reactor Design, Elsevier Scientific Publishing Company, New York, NY, 1977

REFERENCES

9. Pray, H.A. and Berry, W.E., "Corrosion of Uranium - Zirconium Alloys in Water up to 100°C", U.S. AEC Report, BMI-893, Battelle Memorial Institute, Dec. 1953
10. "Nuclear Energy Cost Data Base: A Reference Data Base for Nuclear and Coal Fired Powerplant Power Generation Cost Analysis", DOE/NE-0095, U.S. Dept. of Energy, Sept. 1988, p. 87
11. Davis, J.I. and Spaeth, M.L., "AVLIS: A Technical and Economic Forecast", Proceedings of Fuel Cycle 1986, Atomic Industrial Forum, Scottsdale, AR, April 1-4, 1986
12. Lancaster, D.B., Marsh, R.L., Bullen, D.B., Pfeifer, H., Erwin, C.S., and Levin, A.E., "A Technique for the Selection of the Fuel Pin Diameter for a Uranium/Zirconium Alloy-Fueled Pressurized Water Reactor", accepted for publication by Nuclear Technology, July 1991
13. Marsh, R.L., "Fuel Pin Optimization for a Metal Fueled Light Water Reactor", A thesis presented to the faculty, Nuclear Engineering and Health Physics Program, Georgia Institute of Technology, Dec. 1989
14. Tower, S.N., Schulz, T.L., and Vijuk, R.P., "Passive and Simplified System Features for the Advanced Westinghouse 600 MWe PWR", Nuclear Engineering Design, 109, 147 (1988)
15. Vijuk, R.P., Personal Communication, February 1, 1991, 2:40 P.M.
16. Characteristics of Spent Fuel, High-level Waste, and Other Radioactive Wastes Which May Require Long-term Isolation, Appendix 2A. Physical Description of LWR Fuel Assemblies, U.S. Department of Energy, DOE/RW-0184, December 1987, pp. 343-346
17. Webb, B.J., "COBRA-IV PC: A Personal Computer Version of COBRA-IV-I for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores", Battelle Memorial Institute, January 1988

REFERENCES

18. Todreas, N.E. and Kazimi, M.S., Nuclear Systems I, Thermal Hydraulic Fundamentals, Hemisphere Publishing Corporation, 1990, p. 301
19. Tong, L.S., "Critical Heat Fluxes in Rod Bundles", Two-Phase Flow and Heat Transfer in Rod Bundles, ASME, New York, NY, 1969, pp. 31-46
20. Seidel, B.R. and Walters, L.C., "A Decade of Advances in Metallic Fuel", Transactions of the American Nuclear Society, 62, Washington, DC, November 11-15, 1990, pp. 260-261
21. Pray, H.A. and Berry, W.E., "Corrosion of Uranium Alloys in High Temperature Water", BMI-874, Battelle Memorial Institute, 1957
22. Grieser, D.R. and Simons, E.M., "Aqueous Corrosion of Uranium Fuel Element Cores Containing 0 to 20 weight percent Zirconium", BMI-1156, Battelle Memorial Institute, 1957
23. Grieser, D.R. and Simons, E.M., "Apparatus for Visual Study of Corrosion by Hot Water", BMI-998, Battelle Memorial Institute, 1955
24. Waldmen, L.A. and Lojek, J.M., "Erosion of UO_2 Fuel Material", Westinghouse Atomic Power Division, WAPD-PWR-CP-1814, January 1958
25. Reynolds, J.E, Berry, W.E., et. al., "Effects of Ternary Alloying Additions on the Corrosion Resistance of Epsilon Phase Uranium - Zirconium Alloys", BMI-1087, Battelle Memorial Institute, 1956
26. Kass, S. and Goldman, K.M., "Corrosion Behavior of Uranium - Zirconium Alloys in High Temperature Water and Steam", WAPD-T-416, 1958
27. "Evaluation of the Use of Metal Alloy Fuels In Pressurized Water Reactors, Quarterly Progress Report for the Period of April through June, Georgia Institute of Technology, GTNE-MF-03, July 1990

REFERENCES

28. Rough, F.A., "An Evaluation of Data on Zirconium - Uranium Alloys", USAEC Report BMI-1030, Battelle Memorial Institute, August 1955
39. Hewitt, G.F., Delhaya, J.M., and Zuber, N., Multi-phase Science and Technology, Vol. 2, Hemisphere Publishing Corporation, Washington, DC, 1986
30. "Capital Cost: Pressurized Water Reactor Plant", NUREG-0241 COO-2477-5, June 1977
31. Schneider, A., Georgia Institute of Technology, class notes from "Nuclear Fuel Cycle", Winter 1989
32. Reactor Handbook, Second Edition, Revised and Enlarged, Vol. I, Materials, Interscience Publishers, Inc., New York, NY, 1960 pp. 200-203
33. Lancaster, D.B. and Pfeifer, H., "Partial Optimization of the Fuel Pin Diameter for a Passive Pwr", Transactions of American Nuclear Society, 63, Orlando, FL, June 2-6, 1991, p. 438

APPENDIX A

ITMETAL.BAT LISTING

The following listing is a DOS batch file that was created to help automate the design and thermal analyses for the metal fuel alloys.

File: ITMETAL.BAT

c:

cd \

cd apps\dice

DEL TAPE6.TXT

DEL TAPE5.TXT

call p dice4.in

call dice4

COPY 1ST.LIN+2ND.LIN+3RD.LIN+4TH.LIN+5TH.LIN+6TH.LIN+
7TH.LIN+8TH.LIN+9TH.LIN+10TH.LIN+11TH.LIN TAPE5.TXT

call c:\apps\cobra\cobra

T TAPE6.TXT

COPY tape6.txt e:\ERWIN\THESIS\DATA\zr75\%1cobra.out

COPY tape5.txt e:\ERWIN\THESIS\DATA\zr75\%1cobra.in

copy dice4.in e:\ERWIN\THESIS\DATA\zr75\%1dice4.in

copy dice4.out e:\ERWIN\THESIS\DATA\zr75\%1dice4.out

APPENDIX A

DICE4 Code Listing

On the following pages, the Fortran code listing of the program, DICE4, is given. This program was created in an attempt to automate the design process for a metal-fueled PWR. The program is thoroughly commented and should pose little or no difficulties to the reader.

APPENDIX A

PROGRAM DICE4

```
.....
*
*   THIS IS A DESIGN ANALYSIS PROGRAM FOR A METAL-FUELED REACTOR.
*   IT USES THE WESTINGHOUSE AP-600 REACTOR CORE AS A REFERENCE CORE.
*   THE PROGRAM TAKES INPUT VARIABLES (PIN DIA., # OF ASSEMBLIES,
*   FRACTION OF URANIUM IN THE FUEL, SWELLING FRACTION OF THE FUEL,
*   DENSITY OF THE FUEL, THERMAL CONDUCTIVITY OF THE FUEL, SPECIFIC
*   HEAT OF THE FUEL, REACTOR POWER, CLAD THICKNESS, HYDROGEN-TO-
*   URANIUM RATIO, AND HEIGHT-TO-DIAMETER RATIO) AND CALCULATES THE
*   VARIABLES NECESSARY TO DESIGN A METAL-FUELED CORE AND ITS ASSO-
*   CIATED CONTAINMENT. THE PROGRAM ITERATES ON CORE HEIGHT AND THE #
*   OF ASSEMBLIES TO MATCH A DEPARTURE FROM NUCLEATE BOILING RATIO.
*   THIS PROGRAM IS ALSO USED AS A STARTING POINT FOR "COBRA" (THERMAL-
*   HYDRAULICS CODE). THE OUTPUT FILES GENERATED BY THIS CODE ARE
*   INPUT "CARDS" TO BE READ INTO "COBRA". THESE INDIVIDUAL "CARDS"
*   CAN BE MERGED TO FORM THE NEEDED "COBRA" INPUT FILE.
*
*.....
```

C This program was originally written by Craig Nisko (GT-BNE '88,
C GT-MSNE '90) for an uranium dioxide fueled reactor. The original
C program is called "DICE". "DICE1" is the original translation of
C "DICE" to metal fuel applications. Subsequent revisions to the
C program were included in "DICE2", "DICE3", and finally this version,
C "DICE4". This version of the program is thoroughly documented and
C should pose little or no difficulties to the reader.

C Revisions have been made by Holger Pfeifer (GT-BNE '88,
C GT-MSNE '89) and Steve Brwin (GT-BNE '90 and GT-MSNE '91) to reflect
C new data values supplied for AP-600 and errors that were found in
C Robert Marsh's (GT-BNE '88, GT-MSNE '90) original thesis. Revisions
C were also made to allow for more variable input and fewer hard-coded
C values.

C Variable descriptions:

C TITLE = PROBLEM STATEMENT
C PIND = PIN OUTER DIAMETER (IN.)
C PERI = 1/2 CHANNEL PIN PERIMETER FOR COBRA INPUT (IN.)
C FUEL = FUEL DIAMETER (IN.)
C MASSY = NUMBER OF ASSEMBLIES
C FRAS = NUMBER OF FUEL RODS PER ASSEMBLY

APPENDIX A

C	FURD	= NUMBER OF FUEL RODS
C	CLINR	= CLAD INNER RADIUS (IN.)
C	FUELA	= AREA OF FUEL PER PIN (SQ. IN.)
C	PI	= π (6)
C	VOID	= AREAL VOID FRACTION FOR FUEL EXPANSION
C	PITCH2	= PITCH ² (SQ. IN.)
C	PITCH	= PITCH (IN.)
C	GAP	= MINIMUM DISTANCE BETWEEN PINS (IN.)
C	FLOWA	= CHANNEL FLOW AREA (SQ. IN.)
C	CHFL	= 1/2 CHANNEL FLOW AREA FOR COBRA INPUT (SQ. IN.)
C	NDIA	= HYDRAULIC DIAMETER (IN.)
C	HU	= HYDROGEN TO URANIUM RATIO
C	DU	= DENSITY OF URANIUM (LB./IN. ³)
C	DU1	= DENSITY OF URANIUM - ZIRCONIUM FUEL ALLOY (LB./FT. ³)
C	KFUEL	= THERMAL CONDUCTIVITY OF FUEL ALLOY (BTU/HR./FT./deg. F)
C	CFUEL	= SPECIFIC HEAT OF FUEL ALLOY (BTU/LB./deg. F)
C	UWPT	= URANIUM WEIGHT PERCENT
C	DH	= DENSITY OF H ₂ O (LB./IN. ³)
C	MU	= MOLECULAR WEIGHT OF URANIUM
C	MH	= MOLECULAR WEIGHT OF H ₂ O
C	CT	= CLAD THICKNESS (IN.)
C	PRESS	= SYSTEM PRESSURE (PSIA)
C	X	= QUALITY
C	G	= CHANNEL MASS FLUX (LB./FT. ² /HR.)
C	GM	= CHANNEL MASS FLUX (MLB./FT. ² /HR.)
C	SATEN	= SATURATION ENTHALPY (BTU/LB.)
C	INEN	= INLET ENTHALPY (BTU/LB.)
C	TMFLR	= TOTAL MASS FLOW RATE (LB./HR.)
C	HD	= HEIGHT TO DIAMETER RATIO
C	LENGT	= ACTIVE CORE HEIGHT (IN.)
C	COVOL	= CORE VOLUME (FT. ³)
C	REFWAT	= AMOUNT OF WATER IN REFERENCE (AP-600) VESSEL (IN. ³)
C	WATVOL	= CALCULATED WATER VOLUME FOR METAL VESSEL (IN. ³)
C	WATRAT	= RATIO OF CALC. WATER VOL. TO REFERENCE WATER VOL.
C	CONTSI	= CONTAINMENT SIZE NECESSARY FOR VESSEL WATER VOL. (FT. ³)
C	CBF	= CORE BY-PASS FLOW
C	LATF	= LATTICE FRACTION
C	POW	= THERMAL POWER (MW)
C	Q	= THERMAL POWER (BTU/HR.)
C	QAVG	= AVERAGE HEAT FLUX (BTU/HR.- FT. ²)
C	QONB	= CRITICAL HEAT FLUX (BTU/HR.- FT. ²)
C	QAC	= ACTUAL HEAT FLUX (BTU/HR.- FT. ²)
C	QAVGM	= AVERAGE HEAT FLUX (MBTU/HR.- FT. ²)
C	POWVG	= POWER DENSITY (WATTS/GRAM-URANIUM)
C	LINPOW	= LINEAR POWER (KW/FT.)
C	TDNBR	= TARGET DNBR

APPENDIX A

```

C  DNBR  = DNBR
C  DIFF  = DELTA DNBR (DNBR - TDNBR)
C  TADJ  = TARGET ADJUSTMENT
C  ITNU  = ITERATION NUMBER
C  MAXIT  = MAXIMUM NUMBER OF ITERATIONS

```

C Variable definitions:

```

      IMPLICIT NONE

```

C Note: Variables set to single precision only due to the uncertainty
C in the data.

```

      REAL    CT,PI,MU,DU,DH,MU,MH,PRESS,X,SATEN,INEN,TMFLR
      REAL    PIND,VOID,LENGT,MASSY,TADJ,TDNBR,CLINR,FUELA
      REAL    FLOWA,MDIA,DNBR,GDNB,QAC,QAVG,G,FURD,COVOL
      REAL    CHFL,PERI,FUELD,POWVG,LINPOW,DU1,REFWAT,WATVOL
      REAL    QH,QAVGH,UMPT,POW,KFUEL,CFUEL,CBF,PITCH,CONTS1
      REAL    DIFF,DIFFO,G,LATF,GAP,WATRAT,MD,FRAS,PITCH2
      INTEGER I,FLAG,ITNU,MAXIT
      CHARACTER*60 TITLE

```

C Opening necessary files

```

      OPEN (1,FILE = '2ND.LIN')
      OPEN (2,FILE = '4TH.LIN')
      OPEN (3,FILE = '6TH.LIN')
      OPEN (4,FILE = '8TH.LIN')
      OPEN (5,FILE = '10TH.LIN')
      OPEN (8,FILE = 'DICE4.IN')
      OPEN (9,FILE = 'DICE4.OUT')

```

C Setting all fixed parameters

```

      FLAG = 0
      PI = 3.14159

```

C Note: The water density was incorrectly given in the Marsh thesis.
C The water density should have been 62.4 lb/ft³.

```

      DH = 62.4 / (12.0 * 12.0 * 12.0)
      MU = 238.0
      MH = 18.0

```

APPENDIX A

C Data for DNBR calculations checked by COBRA

PRESS = 2250.0
X = -0.0988
SATEN = 701.9
INEN = 521.7
TMFLR = 7.36E07

C Note: FRAS is calculated as a 17x17 assembly minus 25 rods for
C control and instrumentation

FRAS = 264.0
LATF = 0.9060

C Note: Core by-pass is assumed constant for AP-600 & metal reactor

CBF = 0.075

C Read input variables from file "DICE4.IN"

READ(8,*) TITLE
READ(8,90) PIND,NASSY,UWPT,VOID,DU1

C Note: NASSY is only an initial guess of the number of fuel
C assemblies

READ(8,100) TADJ,MAXIT,KFUEL,CFUEL
READ(8,110) POW,CT,MU,MD

C Note: This is the lattice H/U ratio for 17x17 OPA fuel.
C An active core H/U ratio would be more correct, but
C formulas would be more complex. This value is at STP.
C The Marsh thesis had incorrect pin dimensions. The
C theoretical density of the UO2 was changed to 95%.

C Note: The following values are variable input values but are
C held constant at the values shown below for the thesis.

C POW for AP-600 = 1933 MW
C CT for AP-600 = 0.0225 IN.
C MU for AP-600 = 4.802
C MD for AP-600 = 1.252

APPENDIX A

C Beginning calculations for design parameters

$$Q = POW * 3413.0 * 1000$$

$$DU = DU1 / (12.0 * 12.0 * 12.0) * UWPT$$

C Note: Since NASSY is an initial guess, FURD is also an initial
C guess for the number of fuel rods

$$FURD = NASSY * FRAS$$

C Note: 2.247 was the DNBR initially calculated by the Marsh thesis.
C This value has been updated to 2.586 utilizing new AP-600
C numbers. Since it is adjusted, the value is not important.
C This will only help convergence of the program.

$$TDNBR = 2.586/TADJ$$

C Calculate clad inner radius

$$CLINR = (PIND / 2) - CT$$

C Fuel area per pin (fuel coextruded with clad)

$$FUELA = (PI * (CLINR**2) / (1. + VOID))$$

C Calculate pitch

$$PITCH2 = ((MU * FUELA * DU * MH) / (2 * DH * MU)) + (PI * (PIND**2) / 4)$$

$$PITCH = \text{SQRT}(PITCH2)$$

C Calculate channel flow area

$$FLOWA = PITCH2 - (PI * PIND**2 / 4)$$

C Calculate hydraulic diameter

$$HDIA = 4 * FLOWA / (PI * PIND)$$

C Calculate length of core

$$\text{LENGT} = HD * 2.0 * \text{SQRT}(((PITCH2 * FURD) / (\text{LATF} * PI)))$$

C Call W-3 correlation to find critical heat flux for given conditions

```
CALL W3 (PRESS, X, HDIA, SATEN, INEN, TMFLR, FURD, CBF,
%      FLOWA, G, QDNB)
```

APPENDIX A

```

C Calculation of actual heat flux

C Assuming: 2.50 for the peaking factor
C           97.4% of energy deposited in fuel

C Note: The peaking factor was 2.34 in the original program. A new
C       value for the total peaking factor is 2.50.
C       (conversion factor - 144 in.**2/ft.**2)

      QAC = ((2.50 * 0.974 * Q) / (FURD * PI * PIND * LENGT)) * 144.0

C Calculate DNBR

      DNBR = QDNB / QAC

C Iterate on the # of assemblies to match DNBR till MAXITs exceeded

      DO 10 I = 1, MAXIT
        IF (FLAG .EQ. 2) GOTO 20
        CALL ITRAT (DNBR,TDNBR,NASSY,FLAG,DIFF,DIFFO)
        IF (FLAG .EQ. 2) GOTO 20

C Recalculate the number of fuel rods necessary

      FURD = NASSY * FRAS

C Recalculate the length of the core

      LENGT = HD * 2.0 * SQRT(((PITCH2 * FURD) / (LATF * PI)))

C Calculate critical heat flux based on new given conditions

      CALL W3 (PRESS, X, HDIA, SATEN, INEN, THFLR, FURD,
%          CBF, FLOWA, G, QDNB)

C Recalculate the actual heat flux

      QAC = (2.50 * 0.974 * Q) / (FURD * PI * PIND * LENGT)
%          * 144.0

C Recalculate the DNBR

      DNBR = QDNB / QAC
      ITNU = I

10 CONTINUE

```


APPENDIX A

C Test for convergence

IF (ITNU .GE. MAXIT) WRITE(6,*)'DID NOT CONVERGE!!!',ITNU

C Calculate average heat flux

20 QAVG = QAC/2.50

C Recalculate the number of fuel rods necessary

FURD = MASSY * FRAS

C Calculate core volume (conversion factor: $5.787 \times 10^{-4} \text{ ft}^3/\text{in}^3$)

COVOL = ((PITCH2 * LENGT * FURD / LATF) * .0005787)

C Calculate power density (W/g-U) and linear power (kW/Ft)

C (conversion factors 453.5924 Grams/Lb, 12 in/Ft, 3.413 Btu/Hr/Watt)

POWVG = Q * .974 / (FURD * FUELA * LENGT * DU * 453.5924 * 3.413)

LINPOW = Q * .974 * 12. / (FURD * LENGT) / 3413.

C Note: Calculating containment size required in Ft^3

C This method assumes a reference containment size from
C IPWR EEDB-IX (1 658 201 Ft^3). Due to the larger core of
C AP-600 (larger vessel volume), it's containment size is 1.0787
C times larger. The amount of reference water is calculated by
C taking standard AP-600 17x17 OFA assemblies to make up the
C reactor core. The REFWAT is 145 assemblies * 264 fuel rods per
C assembly * 144 in (length) * .1362 sq. in. (channel flow area)
C / (1 - core by-pass flow). The core by-pass flow is assumed to
C be the same in the metal core as in the AP-600 core, but is
C shown for methodology purposes. The ratio of the metal-fueled
C core water volume to the AP-600 core water volume is multiplied
C by 39.7%. This reflects the amount of water in the vessel as
C compared to the entire primary system. This assumes that the
C change in core volume is approximately equal to the change in
C the vessel volume.

REFWAT = (145. * 264. * 144. * .1362 / (1 - CBF))

WATVOL = (FURD * LENGT * FLOWA / (1 - CBF))

WATRAT = WATVOL / REFWAT

CONTS1 = 1658201 * 1.0787 * ((0.397 * WATRAT) + 0.603)

APPENDIX A

C Calculations that follow are for a COSRA flow channel

```
FUELD = PIND - (2 * CT)
GAP = PITCH - PIND
PERI = PI * PIND / 2.
CHFL = FLOWA / 2.
GM = G/1.E6
QAVGM = QAVG/1.E6
```

C Write design parameters to file, "DICE4.OUT"

```
WRITE(9,*) TITLE
WRITE(9,*)
WRITE(9,190) ITNU,DIFF,TADJ
WRITE(9,*)
```

C Note: An error message has been included for when the pin
C diameter is equal to or exceeds the pitch. This will
C occur at large weight percents of Zirconium due to
C holding the Hydrogen-to-Uranium ratio constant. This
C sets an upper geometric limitation on design.

```
IF (PIND .LT. PITCH) THEN
  WRITE(9,200) PIND,PITCH,FLOWA
ELSE
  WRITE(9,201) PIND,FLOWA
ENDIF
WRITE(9,*)
WRITE(9,205) NDIA,FUELD
WRITE(9,*)
WRITE(9,210) G,QAVG,LENGT
WRITE(9,*)
WRITE(9,220) COVOL,FURD
WRITE(9,*)
WRITE(9,225) NASSY,CONTSI
WRITE(9,*)
WRITE(9,230) POWMG, LINPOW
```

APPENDIX A

C Write COBRA input file "cards"

```
WRITE(1,1010)1,1,TITLE
WRITE(2,1020)1,1,CHFL,PER1,PER1,2,GAP
WRITE(2,1030)1,2,CHFL,PER1,PER1
WRITE(3,1040) PIND
WRITE(3,1050) KFUEL,CFUEL,DU1,FUELD,CT
WRITE(4,1060) LENGT
WRITE(5,1070) GM,QAVGM
RETURN
```

C Format statements for reading and writing data

C Read format statements

```
90 FORMAT (5F12.5)
100 FORMAT (F12.5,112,2F12.5)
110 FORMAT (4F12.5)
```

C Write format statements for design parameters

```
190 FORMAT (2X,'# OF ITERATIONS',2X,I5,2X,'DIFF',2X,E8.2E2,
      &      2X,'COBRA ADJ',F7.3)
200 FORMAT (2X,'PIND',2X,F6.4,'IN',2X,'PITCH',2X,F7.5,'IN',
      &      2X,'CH FLOW AREA',2X,F7.5,'IN.**2')
201 FORMAT (2X,'PIND',2X,F6.4,'IN',2X,'PITCH < OR = PIN DIA.',
      &      2X,'CH FLOW AREA',2X,F7.5,'IN.**2')
205 FORMAT (2X,'EQUIVALENT DIA. ',F5.3,'IN' FUEL DIA ',F5.3,'IN')
210 FORMAT (2X,'MASS FLUX',2X,E10.4E2,2X,'AVG. HEAT FLUX',E12.4,
      &      2X,'LENGTH',F8.3,'IN')
220 FORMAT (2X,'CORE VOL "Ft**3"',2X,F11.3,' # OF FUEL RODS',2X,F9.2)
225 FORMAT (2X,'# ASSEMBLIES ',E9.4,' CONTAINMENT SIZE "Ft**3" ',E9.4)
230 FORMAT (2X,'POWER "W/G-U"',2X,F9.4,2X,'LIN POWER "KW/FT"',2X,F9.3)
```

C Write statements for COBRA input file

```
1010 FORMAT(4X,I1,4X,I1,4X,A60)
1020 FORMAT(11,I4,3F5.3,I5,F5.3)
1030 FORMAT(11,I4,3F5.3)
1040 FORMAT(' 1 1',F5.3,' 1.55 1 .5 2 .5')
1050 FORMAT (F5.3,F5.4,F5.1,F5.3,'7.569.0802 410.',F5.4,'1.0e6')
1060 FORMAT(F5.1,4X,'0 .1 .01 .001 .5 .5 .0 .0 .0 .8',
      &      ' 1.6 1.0')
1070 FORMAT(' 2250. 529.1',E10.4,E10.5)
```

END

APPENDIX A

```
*****
*
*   Begin subroutines called by "DICE4"
*
*****
```

C Tong's W-3 correlation for calculation of critical heat flux

```
SUBROUTINE W3 (P,X,HD,SEN,IEN,TMFLR,FURD,CBF,FLOWA,G,QDNB)
```

C See main program, "DICE4", for variable descriptions

C Variable definitions:

```
IMPLICIT NONE
REAL    P,X,HD,SEN,IEN,TMFLR,FURD,CBF,FLOWA,QDNB,G
```

C Calculate channel mass flux

```
G=(( TMFLR * (1 - CBF)) / (FURD * FLOWA / 144))
```

C W-3 correlation:

```
QDNB = (((2.022 + 0.0004302 * P) + ( 0.1722 + 0.0000984 * P)
&      *EXP((18.77 + 0.004129 * P) * X))
&      *((0.1484 + 1.596 * X + 0.1729 * X * ABS(X))
&      *(G / 1.0E06) + 1.037) * (1.157 + 0.869 * X)
&      *(0.2664 + 0.8357 * EXP(-3.15 * HD))
&      *(0.8258 + 0.000794 * (SEN - IEN))) * 1E06
RETURN
END
```

C Iteration on the number of assemblies to match DNBR to target DNBR

```
SUBROUTINE ITRAT (DNBR,TDNBR,NASSY,FLAG,DIFF,DFFO)
```

C See main program, "DICE4", for variable descriptions

C Variable definitions:

```
IMPLICIT NONE
INTEGER FLAG
REAL    DIFF,DNBR, TDNBR, NASSY, DFFO
```

APPENDIX A

C Begin iteration of assemblies

```
DIFF = DNBR - TDNR
IF ((DIFF .LT. 0.0) .AND. (FLAG .EQ. 0)) THEN
    MASSY = MASSY + 1.0
    DIFFO = DIFF
    RETURN
ELSEIF (DIFF .GT. 0.0) THEN
    MASSY = MASSY - 0.01
    FLAG = 1
    DIFFO = DIFF
    RETURN
ENDIF
IF (DIFF .LT. DIFFO) THEN
    FLAG = 2
    RETURN
ELSE
    MASSY = MASSY + 0.01
    FLAG = 2
    RETURN
ENDIF
END
```

APPENDIX B

Typical COBRA-IV-PC Input and Output Files

On the following page is a typical COBRA-IV-PC input file. This file is created by the ITMETAL.BAT file. The batch file takes certain permanent input lines and combines them with the input "cards" created by DICE4. To see which parameters are changed in the COBRA-IV-PC input file, refer back the top of page 103 in Appendix A.

Immediately following the input file, the generated output file follows. This file should be self-explanatory.

APPENDIX B

Typical COBRA-IV-PC input file:

```

100
  1  1  50% U 50% Zr 15% RADIAL FUEL EXPANSION 0.75" DIAMETER
  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
500.467.1 .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 .22046 .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
2000.634.0 .02565 .18813 671.90 1136.30 .19006 .27547 .000857
2100.642.9 .02616 .17491 683.60 1128.50 .18637 .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  0  0
.079 -.25 0.0
  3  14
  0.0 .35 .10 .40 .20 .55 .30 .80 .40 1.15 .50 1.35
  .55 1.45 .60 1.55 .65 1.67 .70 1.60 .75 1.45 .80 1.30
  .90 0.65 1.0 .20
  4  2  2  0  0  0  0
  1  1 .1511.1781.178 2 .113
  1  2 .1511.1781.178
  7  2  0  8  1  1  0
.011 1 .180 1 .323 1 .466 1 .609 1 .751 1
.894 11.037 1
  1  1.8
  2  1.8
  8  1  1  3  1  2  1  0
  1  1 .750 1.55 1 .5 2 .5
9.265.0645611.8 .7057.569.0802 410..02251.0e6
1000.0 0.0 0.0 0.0
  9  0  0  0  0  0
162.9 0 .1 .01 .001 .5 .5 .0 .0 .0 .8 1.6 1.0
 12  0 20 20 7
 10  0  0  0
.0035
 11  1  1  0  0  0  0  0  0

```

APPENDIX B

2250. 529.1 .18458+01.133488+00
12 3 0 0 7 0 0 0 0

APPENDIX B

Typical COBRA-IV-PC output file:

* * * * *

COBRA-IV PC - ADAPTED FROM COBRA IV-1
 PREPARED FOR THE ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
 UNDER CONTRACT E(45-1):1830
 AND FOR THE NUCLEAR REGULATORY COMMISSION - RELEASED MARCH 1976
 ORIGINAL SOURCE MODIFIED BY DR. AMIN ANGHAI
 DEPARTMENT OF NUCLEAR ENGINEERING
 OREGON STATE UNIVERSITY
 ADAPTED FOR USE WITH IBM-PC AND IBM-PC COMPATIBLES BY BRENT J. WEBB
 PACIFIC NORTHWEST LABORATORY
 OPERATED BY BATTELLE MEMORIAL INSTITUTE FOR
 US DEPARTMENT OF ENERGY UNDER CONTRACT DE-AC06-76RLO-1830
 RESTART AND DUMP OPTIONS NOT SUPPORTED

COBRA DIMENSION PARAMETERS

MC= 15	MD= 24	ME= 15	MW= 6
MM= 6	MP= 20	MX= 21	MY= 21
MY= 4	MK= 5	MZ= 12	MS= 24
MA= 6	ML= 10	MT= 6	MI= 4
MO= 7	NI= 8	N1= 6	N1= 540

INPUT FOR CASE 1 50% U 50% Zr 15% RADIAL FUEL EXPANSION 0.75" DIAMETER

DATE 8/ 3/1991 T

SUMMARY OF INPUT OPTIONS

GROUP	N1	N2	N3	N4	N5	N6	N7	N8	N9
1	18	0	0	0	0	0	0	0	0
2	0	0	0	1	0	0	0	0	0
3	14	0	0	0	0	0	0	0	0
4	2	2	0	0	0	0	0	0	0
7	2	0	8	1	1	0	0	0	0
8	1	1	3	1	2	1	0	0	0
9	0	0	0	0	0	0	0	0	0
10	0	0	0	0	0	0	0	0	0
11	1	1	0	0	0	0	0	0	0
12	3	0	0	7	0	0	0	0	0

OF LUID PROPERTY TABLE

P	T	VF	VG	MF	MG	VISC.	KF	SIGMA
14.7	211.90	.01672	26.80000	180.15	1150.50	.75234	.39059	.00379
100.0	327.90	.01774	4.43400	298.61	1187.80	.51171	.39247	.00299
500.0	467.10	.01975	.92830	449.50	1205.30	.27771	.36513	.00203
1000.0	544.80	.02159	.44590	542.40	1192.40	.23413	.33125	.00149
1250.0	572.60	.02250	.34540	578.60	1181.60	.22046	.31512	.00130
1500.0	596.40	.02346	.27690	611.50	1168.70	.20774	.29977	.00113
1750.0	617.30	.02450	.25680	642.30	1153.20	.19890	.28762	.00099

APPENDIX B

2000.0	636.00	.02565	.18813	671.90	1136.30	.19006	.27347	.00086
2100.0	642.90	.02616	.17691	683.60	1128.50	.18657	.27061	.00081
2200.0	649.60	.02670	.16270	695.30	1120.30	.18310	.26597	.00076
2300.0	656.10	.02728	.15133	707.00	1111.40	.18003	.26198	.00072
2400.0	662.30	.02791	.14067	718.80	1101.80	.17697	.25800	.00068
2500.0	668.30	.02860	.13059	730.90	1091.40	.17389	.25401	.00063
2750.0	682.50	.03077	.10717	763.00	1060.40	.16622	.24405	.00054
2850.0	687.80	.03192	.09811	777.30	1045.00	.16315	.24007	.00050
3000.0	695.30	.03431	.08404	802.50	1015.50	.15029	.11920	.00045
3100.0	700.50	.03701	.07322	825.40	986.80	.09351	.11881	.00041
3200.0	705.30	.04805	.05444	890.60	919.90	.10166	.13405	.00038

FRICITION FACTOR CORRELATION

CHANNEL TYPE 1 $FRIC = .079 \cdot RE^{-.250} + .0000$
WALL VISCOSITY CORRECTION TO FRICTION FACTOR IS INCLUDED

SINGLE PHASE HEAT TRANSFER CORRELATION

$hFILM = K/D (.023 \cdot RE^{.400} (.800) \cdot PR^{.400} (.400) + .000)$

TWO-PHASE FLOW CORRELATIONS

NO SUBCOOLED VOID CORRELATION
HOMOGENEOUS BULK VOID MODEL
HOMOGENEOUS MODEL FRICTION MULTIPLIER

HEAT FLUX DISTRIBUTION

X/L	RELATIVE FLUX
.000	.350
.100	.400
.200	.550
.300	.800
.400	1.150
.500	1.350
.550	1.450
.600	1.550
.650	1.670
.700	1.600
.750	1.450
.800	1.300
.900	.650
1.000	.200

SUBCHANNEL INPUT DATA

CHANNEL NO.	TYPE	AREA (SQ-IN)	WETTED PERIM. (IN)	HEATED PERIM. (IN)	HYDRAULIC DIAMETER (IN)	(ADJACENT CHANNEL NO., SPACING, CENTROID DISTANCE)
-------------	------	--------------	--------------------	--------------------	-------------------------	--

APPENDIX B

1	1	.151000	1.178000	1.178000	.912733	(	2	.113	.000)(	0	.000	.000)(	0	.000	,
2	1	.151000	1.178000	1.178000	.912733	(	0	.000	.000)(	0	.000	.000)(	0	.000	,

SPACER DATA

SPACER TYPE NO.		1	1	1	1	1	1	1	1
LOCATION (X/L)		.011	.180	.323	.466	.609	.751	.894	1.037

0 SPACER TYPE 1

CHANNEL DRAG		CHANNEL DRAG		CHANNEL DRAG		CHANNEL DRAG	
NO.	COEFF.	NO.	COEFF.	NO.	COEFF.	NO.	COEFF.
1	1.800	2	1.800				

OROD INPUT DATA

ROD TYPE DIA.		RADIAL POWER		FRACTION OF POWER TO ADJACENT CHANNELS (ADJ. CHANNEL NO.)														
NO.	NO.	(IN)	FACTOR															
1	1	.7500	1.5500	.5000(	1)	.5000(	2)	.0000(	0)	.0000(	0)	.0000(	0)	.0000(	0)	.0000(	0)	

OTHERMAL PROPERTIES OF FUEL MATERIAL

3 ORDER OF COLLOCATION

FUEL PROPERTIES				CLAD PROPERTIES					
TYPE	COND.	SP. HEAT	DENSITY	DIA.	COND.	SP. HEAT	DENSITY	THICK.	GAP COND.
NO. (B/HR-FT-F)	(B/LB-F)	(LB/FT3)	(IN.)	(B/HR-FT-F)	(B/LB-F)	(LB/FT3)	(IN.)	(B/HR-FT-F)	
1	9.27	.0645	611.8	.7050	7.57	.0802	410.0	.0225	*****

NONUNIFORM FUEL THERMAL CONDUCTIVITY (FUEL TYPE 1)

REFERENCE TEMP. 1000.0(F)/ K/KO = 1. + .00000E+00*(T-T0) + .00000E+00*(T-T0)**2 + .00000E+00

IMPLICIT SOLUTION WITH INLET FLOWS SPECIFIED

0CALCULATION PARAMETERS

LATERAL RESISTANCE FACTOR	.5000	CHANNEL LENGTH	168.9000 1
(S/L) PARAMETER	.5000	NUMBER OF AXIAL NODES	12
TURBULENT MOMENTUM FACTOR	.0000	AXIAL NODE LENGTH	14.0750 1
CHANNEL ORIENTATION	.0000 DEGREES	TOTAL TRANSIENT TIME	.0000 8
ROLL OPTION (0 - NO ROLL)	0	NUMBER OF TIME STEPS	0
		NOMINAL TIME STEP	***** 8

0DATA FOR IMPLICIT SOLUTION

EXTERNAL ITERATION LIMIT	20	MINIMUM INTERNAL ITERATIONS	7
INTERNAL ITERATION LIMIT	20	FRACTION DONOR CELL USTAR	.0000
CONVERGENCE FACTORS		ACCELERATION FACTORS	
EXTERNAL (DW/W)	.1000	CROSSFLOW SOLUTION	1.6000
INTERNAL (DW/W)	.0100	LATERAL DELTA-P	.8000
FLOW (DF/F)	.0010	FLOW	1.0000

0MIXING CORRELATIONS

SUBCOOLED MIXING, BETA = .0035

BOILING MIXING, BETA IS ASSUMED SAME AS SUBCOOLED

0OPERATING CONDITIONS

SYSTEM PRESSURE = 2250.0 PSIA

APPENDIX B

INLET ENTHALPY = 523.6 BTU/LB
 AVG. MASS VELOCITY = 1.865 MILLION LB/(HR-80FT)
 INLET TEMPERATURE = 529.1 DEGREES F
 AVG. HEAT FLUX = .133680 MILLION BTU/(HR-80FT)

0 UNIFORM INLET TEMPERATURE

0 FLOWS SPLIT FOR EQUAL PRESSURE GRADIENT (CHANNEL-FLOW)

1 - 5.432E-01 2 - 5.432E-01

DATA FROM ITERATIVE SOLUTION

ITERATION NO.	TOTAL INT ETERNAL ITERATIONS	LAST NODE OUT OF CONVERGENCE		MAXIMUM ERROR		FLOW	ENTHALPY
		W	P	INTERNAL	EXTERNAL		
1	140	0	0	12.7815	.0002	.0000	.0502
2	129	0	0	11.1608	.0001	.0000	.0000
3	127	0	0	2.4596	.0405	.0000	.0000

CHANNEL EXIT SUMMARY RESULTS

CASE 1 50% U 50% Zr 15% RADIAL FUEL EXPANSION 0.75" DIAMETER

DATE 8/ 3/1991 TIME

MASS BALANCE - -

MASS FLOW IN .10865E+01 LB/SEC
 MASS FLOW OUT .10865E+01 LB/SEC
 MASS FLOW ERROR .00000E+00 LB/SEC

ENERGY BALANCE - -

FLOW ENERGY IN .56891E+03 BTU/SEC
 ENERGY ADDED .10261E+03 BTU/SEC
 FLOW ENERGY OUT .72308E+03 BTU/SEC
 ENERGY ERROR .31558E+02 BTU/SEC

CHANNEL (NO.)	ENTHALPY (BTU/LB)	TEMPERATURE (DEG.F)	DENSITY (LB/FT3)	EQUIL QUALITY	VOID FRACTION	FLOW (LB/SEC)	MASS FLUX (MLB/HR-FT2)
1	665.53	631.98	39.37	.000	.000	.5432	1.8650
2	665.53	631.98	39.37	.000	.000	.5432	1.8650

CHANNEL RESULTS

CASE 1 50% U 50% Zr 15% RADIAL FUEL EXPANSION 0.75" DIAMETER

DATE 8/ 3/1991 TIME

BUNDLE AVERAGED RESULTS

DISTANCE (IN.)	DELTA-P (PSI)	ENTHALPY (BTU/LB)	TEMPERATURE (DEG-F)	DENSITY (LB/CU-FT)	EQUIL. QUALITY	VOID FRACTION	FLOW (LB/SEC)	MASS FLUX (MLB/HR-FT2)	VELOCITY (FT/SEC)
.0	13.6309	523.63	529.10	47.13	.000	.000	1.0865	1.8650	10.9922
14.1	13.1890	528.15	532.88	46.93	.000	.000	1.0865	1.8650	11.0386
28.1	11.6326	533.49	537.35	46.70	.000	.000	1.0865	1.8650	11.0934
42.2	10.0699	540.46	543.17	46.40	.000	.000	1.0865	1.8650	11.1649
56.3	9.6289	549.96	550.61	45.91	.000	.000	1.0865	1.8650	11.2833
70.4	8.0367	562.92	560.56	45.24	.000	.000	1.0865	1.8650	11.4521
84.4	7.5990	578.38	572.43	44.46	.000	.000	1.0865	1.8650	11.6534
98.5	5.9638	595.87	585.09	43.47	.000	.000	1.0865	1.8650	11.9172
112.6	5.5293	615.51	599.12	42.38	.000	.000	1.0865	1.8650	12.2237

APPENDIX B

126.7	3.8358	634.72	612.16	41.25	.000	.000	1.0865	1.8650	12.5598
140.8	2.1212	650.89	622.73	40.27	.000	.000	1.0865	1.8650	12.8652
154.8	1.7157	660.80	628.99	39.65	.000	.000	1.0865	1.8650	13.0647
168.9	.0000	665.53	631.98	39.37	.000	.000	1.0865	1.8650	13.1599

CHANNEL RESULTS

CASE 1 50% U 50% Zr 15% RADIAL FUEL EXPANSION 0.75" DIAMETER DATE 8/ 3/1991 TIME
TIME = .00000 SECONDS PRESSURE = 2250.0 PSIA DATA FOR CHANNEL 1

DISTANCE (IN.)	DELTA-P (PSI)	ENTHALPY (BTU/LB)	TEMPERATURE (DEG-F)	DENSITY (LB/CU-FT)	EQUIL. QUALITY	VOID FRACTION	FLOW (LB/SEC)	MASS FLUX (MLB/HR-FT2)	VELOCITY (FT/SEC)
.0	13.6309	523.63	529.10	47.13	.000	.000	.5432	1.8650	10.9922
14.1	13.1890	528.15	532.88	46.93	.000	.000	.5432	1.8650	11.0386
28.1	11.6326	533.49	537.35	46.70	.000	.000	.5432	1.8650	11.0934
42.2	10.0699	540.46	543.17	46.40	.000	.000	.5432	1.8650	11.1649
56.3	9.6289	549.96	550.61	45.91	.000	.000	.5432	1.8650	11.2833
70.4	8.0367	562.92	560.56	45.24	.000	.000	.5432	1.8650	11.4521
84.4	7.5990	578.38	572.43	44.46	.000	.000	.5432	1.8650	11.6534
98.5	5.9638	595.86	585.09	43.47	.000	.000	.5432	1.8650	11.9172
112.6	5.5293	615.51	599.12	42.38	.000	.000	.5432	1.8650	12.2237
126.7	3.8358	634.72	612.16	41.25	.000	.000	.5432	1.8650	12.5598
140.8	2.1212	650.89	622.73	40.27	.000	.000	.5432	1.8650	12.8652
154.8	1.7157	660.80	628.99	39.65	.000	.000	.5432	1.8650	13.0647
168.9	.0000	665.53	631.98	39.37	.000	.000	.5432	1.8650	13.1599

CHANNEL RESULTS

CASE 1 50% U 50% Zr 15% RADIAL FUEL EXPANSION 0.75" DIAMETER DATE 8/ 3/1991 TIME
TIME = .00000 SECONDS PRESSURE = 2250.0 PSIA DATA FOR CHANNEL 2

DISTANCE (IN.)	DELTA-P (PSI)	ENTHALPY (BTU/LB)	TEMPERATURE (DEG-F)	DENSITY (LB/CU-FT)	EQUIL. QUALITY	VOID FRACTION	FLOW (LB/SEC)	MASS FLUX (MLB/HR-FT2)	VELOCITY (FT/SEC)
.0	13.6309	523.63	529.10	47.13	.000	.000	.5432	1.8650	10.9922
14.1	13.1890	528.15	532.88	46.93	.000	.000	.5432	1.8650	11.0386
28.1	11.6326	533.49	537.35	46.70	.000	.000	.5432	1.8650	11.0934
42.2	10.0699	540.46	543.17	46.40	.000	.000	.5432	1.8650	11.1649
56.3	9.6289	549.96	550.61	45.91	.000	.000	.5432	1.8650	11.2833
70.4	8.0367	562.92	560.56	45.24	.000	.000	.5432	1.8650	11.4521
84.4	7.5990	578.38	572.43	44.46	.000	.000	.5432	1.8650	11.6534
98.5	5.9638	595.87	585.09	43.47	.000	.000	.5432	1.8650	11.9172
112.6	5.5293	615.51	599.12	42.38	.000	.000	.5432	1.8650	12.2237
126.7	3.8358	634.72	612.16	41.25	.000	.000	.5432	1.8650	12.5598
140.8	2.1212	650.89	622.73	40.27	.000	.000	.5432	1.8650	12.8652
154.8	1.7157	660.80	628.99	39.65	.000	.000	.5432	1.8650	13.0647
168.9	.0000	665.53	631.98	39.37	.000	.000	.5432	1.8650	13.1599

APPENDIX B

CASE 1 50% U 50% Zr 15% RADIAL FUEL EXPANSION 0.75" DIAMETER

DATE 8/ 3/1991 TIME

TIME = .00000 SECONDS

DIVERSION CROSSFLOW BETWEEN ADJACENT CHANNELS, W(I,J) (LB/SEC-FT).,

AXIAL ZONE	W(1, 2)
.0 - 14.1	.00000
14.1 - 28.1	.00000
28.1 - 42.2	.00000
42.2 - 56.3	.00000
56.3 - 70.4	.00000
70.4 - 84.4	.00000
84.4 - 98.5	.00000
98.5 - 112.6	.00000
112.6 - 126.7	.00000
126.7 - 140.8	.00000
140.8 - 154.8	.00000
154.8 - 168.9	.00000

CASE 1 50% U 50% Zr 15% RADIAL FUEL EXPANSION 0.75" DIAMETER

DATE 8/ 3/1991 TIME

TIME = .00000 SECONDS

TEMPERATURE DATA FOR ROD 1 (FUEL TYPE 1 - CYLINDER)

ROD O.D. - .750 (IN.)

ZONE-(FUEL DIA.(IN.)) - 1-(.705)

AXIAL ZONE (IN.)	HEAT FLUX (MBTU/HR-FT2)	DNBR	CHANNEL	MSURF (B/M-F-FT2)	FLUID	CLAD	TEMPERATURES(F.) (RELATIVE RADIUS(R/R))				
							T(1) (1.000)	T(2) (.833)	T(3) (.667)	T(4) (.500)	T(5) (.333)
.0 - 14.1	.0768	22.976	2	4432.4	532.9	550.2	570.6	610.1	642.5	667.7	68
14.1 - 28.1	.0907	19.059	2	4435.5	537.3	557.8	581.8	628.5	666.7	696.4	71
28.1 - 42.2	.1183	14.195	2	4439.8	543.2	569.8	601.1	662.1	711.9	750.7	77
42.2 - 56.3	.1614	9.991	2	4589.6	550.6	585.8	628.5	711.7	779.8	832.7	87
56.3 - 70.4	.2202	6.925	2	4579.9	560.6	608.6	666.9	780.3	873.2	945.3	99
70.4 - 84.4	.2625	5.415	2	4568.4	572.4	629.9	699.3	834.6	945.2	1031.3	109
84.4 - 98.5	.2970	4.401	2	4663.6	585.1	648.8	727.4	880.4	1005.6	1103.0	117
98.5 - 112.6	.3336	3.543	2	4765.8	599.1	669.1	757.4	929.3	1069.9	1179.3	125
112.6 - 126.7	.3263	3.258	2	4743.8	612.2	681.0	767.3	935.5	1073.1	1180.1	125
126.7 - 140.8	.2745	3.470	2	4861.7	622.7	679.2	751.8	893.3	1009.1	1099.1	116
140.8 - 154.8	.1684	5.066	2	4849.1	629.0	663.7	708.3	795.0	866.0	921.2	96
154.8 - 168.9	.0803	9.661	2	4843.1	632.0	648.6	669.8	711.2	745.0	771.4	79

CASE 1 50% U 50% Zr 15% RADIAL FUEL EXPANSION 0.75" DIAMETER

DATE 8/ 3/1991 TIME

TIME = .00000 SECONDS

W-3 CRITICAL HEAT FLUX SUMMARY

DISTANCE	FLUX	DNBR	ROD	CHANNEL
.0	.000	.000	0	0

APPENDIX B

14.1	.077	22.976	1	2
28.1	.091	19.059	1	2
42.2	.118	14.195	1	2
56.3	.161	9.991	1	2
70.4	.220	6.925	1	2
84.4	.262	5.415	1	2
98.5	.297	4.401	1	2
112.6	.334	3.543	1	2
126.7	.326	3.258	1	2
140.8	.275	3.470	1	2
154.8	.168	5.066	1	2
168.9	.080	9.661	1	2

ITERATIONS = 3

SPECIFIED NUMBER OF TIME STEPS COMPLETED

APPENDIX C

The following pages give the spreadsheet parameters and formulae necessary to calculate the costs for AP-600 and one metal design. It is the intention of the author, that the reader will be able to reproduce the cost optimization process with this data.

APPENDIX C

A1: [W23] 'Table 3: Spreadsheet for cost optimization
 A3: [W23] 'Constants:
 A4: [W23] '-----
 A5: [W23] 'Interest rate
 B5: [W12] 0.0435
 C5: [F3] '(assumes no inflation Page 19 of DOE/NE-0095)
 A6: [W23] 'Analysis period
 B6: [W12] 30
 C6: [F3] 'yr
 A7: [W23] 'P/A,i,n
 B7: [W12] $((1+B5)^{B6}-1)/(B5*(1+B5)^{B6})$
 C7: [F3] '(discrete)
 A8: [W23] 'Vessel Cost
 B8: [W12] 9213
 C8: [F3] '(basis is active core vol, 1987 numbers not escalated or moved to startup)
 A9: [W23] 'Core Mass Flow Rate
 B9: [W12] 20444.44
 C9: [F3] 'lb/sec
 A10: [W23] 'Pump Cap. Cost/ hp
 B10: [W12] 150.7
 C10: [F3] '\$/hp
 A11: [W23] 'Water dens. at 560
 B11: [W12] 45.9
 C11: [F3] 'lb/ft³
 A12: [W23] 'Elec. Cost for pumps
 B12: [W12] 0.04
 C12: [F3] '\$/kwhr
 A13: [W23] 'Oxide Fab/pin
 B13: [W12] 340
 C13: [F3] '\$/pin basis \$220/kg fab
 A14: [W23] 'Clad thickness
 B14: [W12] 0.0225
 C14: [F3] 'inches
 A15: [W23] 'Zr wieght fraction
 B15: [W12] 0.5
 A16: [W23] 'Zr cost/lb
 B16: [W12] 45
 C16: [F3] '\$/lb
 A17: [W23] 'Reactor
 B17: [W12] 1933
 C17: [F3] 'MW thermal
 A18: [W23] 'Fraction pow.in pins
 B18: [W12] 0.974
 A19: [W23] 'Cycle length
 B19: [W12] 24
 C19: [F3] 'months
 A20: [W23] 'Capacity factor

APPENDIX C

B20: [W12] 0.75
 A21: [W23] 'Tails enrichment
 B21: [W12] 0.002
 C21: [F3] 'in wieght fraction
 A22: [W23] 'Value function tails
 B22: (F2) [W12] $(1-2*B21)*\text{ALN}((1-B21)/B21)$
 A23: [W23] 'Value function feed
 B23: (F2) [W12] $(1-2*0.00711)*\text{ALN}((1-0.00711)/0.00711)$
 A24: [W23] 'Cost of SWU
 B24: [W12] 55
 C24: [F3] '\$/kgSWU
 A25: [W23] 'Fab lead time
 B25: [W12] 1
 C25: [F3] 'yr
 A26: [W23] 'Enrich. lead time
 B26: [W12] 2
 C26: [F3] 'yr
 A27: [W23] 'Ore lead time
 B27: [W12] 3
 C27: [F3] 'yr
 A28: [W23] 'Fab time val Adj
 B28: [W12] $(1+B5)*B25$
 A29: [W23] 'Enrich time val adj
 B29: [W12] $(1+B5)*B26$
 A30: [W23] 'Ore Time val adj
 B30: [W12] $(1+B5)*B27$
 A31: [W23] 'U308 cost
 B31: [W12] 25
 C31: [F3] '\$/LB
 A32: [W23] 'Discharge Burnup
 B32: [W12] 45000
 C32: [F3] 'MWd/MTU
 A33: [W23] 'refueling periods
 B33: [W12] $+B6*12/B19-1$
 A34: [W23] 'int. over refu. per.
 B34: [W12] $(1+B5)*(B19/12)-1$
 A35: [W23] 'Present worth refueling
 B35: [W12] $((1+B34)*B33-1)/(B34*(1+B34)*B33)$
 A36: [W23] 'Contain. Cost
 B36: [W12] 25.225
 C36: [F3] '\$/FT³
 B38: [W12] 'AP-600 Total
 A39: [W23] ''
 A40: [W23] 'pin dia (mm)
 B40: [W12] $+B42*25.4$
 C40: $+C42*25.4$

APPENDIX C

A41: [W23] 'pitch (in)
 B41: [W12] 0.496
 C41: 0.50869
 A42: [W23] 'pin dia (in)
 B42: [W12] 0.374
 C42: 0.45
 A43: [W23] 'core vol (ft³)
 B43: (F3) [W12] 866.325
 C43: (F3) 284.548
 A44: [W23] 'contain size (ft³)
 B44: (S3) [W12] 1789000
 C44: (S3) 1241000
 A45: [W23] 'core+cont cost (\$)
 B45: (S3) [W12] ((+B43*\$B\$8)+(B44*\$B\$36))
 C45: (S3) ((+C43*\$B\$8)+(C44*\$B\$36))
 A47: [W23] 'delta P
 B47: [W12] 13.1416
 C47: 88.9074
 A48: [W23] 'delta P/ft
 B48: (F4) [W12] +B47*12/B54
 C48: (F4) +C47*12/C54
 A49: [W23] 'pump capital cost
 B49: (S4) [W12] +B47*\$B\$9*144*\$B\$10/(550*\$B\$11)
 C49: (S4) +C47*\$B\$9*144*\$B\$10/(550*\$B\$11)
 A50: [W23] 'pump operating cost
 B50: (S4) [W12] +B47*\$B\$9*144/(550*\$B\$11)*(0.746*8760*\$B\$12*\$B\$7)
 C50: (S4) +C47*\$B\$9*144/(550*\$B\$11)*(0.746*8760*\$B\$12*\$B\$7)
 A51: [W23] 'pump costs
 B51: (S4) [W12] +B50+B49
 C51: (S4) +C50+C49
 A54: [W23] 'length
 B54: [W12] 144
 C54: 99.374
 A55: [W23] 'H/D
 B55: [W12] 1.252
 C55: 1.252
 A56: [W23] 'DNBR
 B56: [W12] 3.258
 C56: 3.258
 A57: [W23] 'temp clad surface max
 B57: [W12] 683.7
 C57: 683.2
 A58: [W23] 'temp pellet surface max
 B58: [W12] 1187.6
 C58: 939.7
 A59: [W23] 'temp centerline max

APPENDIX C

B59: (W12) 2919.7
 C59: 1893.7
 A60: (W23) 'power W/g
 B60: (W12) 27.8053
 C60: (F2) 121.5805
 A61: (W23) 'linear power kw/ft
 B61: (W12) 4.098
 C61: (F2) 13.123
 A62: (W23) 'V avg ft/sec
 B62: (F2) (W12) 11.7535
 C62: (F2) 33.4528
 A63: (W23) 'V in
 B63: (F2) (W12) 11.0865
 C63: (F2) 33.4482
 A64: (W23) ' G
 B64: (S2) (W12) 1881000
 C64: (S2) 5675000
 A65: (W23) 'avg heat flux
 B65: (S2) (W12) 142800
 C65: (S2) 380200
 A69: (W23) 'no pins
 B69: (W12) 36280
 C69: 17324.34
 A70: (W23) 'fabrication cost max
 B70: (S2) (W12) +\$8\$13*B69*\$8\$28
 C70: (S2) +\$8\$13*C69*\$8\$28
 A71: (W23) 'amount of zircalloy
 B71: (S2) (W12) $(B42*B42/4 - (B42 - $8$14^2)^2/4)*\pi*854*0.23727*B69$
 C71: (S2) $(C42*C42/4 - (C42 - $8$14^2)^2/4)*\pi*C54*0.23727*C69$
 A72: (W23) 'amount of Zr fuel(lb)
 C72: (S3) +C82*\$8\$15*1000/0.454
 A73: (W23) 'total Zr
 C73: (S3) +C72+C71
 A74: (W23) 'cost of zircalloy
 B74: (S2) (W12) '*/
 C74: (S2) +C73*\$8\$16
 A75: (W23) 'cost of Zr/pin
 B75: (F2) (W12) +B74/B69
 C75: (F2) +C74/C69
 A76: (W23) 'min fab cost
 B76: (S2) (W12) +\$8\$70
 C76: (S3) +C75*C69*\$8\$28
 A77: (W23) 'nin +50
 B77: (S2) (W12) +\$8\$70
 C77: (S3) $(50 - C75)*C69*$8$28$
 A78: (W23) 'min +100

APPENDIX C

B78: (S2) (W12) +S8S70
 C78: (S3) (100-C75)*C69*S8S28
 A79: (W23) 'min +150
 B79: (S2) (W12) +S8S70
 C79: (S3) (150-C75)*C69*S8S28
 A80: (W23) 'min +200
 B80: (S2) (W12) +S8S70
 C80: (S3) (200-C75)*C69*S8S28
 A82: (W23) 'MTU total in core MTU
 B82: (F2) (W12) +S8S17*S8S18/860
 C82: (F2) +S8S17*S8S18/C60
 A83: (W23) 'core cycle burnup MWd/MTU
 B83: (F2) (W12) +S8S19*30.4*S8S17*S8S20/882
 C83: (F2) +S8S19*30.4*S8S17*S8S20/C82
 A84: (W23) 'initial enrichment %
 B84: (F3) (W12) (883/9000)*(897+1)/(2*B97)+1.5
 C84: (F3) (C83/9000)*(C97+1)/(2*C97)+1.5
 A85: (W23) 'initial enrich
 B85: (F3) (W12) +B84/100
 C85: (F3) +C84/100
 A86: (W23) 'Er feed/Er prod
 B86: (F2) (W12) (885-S8S21)/(0.00711-S8S21)
 C86: (F2) (C85-S8S21)/(0.00711-S8S21)
 A87: (W23) 'feed to enrichment (MTU)
 B87: (F2) (W12) +B86*B82
 C87: (F2) +C86*C82
 A88: (W23) 'value function
 B88: (F2) (W12) (1-2*B85)*ALN((1-B85)/B85)
 C88: (F2) (1-2*C85)*ALN((1-C85)/C85)
 A89: (W23) 'SWU/kg of product
 B89: (S2) (W12) (B88-S8S22)-B86*(S8S23-S8S22)
 C89: (S2) (C88-S8S22)-C86*(S8S23-S8S22)
 A90: (W23) 'SWU's
 B90: (S2) (W12) 1000*B82*B89
 C90: (S2) 1000*C82*C89
 A91: (W23) 'cost of enrich
 B91: (S2) (W12) +B90*(S8S24)*S8S29
 C91: (S2) +C90*S8S24*S8S29
 A92: (W23) 'mining cost
 B92: (S2) (W12) +B87*1000*S8S30*S8S31/0.848/0.454
 C92: (S2) +C87*1000*S8S30*S8S31/0.848/0.454
 A93: (W23) 'initial fuel cost
 B93: (S2) (W12) +B92+B91+B70
 C93: (S2) +C92+C91+C70
 A96: (W23) 'no of rods refueled
 B96: (F2) (W12) +B69*B83/S8S32

APPENDIX C

C96: (F2) +C69*C83/88832
 A97: (W23) 'no fuel loads
 B97: (F3) (W12) +869/896
 C97: (F3) +C69/C96
 A98: (W23) 'ref fab cost max
 B98: (S2) (W12) +870/897
 C98: (S2) +C70/C97
 A99: (W23) 'min
 B99: (S2) (W12) +88898
 C99: (S3) +C76/C97
 A100: (W23) 'min=50
 B100: (S2) (W12) +88898
 C100: (S3) +C77/C97
 A101: (W23) 'min=100
 B101: (S2) (W12) +88898
 C101: (S3) +C78/C97
 A102: (W23) 'min=150
 B102: (S2) (W12) +88898
 C102: (S3) +C79/C97
 A103: (W23) 'min=200
 B103: (S2) (W12) +88898
 C103: (S3) +C80/C97
 A105: (W23) 'amount zircalloy(lb)
 B105: (S2) (W12) +871/897
 C105: (S2) +C71/C97
 A106: (W23) 'cost of zircalloy
 B106: (S2) (W12) +8105*88816
 C106: (S2) +C105*88816
 A107: (W23) 'MTU in refuel
 B107: (W12) +882/897
 C107: +C82/C97
 A108: (W23) 'discharge burnup
 B108: (W12) +88832
 C108: +88832
 A109: (W23) 'refuel enrich (%)
 B109: (F3) (W12) (8108/9000)*(897+1)/(2*897)+1
 C109: (F3) (C108/9000)*(C97+1)/(2*C97)+1
 A110: (W23) 'refuel enrich
 B110: (F3) (W12) +8109/100
 C110: (F3) +C109/100
 A111: (W23) 'Er feed/Er prod
 B111: (F2) (W12) (8110-88821)/(0.00711-88821)
 C111: (F2) (C110-88821)/(0.00711-88821)
 A112: (W23) 'feed to enrichment (MTU)
 B112: (F2) (W12) +8111*8107
 C112: (F2) +C111*C107

APPENDIX C

```

A113: (W23) 'value function
B113: (F2) (W12)  $(1-2*B110)*BLN((1-B110)/B110)$ 
C113: (F2)  $(1-2*C110)*BLN((1-C110)/C110)$ 
A114: (W23) '8MJ/kg of product
B114: (S2) (W12)  $(B113-88522)-B111*(88523-88522)$ 
C114: (S2)  $(C113-88522)-C111*(88523-88522)$ 
A115: (W23) '8MJ's
B115: (S2) (W12)  $1000*B107*B114$ 
C115: (S2)  $1000*C107*C114$ 
A116: (W23) 'cost of enrich(24 mo)
B116: (S2) (W12)  $+B115*(88524)*88529$ 
C116: (S2)  $+C115*88524*88529$ 
A117: (W23) 'present worth enr
B117: (S3) (W12)  $+B116*88535$ 
C117: (S3)  $+C116*88535$ 
A118: (W23) 'mining cost(24 mo)
B118: (S2) (W12)  $+B112*1000*88530*88531/0.848/0.454$ 
C118: (S2)  $+C112*1000*88530*88531/0.848/0.454$ 
A119: (W23) 'present worth ore
B119: (S3) (W12)  $+B118*88535$ 
C119: (S3)  $+C118*88535$ 
A120: (W23) '24 mo refuel cost(max)
B120: (S2) (W12)  $+B118+B116*B98$ 
C120: (S2)  $+C118+C116*C98$ 
A121: (W23) 'present refuel value
B121: (S2) (W12)  $+88535*B120$ 
C121: (S2)  $+88535*C120$ 
A122: (W23) 'max refuel value
B122: (S2) (W12)  $+B121$ 
C122: (S2)  $+C121$ 
A124: (W23) '24 mo refuel cost(mid)
C124: (S3)  $+C102+C116+C118$ 
A125: (W23) 'present refuel value
C125: (S3)  $+88535*C124$ 
A126: (W23) 'mid refuel value
C126: (S3)  $+C125$ 
A128: (W23) '24 mo refu cost(min)
C128: (S3)  $+C99+C116+C118$ 
A129: (W23) 'present refuel value
C129: (S3)  $+88535*C128$ 
A130: (W23) 'adjusted refuel value
C130: (S3)  $+C129$ 
A132: (W23) 'total max
B132: (S3) (W12)  $+B45-B51-B70-B91-B92-B122$ 
C132: (S3)  $+C45+C51+C70+C91+C92+C122$ 
A133: (W23) 'total best guess

```

APPENDIX C

B133: (S3) (W12) +888132
C133: (S3) +C45+C51+C79+C91+C92+C124
A134: (W23) 'total min
B134: (S3) (W12) +888132
C134: (S3) +C45+C51+C76+C91+C92+C130

APPENDIX B

A STUDY OF METALLIC FUEL CYCLE IN PWR

(A Written Portion of Qualifying Exam)

Presented by

KWANSIK CHOI

Nuclear Engineering and Health Physics Program

School of Mechanical Engineering

Georgia Tech

CONTENTS

LIST OF FIGURES
LIST OF TABLES

1. INTRODUCTION
2. REACTOR DESCRIPTIONS
 - 2.1 Conventional PWR
 - 2.2 Advanced PWR
 - 2.3 Metal-Fueled PWR
 - 2.4 Integral Fast Reactor (IFR)
3. METAL FUEL
 - 3.1 Early Developments
 - 3.1.1 EBR-I
 - 3.1.2 LAMPRE
 - 3.1.3 FERMI
 - 3.1.4 DFR
 - 3.1.5 EBR-II
 - 3.2 Fuel Characteristics
 - 3.2.1 Irradiation Growth
 - 3.2.2 Swelling and Fission-Gas Release
 - 3.2.3 Fuel/Cladding Mechanical Interaction
 - 3.2.4 Fuel/Cladding Chemical Interaction
 - 3.2.5 Material Redistribution
 - 3.2.6 Corrosion Behavior
 - 3.3 Utilization in Thermal Reactors
 - 3.3.1 N-Reactor
 - 3.3.2 Magnox Reactor
 - 3.4 Recent Research Activities
 - 3.4.1 U-Pu-Zr Fuel Alloy in IFR
 - 3.4.2 U-Zr Fuel Alloy in PWR
4. FUEL CYCLES
 - 4.1 General Descriptions
 - 4.1.1 PWR Fuel Cycle
 - 4.1.2 IFR Fuel Cycle
 - 4.2 Front-End Processes
 - 4.2.1 Uranium Ore Concentrate Processing
 - 4.2.2 Oxide Fuel Fabrication
 - 4.2.3 Metal Fuel Fabrication
 - 4.2.3.1 Metal Production
 - 4.2.3.2 Advanced Enrichment Technique
 - 4.2.3.3 Liquid Metal-Bonded Fuel Elements
 - 4.2.3.4 Metallurgically Bonded Fuel Elements
 - 4.3 Reprocessing
 - 4.3.1 PUREX Process
 - 4.3.2 Pyrochemical Process
 - 4.4 Waste Management
 - 4.4.1 PUREX Wastes
 - 4.4.2 Pyroprocess Wastes
5. THESIS PROPOSAL

LIST OF FIGURES

	Page
Fig.2.1 Schematic Diagram of PWR System	5
Fig.2.2 Elevation View of PWR Reactor Vessel and Internals	7
Fig.2.3 Cross Section of PWR Reactor Vessel	7
Fig.2.4 Cross Section of Fuel Assembly (17 x 17)	8
Fig.2.5 Elevation View of PWR Fuel Assembly	9
Fig.2.6 Schematic Diagram of AP600	11
Fig.2.7 AP600 Reactor Coolant System	12
Fig.3.1 Cross Section of EBR-I	20
Fig.3.2 EBR-I Fuel Element Mark IV	20
Fig.3.3 LAMPRE Fuel Element	21
Fig.3.4 Cross Section of the ENRICO FERMI Reactor	22
Fig.3.5 FERMI Fuel Element	22
Fig.3.6 Cross Section of the Dounreay Fast Reactor (DFR)	24
Fig.3.7 DFR Fuel Element	24
Fig.3.8 Cross Section of EBR-II	26
Fig.3.9 EBR-II Fuel Assembly Mark I	26
Fig.3.10 EBR-II Driver Fuel Designs	27
Fig.3.11 Effect of Fuel Swelling on Fission-Gas Release in Metal Fuels	32
Fig.3.12 Comparision of Performance of LMR Structural Materials	34
Fig.3.13 Material Redistribution of U-Pu-Zr Fuel Irradiated in EBR-II	36
Fig.3.14 Burnup Trends of EBR-II Metallic Fuels	44
Fig.4.1 Nuclear Fuel Cycle for PWRs	50
Fig.4.2 Nuclear Fuel Cycle for IFR Concept	52

Fig.4.3	Present & Future Processes for Oxide Fuel Manufacture	55
Fig.4.4	Present & Future Processes for Metal Fuel Manufacture	56
Fig.4.5	A Route for Production of UF_4 from UO_3	58
Fig.4.6	Procedure of Injection Casting of Metal Fuel Pins	67
Fig.4.7	Composite Billet for Coextrusion of EBR-II Mark III Blanket Rod	67
Fig.4.8	PUREX Process for Oxide Fuel	70
Fig.4.9	Pyrochemical Process for IFR Core and Blanket	73
Fig.4.10	Electrorefining for Recovery of U & Pu	75
Fig.4.11	Halide Slagging for Recovery of Pu from Blanket U	76
Fig.4.12	Radiological Risk Factors of Fission Products and Actinides in PWR Spent Fuels	80
Fig.4.13	Treatment Methods of Solid Wastes from Pyroprocess	83
Fig.5.1	IFR Fuel Cycle Combined with Metal-Fueled PWR	86

LIST OF TABLES

	Page
Table 2.1 Worldwide Nuclear Power Units by Reactor Type	4
Table 2.2 Design Parameters of a Metal Fuel Core Compared to Those of AP600	13
Table 2.3 Typical Design Parameters of Metal Core in IFR	4
Table 3.1 Fast Reactors with Metal Fuel	18
Table 3.2 EBR-I Fuel Element Variants	20
Table 3.3 LAMPRE Fuel Element Design Parameters	21
Table 3.4 FERMI Fuel Element Design Parameters	22
Table 3.5 DFR Fuel Element Design Parameters	24
Table 3.6 DFR Fuel Element Variants	24
Table 3.7 EBR-II Fuel Element Mark I Design Parameters	26
Table 3.8 EBR-II Fuel Element Variants	26
Table 3.9 EBR-II : Development Progress from Mark Ia to Mark II Fuel Element	27
Table 3.10 Advanced Design of Metal Fuel	27
Table 3.11 Some characteristics of N-reactor and Magnox Reactors	41
Table 3.12 Corrosion Rates of Fuels in High Temperature Water	47
Table 4.1 Comparison of Process Waste Volumes in L/MWy(th)	79
Table 4.2 Solid Wastes from the Pyroprocess of IFR	82

1. INTRODUCTION

Most commercial Pressurized Water Reactors (PWRs), operated worldwide, employ uranium dioxide (UO_2) as their fuel material. However, other options have been tested throughout the nuclear era; oxide in Fast Breeder Reactor (FBR) and metal in FBRs. Recently, metal fuel in PWRs, the option abandoned at early days of the fuel development, is under reevaluation. This is due to some substantial progress in the research of metal fuel in EBR-II and FFTF. The metal fuel has passed high burnups of 15 at.% and is still successfully under irradiation in EBR-II. The metal fuel has demonstrated a closed fuel cycle under Integral Fast Reactor (IFR) Program. It has inherently safe features due to its material properties. Further, metal fuel matches with Atomic Vapor Laser Isotope Separation (AVLIS).

Since there is little experience with metal fuels in PWRs, their performance characteristics are mainly reviewed for U-Zr, U-Mo or U-Fs alloys in some FBRs and for U metal fuel, adjusted with some impurities, in some thermal reactors. In recent years, U-Pu-Zr alloys have been successfully irradiated and their data base could be applied to our design of metal fuel in PWRs. It could be beneficial to both the PWR and IFR fuel cycles if transuranics separated from the discharged metal fuels from PWRs would be loaded into the startup core inventory of IFR.

Progresses in the reactor designs of PWRs are reviewed with IFR

concepts in Ch.2. The historical developments of metal fueled fast reactors and related fuel performance are described in Ch.3. The selective processes involved in the metal fuel cycles are illustrated comparatively in Ch.4. Finally, The future work for Thesis is proposed in Ch.5.

2. REACTOR DESCRIPTIONS

The light water reactor design has been widely deployed around the world. There are 234 PWRs and 88 BWRs in operation worldwide, and 72 PWRs and 12 BWRs under construction or in order. About 65 percent of nuclear powered electricity is supplied only by PWR type and another 21 percent by BWR type (Table 2.1).[2a] Other reactor types have been tried with limited success.

After Three Mile Island-2 accident, advanced reactor programs were initiated to enhance safety features, mostly toward inherent or passive safety systems. Advanced PWRs, like AP600, are based on the successful experience of existing PWRs. IFR concept is also based on the remarkable progress of EBR-II technology and its related fuel cycle. Also, a metal-fueled PWR concept is extracted from both successes.

2.1 Conventional PWR

A schematic diagram of a PWR system is shown in Fig.2.1. The standard Pressurized Water Reactor [2b,2c] consists of mainly the nuclear steam supply system (NSSS) enclosed within a containment building and the secondary system outside of the containment building. The NSSS can be divided into the reactor vessel, pressurizer, and two or more primary coolant loops with reactor coolant pumps and steam generators. The secondary system can also be divided into the turbine, condenser, feedwater heaters

Table 2.1 Worldwide Nuclear Power Units by Reactor Type [2a]

Reactor Type	# Units (in operation)	Net MWe	# Units	Net MWe (total)
Pressurized light-water reactors (PWR)	234	200 890	306	268 190
Boiling light-water reactors (BWR)	88	71 715	100	83 512
Gas-cooled reactors, all varieties	39	13 493	39	13 493
Heavy-water reactors, all varieties	29	15 710	50	25 694
Graphite-moderated light-water reactors (LGR)	19	14 912	20	15 837
Liquid metal fast-breeder reactors (LMFBR)	4	1 178	10	5 202

Table 2.3 Typical Design Parameters of Metal Core in IFR

Fuel Materials	U-Pu-10% Zr, U-10% Zr
Fuel Smear Density	75%
Pin Diameter	7.6 mm (0.3 in.)
Cladding Thickness	0.46 mm (0.018 in.)
Peak Linear Power	50 kW/m (15 kW/ft)
Peak Discharge Burnup	150 MWd/kg

Fig.2.1 Schematic Diagram of PWR System [2b]

and feedwater pump.

Fig.2.2 and 2.3 illustrate elevation and cross section views of a PWR reactor vessel and internal structure, respectively. Water flows downward through the annular space between reactor vessel and thermal shield and then upward through the fuel assemblies. The core of a PWR is approximately cylindrical with a equivalent diameter of 12 ft. The core is made up of some 40,000 fuel rods combined in assemblies, each containing about 200 rods vertically arranged in a 16x16 or 17x17 square array roughly 8.5 in. on the side.

The control rods are combined in clusters of up to 24 rods in some assemblies. As shown in Fig.2.3, approximately one third of the fuel assemblies have control rod clusters distributed throughout the core. The control rods are inserted into the fuel assemblies by means of guide tubes. Fig.2.4 shows cross section of a typical PWR fuel assembly containing control rods. The control rod is a stainless steel (or Inconel) tube containing either compressed boron carbide (B_4C) powder or an alloy of silver, indium and cadmium as neutron absorber.

As a basic component of the core, the cylindrical fuel pellet is composed of slightly enriched uranium oxides. The pellets are typically 0.32 in. in diameter and about 0.6 in. in length. The pellets are stacked inside a Zircaloy (zirconium alloy with some impurities ; low neutron absorber) cladding with a wall thickness of 0.025 in. A fuel rod is also shown with the assembly in Fig.2.5.

Fig.2.2 Elevation View of PWR Reactor Vessel and Internals [2b]

Fig.2.3 Cross Section of PWR Reactor Vessel [2b]

Fig.2.4 Cross Section of Fuel Assembly (17 x 17) [2b]

Fig.2.5 Elevation View of PWR Fuel Assembly [2b]

The water serves as coolant, moderator and reflector, and is maintained under a pressure of 2250 psi by means of the pressurizer. When pressure in the primary system (NSSS) decreases, some water in the pressurizer flashes into steam to keep pressure constant. Conversely, when the pressure decreases, steam is condensed by the water spray from the reactor coolant inlet line to reduce pressure.

Modern PWRs have two to four independent steam generator loops in order to reduce the equipment size and to increase plant safety. The water passes through the primary loop at 2250 psi while water flows through the secondary at 1100 psi., where it is converted into steam at about 290 C. The steam is expanded in a turbine and then condensed back to liquid water which is recycled in the loop. Consequently, steam generators transfer heat from the primary coolant to secondary feedwater to generate steam for driving the turbine, eventually generating electricity.

2.2 Advanced PWR

Based on the collective understanding of the existent PWR plant safety and operation, advanced PWR development was initiated under the name of AP600 by Westinghouse Corp. This advanced design will probably be simpler, smaller in size, easier to license and construct due to its standardized design, and less costly.[2d]

In a conventional reactor, an emergency requires a series of operator actions such as opening or closing a valve, or turning on a pump etc. By contrast, AP600 system doesn't rely on any operator

Fig.2.6 Schematic Diagram of AP600 [2j]

Fig.2.7 AP600 Reactor Coolant System [2e]

Table 2.2 Design Parameters of a Metal Fuel Core Compared to Those of AP600 [2h]

	AP600	Metal Alloy Core
Fuel Material	UO ₂	U 50 w/o Zr
Pin Diameter (mm)	9.1	20.3
Pin Pitch (mm)	12.6	24.1
Active Fuel Length (m)	3.66	4.62
Number of Fuel Pins	38,280	16,681
Core Volume (liters)	24,525	49,300
Power Density (kW/liter)	79.4	39.4
Linear Power (kW/m)	12.6	22.9
Pressure Drop (kpa)	78.5	68.7
Pressure Drop/Meter (kpa/m)	21.4	14.9
Fuel Loading (MTU)	62.1	89.1
Power W/(gm-U)	28.4	19.8
Cycle Length (months)	18	18
Discharge Burnup (MWD/MTU)	45,000	45,000
Cycle Burnup (MWD/MTU)	11,979	8,349
No. of Pins Refueled/Cycle	10,190	3,095
No. of Fuel Batches in Core	3.8	5.4
Reload Enrichment (w/o U ²³⁵)	4.2	4.0

action but on phenomena, such as natural convection or gravity, initiated automatically in case of loss-of-coolant-accident (LOCA). Advanced PWR concepts employ passive safety features, while the conventional PWRs require its active safety features.

The AP600 will be a 600 MWe plant, configured similarly to a conventional PWR and have passive emergency core cooling and decay heat and containment cooling systems. Fig.2.6 shows a schematic diagram of an AP600. In the case of a LOCA, the core is flood with enough water to cool it for at least 3 days without any operator action.

The reactor coolant system employs sealed canned motor pumps, mounted in the inverted position and linked to the steam generator. This configuration eliminates a separate pump support (Fig.2.7). The low power density allows low enriched fuel core. Soluble boron and burnable absorbers are used for shutdown and reactivity control. The AP600 enhances plant availability by a longer fuel cycle of 18 months with a three region core. Also, a multi-port nozzle dam allows the inspection and maintenance of steam generator during reactor refueling. The canned motor coolant pump simplifies the auxiliary flow system and is very reliable. Also, the control system and instrumentation are improved by a unified design through distributed, digital microprocessor based technology.[2e]

2.3 Metal-Fueled PWR

A new design of a metal-fueled PWR employs most features of the AP600 in its core design, but the big difference from AP600

would be lower power density, which is about half that of AP600.

This new design employs U-50%Zr fuel, coextruded with Zircaloy cladding. Also, the fuel design allows 15% central void volume, providing accommodation of swelling due to the fuel irradiation. The fuel pin diameter would be about twice that of AP600, because of the low uranium content and the central void. The rest of design parameters is listed in Table 2.2.

In case of LOCA accident, the cladding temperature does not reach high enough to cause Zircaloy-water reaction, possibly eliminating the need for a high pressure emergency cooling system. This advantage comes from the small stored heat due to the high thermal conductivity of metal fuel. Like EBR-II, the overpower accident initiated by control rod withdrawal, could be less severe due to the smaller reactivity swing of metal fuel than oxide fuel. The lower pressure drop in the core allows the reactor to rely on the passive design, that is, natural circulation.

The metal fuel characteristics and its fuel cycle are reviewed in Ch.3 and 4, respectively.

2.4 Integral Fast Reactor

The need for a new advanced reactor, after the accident at TMI-2, crystallized the IFR concept, featuring 1) breeding capability, 2) passive or inherent safety, 3) fuel recycle and waste management and 4) proliferation issue. The IFR concepts have been developed at Argonne National Laboratory and are under demonstration of its closed fuel cycle.

The IFR is a self-contained facility in which several sodium-cooled fast reactors of moderate size are collocated with a fuel reprocessing facility. EBR-II is the natural prototype of the IFR and its core has been converted with the U-Pu-Zr alloy fuels. Table 2.3 shows the typical metal core design parameters in EBR-II.

In case of loss-of-flow-without-scam (LOFWS) accident, the reactor is automatically shut down without any safety-system or operator action because of the negative reactivity feedback initiated by the coolant temperature increase. In case of loss-of-heat-sink-without-scam (LOHSWS) event, the reactor is also shut down by the combination of the high conductivity of metal fuel and the thermal inertia of sodium pool. The transient overpower (TOP) accident is possibly reduced by the small reactivity swing of metal fuel.[2i]

Since sodium is noncorrosive to the structural materials, reactive corrosion products are not circulating and depositing around the system. Thus, the radiation exposure could be significantly reduced during maintenance, increasing its capacity factor. The IFR fuel cycle is discussed in Sec.4.1.2.

3. METAL FUEL

3.1 Early Developments

In early fifties, the development of civilian nuclear reactor was focused on metallic fuel and mostly for the fast breeder. In the beginning of the sixties, the fast reactor concepts were shifted to oxide fuel and then the thermal reactor fueled with the oxide was generated. In spite of the successful development of oxide fuels for fast breeders and thermal reactors in the past three decades, metallic fuel has been discussed due to the success in the EBR-II. Especially, since mid-eighties, the successful result of the EBR-II including the pyrometallurgical reprocessing on-site renewed the interest in the metallic fuel.

Table 3.1 summarizes the use of metallic fuel in fast breeders. [3a,3b] Following sections also refer [3a] and [3b].

3.1.1 The experimental breeder reactor EBR-I

The EBR-I was put in operation with a maximum power level of 1.2 MW(th) from 1951 to 1963. The objective of the EBR-I is to promote the understanding of fast breeder reactor. The core contained 7 fuel assemblies, 12 blanket assemblies and outer annular blanket. (Fig.3.1) The fuel assembly was in hexagonal shape with 60 pins. (Fig.3.2) The reactor was cooled with a sodium-potassium (Na-K) alloy and was fueled with four variants, Mark I to Mark IV. (Table 3.2) However, the same blanket was used with all

Table 3.1 Fast Reactors with Metal Fuel [3a]

	Location	Fuel Alloy	In Operation
EBR-I	Idaho USA	U-2% Zr Pu-1.75% Al	1951-63
LAMPRE	Los Alamos USA	Pu-10% Fe (> 410°C)	1956-64
DFR	Dounreay Scotland/UK	U-0.1% Cr U-9% Mo	1963-77
FERMI	Michigan USA	U-10% Mo	1966-72
EBR-II	Idaho USA	U-5% Fissium	since 1964

four variants. During the design progress, the fuel pin diameter got smaller and the fuel and cladding material was optimized.

3.1.2 The LAMPRE fuel

The so-called "Los Alamos Molten Plutonium Experiment LAMPRE" was operated with a maximum thermal output of 1 MW(th) from 1956 to 1964. The reactor was cooled with sodium and was fueled with Pu-10%Fe alloy. However, the molten Pu-Fe eutectic proved to be very corrosive so that a quite low burnup was achieved. This reactor concept was completely discontinued after testing a lot of fuel variants. Table 3.3 and Fig.3.3 describes some features of LAMPRE design.

3.1.3 The FERMI fast reactor

The Enrico Fermi reactor had mostly a thermal power level of 200 MW(th) and was put in operation from 1965 to 1972. The core was made up of more than 100 fuel assemblies and a large number of radial blanket elements.(Fig.3.4) The design feature of the assembly is the 12x12 square cross section.(Fig.3.5) The fuel pin diameter was maintained at 4.01 mm through several stages of development, but the zirconium cladding was originally 0.18 mm thick and coextruded with U-2%Zr fuel. Subsequent irradiation studies led to 0.13 mm thick zirconium cladding, metallurgically bonded with U-10%Mo fuel, which was selected as the reference fuel for the first core of the reactor. The small pin diameter was intended to achieve high specific power.

Table 3.2 EBR-I Fuel Element Variants [3a]

	Mark I	Mark II	Mark III	Mark IV
Fuel	pure U	U-2% Zr	U-2% Zr	Pu-1.75% Al
Bonding	NaK	NaK	metallurg.	NaK
Cladding	347	347	Zircaloy-2	Zircaloy-2
Pin Diameter (mm)	11.43	11.43	10.26	7.6
Fuel Length (mm)	190.5	216	216	215
Fuel Diameter (mm)	9.25, 9.75	9.75	9.25	5.9
Clad Thickness (mm)	0.51	0.51	0.51	0.53
Pins per Bundle			36	60
Longitudinal Fins	2	no	3	3
Linear Rod Power, aver. (W/cm)			= 180	427 110
Fuel Temperature (°C)				max. 495
Max. Burnup (% final)	0.35	0.1	0.3	0.1

Fig. 3.1 Cross Section of EBR-I [3a]

Fig. 3.2 EBR-I Fuel Element Mark IV [3a]

Table 3.3 LAMPRE Fuel Element Design Parameters [3a]

Reactor Power Fuel Coolant	1 MW _e Pu-Fe-Eutectic Sodium
Fuel Element Pin Diameter Fuel Length Fuel Diameter Cladding Clad Thickness Number of Fuel Elements	Rod type 10.9 mm 16.5 cm 8.57 mm Tantalum 0.69 mm 199
Fuel Composition Melting Point Operation Temperature	90% Pu, 10% Fe 410°C av. 637, max. 870°C
Specific Power, aver. Linear Rod Power, aver. Burnup	44 kW/kg Pu = 300 W/cm very low

Fig.3.3 LAMPRE Fuel Element [3a]

Table 3.4 FERMI Fuel Element Design Parameters [3a]

Reactor Power	200 MW _{th} , 60 MW _e
Fuel	U-Mo-Alloy
Coolant	Sodium
Number of Fuel Elements	105
Fuel Element	Pin bundle, square cross section
Pins per Bundle	140
Pin Diameter	4.01 mm
Fuel Length	77.5 cm
Fuel Diameter	3.76 mm
Fuel Composition	90% U, 10% Mo
U-235 Enrichment	25.6% U-235
Cladding	Zr
Clad Thickness	0.13 mm
Bonding Fuel/Clad	metallurgical
Linear Rod Power	= 150 W/cm average
Fuel Temperature	max. 600°C
Burnup	2% fima

Fig. 3.4 Cross Section of the
ENRICO FERMI Reactor
[3a] Fig. 3.5 FERMI Fuel Element
[3a]

Table 3.4 summarizes the main data of the FERMI fuel element [3a].

3.1.4 The Dounreay fast reactor DFR

The DFR was designed for a maximum rating of 72 MW(th) with 60 MW(th) in the core and 12 MW(th) in the blanket. The reactor was put in operation from 1963 to 1977 to draw the information needed to design and operate commercial power breeder reactor. The DFR had a quite complicated core arrangement. (Fig.3.6) The fuel elements were made up of thick single pins with a central hole in the fuel slug.(Fig.3.7)

Several innovative design features were included in this reactor:

- 1) Na-K was used as the coolant, which flowed downward, rather than upward, through the core and the blanket.
- 2) The fuel element was clad with the refractory metals, niobium and vanadium rather than stainless steel.
- 3) A vented design allowed the fission gas to escape from the fuel to the coolant.

Table 3.5 and 3.6 describe the fuel pin designs of four variants. During the design progress, the fuel element got a larger central hole and a maximum burnup of 3.6% was achieved in the Mark IV fuel pin.

Table 3.5 DFR Fuel Element Design Parameters [3a]

Reactor Power	72 MW _e , 15 MW _{th}
Fuel	U-alloys with Cr/Mo
Coolant	NaK 70/30
Number of Fuel Elements	345
Fuel Element	Hollow pin
Pin External Diameter	19 mm
Pin Internal Diameter	6.6-9.6 mm
Fuel Length	53.3 cm
Fuel Composition	U-0.1% Cr; U-9.1% Mo
U-235 Enrichment	45.5/75%
External Cladding	Niobium 0.5 mm
Internal Cladding	Vanadium 0.5 mm
Bonding Fuel/Clad	Na-bonding
Linear Rod Power	≈ 3000 W/cm average
Fuel Temperature	max. 650/840 °C
Burnup	up to 3.6% fima

Fig. 3.7 DFR Fuel Element

Fig. 3.6 Cross Section of the Dounreay Fast Reactor (DFR) [3a]

Table 3.6 DFR Fuel Element Variants [3a]

		Mark II	Mark IIa/b/c	Mark IIIa	Mark IIIb
Fuel Alloy	(w/o)	U-0.1 Cr	U-9.1 Cr	U-9.1 Mo	U-7 Mo
Enrichment	(w/o)	45.5	45.5	75	75
Pin External Diameter	(mm)	19	19	19	19
Pin Internal Diameter	(mm)	6.6	6.6	8.6	9.6
Gap Width external	(mm)	0.15	0.10-0.76	0.76	0.76
Gap Width internal	(mm)	0.25	0.25	0.07	0.07
Burnup achieved	(% fima)	0.1	0.57	3.6	3

3 of 4

3.1.5 The experimental breeder reactor EBR-II

The EBR-II was designed with a maximum power level of 62.5 MW(th) and put into operation in 1964 to demonstrate the engineering feasibility of a sodium-cooled FBR with a steam electric power plant and a pyrometallurgical reprocessing facility. Later, the mission of the reactor was switched over to that of a large fast test bed for the development of the in-plant fuel elements. The first fuel element design, Mark I is summarized in Table 3.7. The design details of the fuel pin variants is listed in Table 3.8. The reactor contains 47 fuel assemblies in the core surrounded by two zones of radial blankets. (Fig.3.8) The fuel assembly with the hexagonal cross section carries 91 fuel pins and upper and lower blanket pins. (Fig.3.9) A special feature of the EBR-II is the use of a metallic fuel alloyed with 5% F_s (Fissium), which is a equilibrium concentration of fission product elements left by the pyroprocess cycle.

At the transition from Mark Ia to Mark II, the gap width between fuel and cladding was increased, consequently reducing the smear density. Also, the fission gas plenum above the fuel was increased by a factor of 3. Fig.3.10 illustrates the dimensional comparison between two fuel designs. First, the increased gap width allowed the retained fission gas to escape from the fuel before the fuel-cladding interaction, eliminating the potential stress between fuel and cladding. Secondly, the increased gas plenum can accommodate the larger amounts of the released fission gas at an increased burnup. The improved design factors mentioned

Fig.3.10 EBR-II Driver Fuel Designs [3a]

Table 3.9 EBR-II : Development Progress from Mark Ia to Mark II Fuel Element [3a]

	Mark Ia	Mark II
Fuel Alloy Radial Gap Smeared Density Fission Gas Plenum Cladding	U-5 Fs 152 85 small 304 L	U-5 Fs 254 75 large 316
Changes during Burnup • Fuel Structure • Fuel Strength • Fission Gas Release • Fission Gas Induced Fuel Swelling	single pores remain separated no change small quite large	pores join to a network fuel becomes soft and yielding up to 80% small
Limitation of Fuel Life • Maximum Burnup Achieved (% fima) • Limitation by	3 FCMI*	10 FCMI, FCCI**, fission gas pressure
Limitation of Linear Rod Power • Max. Fuel Temperature because of Phase Transformation (°C)	ca. 600	ca. 600

* FCMI = fuel clad mechanical interaction

** FCCI = fuel clad chemical interaction

Table 3.10 Advanced Design of Metal Fuel [3a]

Fuel Alloy Zr as Alloying Component	U-2Pu-10 Zr reduces FCCI, raises melting point
Pu as Alloying Component	reduces fuel swelling
Cladding	austenitic or ferritic stainless steel
Burnup Goal	20% fima
Linear Rod Power	up to 480 watts/cm
Fuel Central Temperature	up to 750°C
Clad Temperature	up to 600°C

above enabled the Mark II fuels to achieve the burnups to 10 %
(Table 3.9)

Based on the research with the U-Fs alloy in EBR-II, a lot of different metallic fuel alloys have been studied for the development of an alloy with both a good high-temperature compatibility with stainless steel cladding and enough Pu content for breeding cycle. The alloys studied was U-Pu, U-Pu-Zr, U-Pu-Fs, U-Pu-Ti and U-Pu-Mo. As a result, the ternary alloys of U-Pu-Zr has been chosen to be most promising because of the excellent compatibility between the alloys and austenitic stainless steel.

Table 3.10 describes its design concepts. Further discussion will be delayed to sec.3.4.1.

3.2 Characteristics

As compared to the ceramic compounds, metallic fuels have a higher density as well as a higher thermal conductivity. The higher density allows us to design a smaller core to reduce the capital cost of a reactor. Also, the higher thermal conductivity causes the fission energy to be released faster from the fuel so that less energy is stored inside the fuel. Nevertheless, metal fuel has not been chosen as a fuel material except in some LMFBRs, because of some drawbacks. Metal fuels have comparatively low transition and melting temperatures. Moreover, metal fuels are potential activators so that they could be corroded severely by the invasion of high temperature water.[3c] By the use of alloying additives such as zirconium, niobium and molybdenum, the compatibility of metal fuel with water could be achieved and high melting temperature as well.

Irradiation induces microstructural change of uranium metal, which results in dimensional growth of the fuel. There are two major mechanisms for the fuel growth, a detrimental effect on the fuel performance. One is irradiation growth associated with the anisotropy of uranium crystals. The other is fuel swelling closely related with fission gas retention and solid fission products accumulation in the fuel. Also, some metal fuels such as the U-Pu-Zr alloy undergo material restructuring due to the gradient of the chemical potential between two separate phases.

Some features of metal fuels affecting its performance are discussed in next sections.

3.2.1 Irradiation growth

Under irradiation below about 500 C, single crystals of alpha uranium expand in the $\langle 010 \rangle$ direction and contract in the $\langle 100 \rangle$ direction. There is no noticeable expansion or contraction in the third direction. Therefore, polycrystalline uranium with some preferred orientation undergoes its shape change under irradiation. In addition to the irradiation growth, there is the problem of thermal cycle growth in the alpha uranium. The repeated thermal cycling of the alpha uranium at elevated temperature leads to grain coarsening and gross distortion called thermal cycling growth. The relationship between thermal cycling growth and irradiation growth is not well understood. However, both cause internal stresses, which, in turn, cause grain boundary cracking. The cracking has been a major problem in the irradiation of uranium fuels, especially in the lower temperature ranges. The internal stresses also cause the creep rate of uranium to increase in a manner that is proportional to the neutron flux.[3x]

In addition to the thermal cycling growth in the alpha region, thermal cycling through the alpha-beta transition temperature also causes volume increase in beta phase and surface distortion in alpha phase. Due to the dimensional instability mentioned above, the use of uranium metal as a fuel material is limited. In order to alleviate these anisotropic growths, the grains in the fuel

should be reduced in size and randomly oriented. This requirement can be achieved by quenching the metal from the beta-region or by alloying uranium with a considerable amount of other elements such as Mo, Nb, Pu or Zr. This alloying can stabilize gamma phase down to room temperature.

The adjusted uranium was an alternative to give resistance to irradiation growth. It was developed by adding small amount of alloying elements. However, such fuels have proved to grow too much at high burnups.

3.2.2 Swelling and Fission Gas Release

The gaseous fission products (Xe and Kr), consisting of about 25% of fission yield, tend to precipitate to form gas bubbles at the reactor operating temperature. There is a volume increase mainly due to the fission gas, as shown in Fig.3.11. The coarsening of gas bubbles causes enhanced swelling. The extent to which the bubbles migrate depends on the severity of the temperature gradient from the fuel surface to the center, especially on the thermal conductivity of the fuel. A high thermal conductivity of metal fuel and its alloys generates small temperature gradients, typically about 100 C/cm. On the other hand, its oxide fuel is very poor conductor and generates large temperature gradients, typically about 2000 C/cm. The low temperature and small temperature gradients in the metal fuel provide small driving force for bubble migration inside the matrix. Since bubbles tend to remain quite small, the fission gas can

Fig.3.11 Effect of Fuel Swelling on Fission-Gas Release in Metal Fuels [3p]

hardly be released from the slug until the volume increases by about 20% (Fig.3.11).[3p]

In the volume increase range between 20 and 30%, the porosity become interconnected among themselves and most of fission gas is released. Swelling and fission gas release are important effects determining the behavior of the fuel because these gases create a pressure in the plenum if released or create a stress on the cladding if retained in the matrix. Either the pressure or the stress could cause cladding deformation or failure.

Solid fission products (FPs) such as Mo, Ru, Zr, Pd and the lanthanide series accumulate with burnup as substitutional atoms or precipitates in the matrix. The FP accumulation contributes volume increase of about 2% per atomic percent burnup.[3q] At high burnups, the solid FP accumulation may surpass the volume increase due to both irradiation growth and swelling to retard the inter-connection of porosity.

3.2.3 Fuel/Cladding Mechanical Interaction (FCMI)

Stresses can be imposed on the cladding in two different ways; One is by released fission gas in the plenum and the other is by direct contact with the fuel slug. The direct contact can be caused by three major factors; 1) swelling due to gaseous and solid fission products, 2) volume increase due to phase transition, 3) differential thermal expansion between fuel and cladding.

The fission gas pressure can be reduced by providing a larger plenum and lowered cladding temperature. However, economic design

Fig.3.12 Comparison of Performance of LMR Structural Materials [3r]

favors compact pin lengths and high coolant outlet temperature. Also, the contact stresses can be reduced with a fuel design of low smear density (~ 75%). At the high burnup (>20%) expected in new metal fuels, the direct contact stresses may become more important because solid FPs accumulation causes volume increase, blocks the interconnected paths between pores and plenum, and hardens the fuel.

There are some efforts toward developing irradiation and temperature-resistant cladding. Performance difference among austenitic (S.S.316), Ti-stabilized austenitic (D9), ferritic (HT9) and oxide dispersion-strengthened ferritic (DSF) stainless steel has been estimated as shown in Fig.3.12. HT9 is chosen as an alloy for long-life advanced reactor design.[3e] DSF alloys are most promising structural material to achieve higher system burnups and higher operating temperatures.[3r]

3.2.4 Fuel/Cladding Chemical Interaction (FCCI)

At operating temperatures, fuel and cladding constituents interdiffuse. The interdiffusion has been observed as diffusion of Fe and Ni as cladding constituents into the fuel, with corresponding diffusion of lanthanide fission products (La,Ce,Nd, Sm,Pr) into the cladding. This FCCI results in a layer of cladding which contains nearly 20 wt % of the lanthanides. The lanthanides is known to harden and embrittle the cladding, deteriorating the mechanical integrity of the FCCI layer. Also, Fe and Ni, alloyed with the fuel, lower the fuel melting point. In

Fig.3.13 Material Redistribution of U-Pu-Zr Fuel Irradiated in EBR-II [3g]

off-normal condition, hence, the cladding can be penetrated rapidly if temperatures surpass the U-Fe eutectic, 715 C and a molten phase occurs.[3s]

3.2.5 Material Redistribution

In the irradiation tests of U-Pu-Zr alloy fuel [3t], for example, chemical redistribution involves the inward migration of zirconium and outward migration of uranium. However, the migration of plutonium is relatively small as shown in Fig.3.13. This material restructuring is related to the phase transition as the anticipated temperature in the middle ring of three concentric zones corresponds to the transition temperature of 600 to 650 C, where the phase changes from alpha to gamma uranium. Actually, interdiffusion of fuel constituents under irradiation is a complex phenomena due to the temperature gradients and the flux.[3g]

As the plutonium concentration of the alloy increases, the radial redistribution becomes more pronounced. Thus, fuel containing 8 wt % or less does not show noticeable material restructuring. It is known that radial restructuring does not limit the performance of metal fuels even though it leads to the lowering of fuel solidus temperature of the Zr-depleted zone.

3.2.6 Corrosion Behavior

Metal uranium fueled PWR will be discussed in this section. The consequences of cladding failure could be serious as uranium metal oxides rapidly in contact with hot water, the

coolant, by the reaction;

On the other hand, the oxide fuel oxidizes much more slowly. This was one of major reasons for metal fuels to be taken over by the oxides in the sixties. Some corrosion data will be compared between the metal alloy and the oxide in sec.3.4 (Table 3.13).

The oxide formed in the reaction is not adherent and powders come off continuously. In boiling water (100 C), the corrosion rate is of the order of 0.01 mils per hour while at 370 C the rate increases to about 6.0 mils per hour. Uranium also reacts with hydrogen to form uranium hydride which in turn reacts with water as follows;

Under conditions in which the corrosion product, hydrogen, can not readily escape, as would be the case inside a failed clad uranium fuel element, this accumulated pressure can increase the destruction rate of uranium.

The ability to withstand the aqueous corrosion is usually inhibited by irradiation. The corrosion resistance can be achieved by the gamma alloys like Mo, Nb or Zr or by the intermetallic compound like U_3Si . [3y]

3.3 Utilization in Thermal Reactors

Uranium (or Plutonium) alloys with considerable enrichment of uranium with U-233 or U-235 are widely used as fuel in research reactors. However, in PWRs, metal fuel has not been selected mainly due to fission gas swelling and metal-water reaction. Metal fuels have been tried in thermal reactors, which achieved low burnups. Table 3.11 lists, comparatively, some features of thermal reactors utilizing metal fuels.

3.3.1 N-Reactor

The N-reactor is of the graphite moderated, light water cooled, plutonium production type. Here, the design parameters and some experience, especially on the fuel system, are discussed for the N-reactor.[32]

N-reactor is fueled with a uranium-zirconium alloy, enriched to about 0.94 or 1.25% U-235, clad with Zircaloy-2. The fuel system uses a concentric tube-in-tube design. Each cylindrical fuel component is produced by the coextrusion of fuel and cladding. There are two components, an inner and an outer tube-shaped element, which are assembled in tube-in-tube geometry. To maintain the annular spacing and axial alignment of the assembly, support and locking clip are located between the inner and outer elements. Also, the reactor does not use burnable absorbers.

The metal uranium alloy fuel contains some impurities such as

aluminum, iron and carbon etc. It was found that operating temperature, exposure and alloying additions and physical restraints control fuel swelling in N-reactor. Also, the corrosion of uranium by water requires that the fuel be protected by high-integrity corrosion resistant cladding and that the reactor be shut down when a fuel failure is detected. For steady-state operations, the average fuel temperature are kept below the phase transition temperature of uranium (665 C). The maximum calculated transient cladding temperature is 280 C. Also, the tube-in tube design accommodates easily slight bowing of the pressure tubes due to the moderator contraction, The excellent heat transfer characteristics of metallurgical bond leads to adequate and uniform cooling. Three fuel enrichments are used in the loading; base (0.95%), radial (1.25%) and natural (0.7%). Inner element cladding strains are primary observed in the form of inner diameter buckling. The buckling appears to accommodate fuel swelling and reduce outer cladding strain. The fuels achieved average burnup in excess of 0.5 at. %.

3.3.2 Magnox Reactor

The Magnox reactors are the gas-cooled, graphite-moderated type developed in United Kingdom. The Magnox has used a uranium metal fuel at the natural enrichment level. The adjusted uranium with some impurities of iron and aluminum meets the dimensional stability required under low-burnup irradiation and thermal cycling. The adjusted uranium is heat treated to produce

Table 3.11 Some Characteristics of N-Reactor and Magnox Reactor

	N-Reactor	Magnox
Moderator	Graphite	Graphite
Coolant	Light Water	CO ₂
Fuel Material	Metallic Uranium (with 2% Zr max. & some impurities)	Metallic Uranium (with ~400ppm Fe & ~1000ppm Al)
Enrichment	0.94 and 1.25 %	natural
Cladding	By Coextrusion with Zircaloy-2	Magnox AL80 (Mg- Al- Be)
Fuel Design	Concentric Tube-in-tube	Extruded finned tube
Achieved Burnups	Avg. 0.5 at.%	Avg. 0.6 at.%

a fine grain structure, which alleviates the irradiation growth.

The fuel rods are cast in a metallic form, machined to diameter and length and then grooved. The cladding materials were made from Magnox AL80, an alloy containing fundamentally magnesium with 0.8 wt.% aluminum and 0.005 wt.% beryllium. This alloy is corrosion-resistant up to 500 C in the coolant (CO_2). The grooved uranium rod is clad with the Magnox can, an extruded finned tube. The endcaps are welded into the ends of the can. The fins provide the coolant flow with turbulence and extended surface area for fast heat removal.

Thermal cycling through alpha-beta transition causes dimensional changes and growth with some cracking in the fuel. Thus, the fuel temperature should be kept below 665 C. The fuel pin failure was mostly caused by cracking between large grains in the can. The problem of grain growth appears to be overcome by the careful attention to the grain structures in as-fabricated can. The Magnox fuels achieved an average burnup in excess of 0.6 at.%.

3.4 Current Research Activities

The successful experience with U-5%Fs as EBR-II fuel is now being applied to the development of a U-Pu-Zr fuel for an advanced fast breeder. Also, the renaissance of metal fuel, based on the successful results of the in-pile tests of the U(-Pu)-Zr fuel in EBR-II, draws the feasibility study of its utilization in PWRs.

3.4.1 The U-Pu-Zr Fuel Alloy in IFR

There have been some technical limitations associated with U-5%Fs in an advanced fast breeder; [3d]

- 1) the radial expansion of fuel into the sodium bond; a porous, cellular, gas-percolating structure of fuel.

- 2) solid-state interdiffusion of iron and nickel from S.S.316 into the fuel and of some fission products from the fuel into the cladding; a brittle ferritic layer in the cladding and a macroscopic peripheral zone in fuel, growing with burnup.

The Pu-containing Fs alloys exhibited higher interpenetration rates than the Pu-free alloys. This poor compatibility and the recent development of the electrorefining process drew upon the U-Pu-Fs alloy back from its front-line. Thus, the U-Pu-Zr ternary system became most promising for the integral fast reactor concepts. The Zr is reported [3d] to provide an alpha phase barrier against the interdiffusion, which is a characteristic phenomenon in the U-Fs alloy. In addition, the U-Pu-Zr alloys

Fig.3.14 Burnup Trends of EBR-II Metallic Fuels [21]

demonstrated its potential of high burnup through the recent irradiation tests.[21]

Fig.3.14 shows the improvements in fuel burnup through the years, including the U-Fs alloy before 1988 and more recently the IFR alloy. The U-Pu-Zr alloy is still under irradiation, accumulating its data base of burnup. It expected to achieve the burnup as high as about 15 at.%.

3.4.2 The U-Zr fuel alloy in PWRs

As stated earlier, the metal fuel becomes attractive to be used in PWRs because there are great advantages compared to the oxide fuel in the safety and cost point of view:[2g]

- 1) lower stored heat due to the lower thermal conductivity.
- 2) lower pressure drop due to larger coolant channel.
- 3) lower fuel cycle cost due to the absence of the conversion step to UO_2 and the replacement of the compacting and sintering process by the injection casting process and the potential use of the pyrochemical techniques.

However, there are some drawbacks associated with the use of the metal fuel in PWRs.

- 1) lower melting temperature compared to the oxide.
- 2) dimensional instability associated with phase transition and fuel swelling.
- 3) potential reaction between the uranium alloy and water.

Based on the experience of N Reactor, as described in sec.3.3.1, the adequacy of the use of metal fuel, especially U-Zr

alloy, in PWRs is reviewed under the contract with DOE, including the study of the metal-water reaction. The possibility of fuel failure should be inhibited under its operation since the failure may cause the oxidation of uranium and subsequent weight loss of uranium powder into the coolant. To deal with this problem, the corrosion resistance of uranium alloy fuel must be improved or/and a high integrity of corrosion resistant cladding should be assured as a first barrier for the potential reaction. Table 3.13 shows the comparative corrosion behavior in the various Zr-content U-Zr alloys and the oxide, concluding that the U-50%Zr could be acceptable as a future metal fuel for PWRs.

Another limitation against high burnup is swelling. This can be eliminated by introducing about 15 vol.% central void in the fuel pin. This remedy is implied in the evolved design of Mark-II irradiated in EBR-II.

The annular fuel coextruded with the zircaloy cladding is recommended after some of the other fuel design concepts are reviewed; The sodium-bonded fuel has been investigated to conclude that the hydrogen generated by the sodium-water reaction may cause disintegration of the cladding by hydrating. Other molten liquid metal such as lead, bismuth, tin or zinc are also looked into, in vain because all of them react readily with uranium metal at the in-reactor temperature. The active mechanical contact concept has been examined. Since the fuel circumference might become coarse due to fission products, bad mechanical contact would occur between the fuel and cladding after some irradiation. The sphere-pac

Table 3.12 Corrosion Rates of Fuels in High Temperature Water

Tested Fuel	Temp. (F)	Sample Shape & Condition	Corrosion Rate (mg/cm ² hr)	Ref.
Pure U	587	Pin /Unclad	6260	BMI-1156
U- 5w% Zr	587	"	3150	"
U-10w% Zr	595	"	2780	"
U-15w% Zr	604	Pin /Unbonded	165	"
U-20w% Zr	610	"	194	"
U-50w% Zr	600	Unclad	0.025	BMI-874
"	600	"	0.1	WAPD-T-416
UO ₂	650	Pin/Clad	0.00135	WAPD-184

concept has been also studied as an alternative fuel design. It is a promising fuel concept for both LWRs and LMFBRs and still under development. Under this concept, the Zircaloy is filled with a two- or three-sized fraction mixture of U-Zr alloy microsphere and pressurized with helium.

During irradiation, the alloy acts as a semi-viscous fluid even under light loads. Therefore, for cladding strain analysis, we assume that the alloy swelling effects create a hydraulic-like pressure on the cladding. Since the swelling of fuel is controlled by alloying additions, physical restraints, operating temperature and exposure, the tensile strength resulting from the fuel swelling can be estimated and then cladding thickness can be calculated with the supply of its mechanical properties.

4. Fuel Cycles

4.1 General Descriptions

4.1.1 PWR Fuel Cycles

As Fig.4.1 shows, the enriched uranium fuel cycle begins when the uranium ore (U_3O_8), so-called yellow cake, is mined. Natural uranium ore contains 1 part of the fissile U-235 in 140 parts of the fertile U-238. The uranium ore is then enriched by taking advantage of the preferential diffusion rate for U-235 as compared to the heavier U-238. However, uranium must be first converted into a gaseous form of UF_6 , prior to the enrichment. The UF_6 is then pumped through the gaseous diffusion barriers, yielding one UF_6 stream that is rich in the U-235 concentration and the other UF_6 stream that is depleted in U-235. This enriched UF_6 is then converted to UO_2 and pressed into pellets; a stable solid that, when contained in Zircaloy cladding, is ready for use as a reactor fuel. When irradiated, the fuel undergoes a chain reaction which produces heat and neutrons. After the fuel has been burned out for about 3 years, the fissile concentration is sufficiently depleted to require refueling. The spent fuel, after the discharge from the reactor, requires about cooling period of at least 3 months so that the short-lived radionuclides decay out enough for fuel handling. The spent fuel is then transported in a shielded cask to a reprocessing plant or extended storage facility.

In the reprocessing plant, U, Pu and fission products are

Fig.4.1 Nuclear Fuel Cycle for PWRs

chemically separated, the depleted uranium being recycled as feed to the enrichment plant, the plutonium being recycled in LWRs or reused as fuel in FBRs, and the fission products being converted into a waste form. The high level waste is stored on-site and then transported to a geological repository for disposal. Under once-through option, the spent fuel, after being stored and conditioned, would be disposed of in same manner as the high level waste.

4.1.2 IFR Fuel Cycle with PWR

In PWRs, even with recycled Pu, less than 1% of natural uranium in the ore is burned up, while all uranium could be fissioned for generation of power through the breeding and recycling process in the fast breeders. This recycling of the fissile material will save the uranium resources significantly, extending them by a factor of hundred. However, an once-through fast breeder would not be economical because it does not breed and moreover requires a significant amount of initial loading of fissile materials (U-235 and Pu) for its core and the fertile material (U-238) for its blanket. Thus, the fast breeder can breed only in a closed fuel cycle, with adequate removal of fission products as high level waste and replacement of the Pu consumed through fission in core by the Pu generated by the neutron capture of U-238 in the blanket.

The Integral Fast Reactor (IFR) is a prototype of fast breeder, under development as a future reactor concept at Argonne National

Fig.4.2 Nuclear Fuel Cycle for IFR Concept [4m]

Laboratory. The IFR could be used as either breeder or actinide-burner. In both option, 2.8 tons of PWR spent fuel transuranics are required for the startup inventory of a 450 MWe IFR.[4a] In breeder option, the transuranic inventory is stored in the fuel cycle for the plant lifetime. However, in actinide-burner option, the transuranics are consumed in the reactor, with its makeup from LWRs. In this report, IFR fuel cycle model is discussed on the basis of its breeder option. To initiate a IFR, 17 yearly discharges of 1000 MWe LWRs need to be supplied.

The significant amount of initial loadings should be supplied by external sources, usually through the extraction of fissionable materials from the LWR spent oxide fuel. In current design of the IFR, the U-Pu-Zr alloy is utilized as its metal fuel. Actinides are extracted in a metallic form from metal fuels and adjusted with some alloying elements of depleted U and Zr to the desired composition of the U-Pu-Zr fuel. This adjusted melt is injection-cast into a slug and, with sodium bonding in the cladding, recycled back into the reactor. (Fig.4.2)

4.2 Fuel Fabrication

The manufacture process starts with uranium ore concentrate whether it is for oxide or metal fuel production. The chemical processes to fabricate the fuel is described. The processes up to uranium tetrafluoride (UF_4) preparation are common to both oxide and metal fuel production.

The development of AVLIS enrichment technique altered the route of both oxide and metal fuel manufacture. For the oxide, UF_4 is converted to uranium hexafluoride (UF_6), which is then concentrated to the reactor-grade UF_6 in a gaseous phase. The enriched UF_6 is transformed to nuclear grade uranium dioxide (UO_2) powder, which is the feed for the UO_2 pellet production. Under the future option of the oxide fuel fabrication, the enriched metal from the AVLIS could be converted to the UO_2 powder.

For the metal, the UF_6 enriched from cascade is reduced to UF_4 , which could be used for enriched metal production. Under the future option, the UF_4 is reduced to metal, which could be fed into a AVLIS plant. The enriched metal could be directly manufactured into a fuel slug. These differences are shown comparatively in Fig.4.3 and 4.4.

4.2.1 Uranium Ore Concentrate Processing

Uranium ore concentrate (UOC) is dissolved in nitric acid to produce a slurry containing impure uranyl nitrate and

Fig. 4.3 Present & Future Front-End Fuel Cycles for Oxide Fueled PWRs

Fig.4.4 Present & Future Front-End Fuel Cycles for Metal-Fueled PWRs

extraction using 30 vol.% tributyl phosphate (TBP) in kerosene as solvent and mixer-settler contactors. The system has been designed to eliminate elements with high neutron capture cross sections (boron and cadmium) and elements with volatile fluorides (chromium, molybdenum, tungsten and vanadium). The pure uranyl nitrate solution contains uranium in 110 g/L.

The pure uranyl nitrate is evaporated to yield a molten concentrate containing uranium in 1100 g/L. Thermal decomposition of the concentrate is carried out in one of several fluidized bed reactors containing a bed of uranium trioxide which is fluidized with air and maintained at a temperature of 300 - 350 C by electrical heating. Molten uranyl nitrate concentrate is pumped into the fluidized bed where it decomposes into uranium trioxide, nitrogen oxides, oxygen and steam. The uranium trioxide overflows continuously from the denitrator and transported pneumatically into storage hoppers.

At Springfields, UK, there was a need to expand UF_6 production capacity. However, the expansion of existing technology of fluidized bed reduction and hydrofluorination did not satisfy its safety and economy. Thus, rotary kiln, which is a continuous process based on countercurrent gas-solid contact in a low pressure system, was developed to meet the requirements. (Fig.4.5) [4c]

Fig.4.5 A Route for Production of UF₄ from UO₃ [4c]

related to the production of UF_6 ; (1) Hydration of the trioxide (2) Reduction of the hydrate to uranium dioxide and (3) Hydrofluorination of uranium dioxide to uranium hexafluoride.

(1) Hydration of uranium trioxide is achieved by adding dilute nitric acid and a wetting agent to the uranium trioxide in a stainless steel trough. The slurry so produced is gradually dried by steam heated jacket of the hydrator. The resulting hydrate in a dry powder form, discharges into the hopper.

(2) The hydrate is conveyed into the kiln barrel by an open helical screw in the base of hopper and transported through the kiln by a system of flights and dam rings. The exothermic reduction takes place at about 480 C. Hydrogen is flowed into the end of the kiln and reacts with the uranium trioxide hydrate countercurrently to produce uranium dioxide.

(3) As the uranium dioxide passes through the hydrofluorination kiln, it reacts with anhydrous HF gas countercurrently. The kiln adopts forced air cooling to keep the required temperature profile from the strongly exothermic reaction.

The uranium tetrafluoride, typically over 98%, is discharged into storage hoppers and is conveyed to the hexafluoride manufacturing plant.

4.2.2 Oxide Fuel Fabrication

4.2.2.1 Uranium Dioxide production

The UF_6 reacts with fluorine in the fluidized bed reactor containing an inert bed of calcium fluoride to produce uranium

hexafluoride (UF_6). The reaction is highly exothermic and the calcium fluoride prevents sintering of the tetrafluoride. The bed is also maintained at 450 - 475 C by forced air cooling. The UF_6 , in gaseous form, can be necessarily enriched in U-235 in an isotope separation plant.

There are three options of conversion process; integrated dry route (IDR), ammonium diuranate (ADU) process and ammonium uranyl carbonate (AUC) process. In the IDR, the UF_6 is converted to very high purity uranium dioxide powder. It reacts with steam and hydrogen in a single rotary kiln. First, the UF_6 vapor reacts with steam to produce the intermediate UO_2F_2 , which passes through the kiln. The UO_2F_2 is reduced to UO_2 by the countercurrent stream of steam and hydrogen. The temperature is controlled in the kiln by number of zone heaters, to achieve the requisite quality. After the quality checks, the ceramic grade powder is homogenized using an orbital screw blender.

ADU process consists of the volatilization of UF_6 in a steam chest and mixing with water to form hydrofluoric acid and uranyl fluoride solution. The solution is transported to a precipitation column where it reacts with ammonium hydroxide causing the uranium to precipitate as fluoride bearing ammonium diuranate. The ammonium diuranate is then calcined in air to produce U_3O_8 , where it is treated with hydrogen to prepare ceramic grade uranium dioxide powder.

In the AUC process, the UF_6 along with CO_2 and NH_3 gas is injected into demineralized water to produce a crystalline

precipitate of ammonium uranyl carbonate. The crystals are filtered, washed with an ammonium carbonate solution and then with methanol. It reacts with steam and cracked ammonia in a fluidized bed, to produce the uranium dioxide.

4.2.2.2 Oxide Pellet and Pin Production

The high purity uranium dioxide powder is used to fabricate the UO_2 fuel pellets. For the requirement of high density pellet, the particle size of powder is reduced, mixed with solvent and organic binder, and spray dried to form a feed about 120 micro meter in diameter. After passing to the desired shape, the binder is removed at 800 C and then the pellets are sintered in hydrogen for 3-4 hrs at 1700 C. They are ground to the required tolerance using silicon carbide wheels on a centerless grinder and after washing and drying, the designed stack of pellets are assembled for fuel pin manufacture.

With the IDR conversion process used, the pellets could have the density greater than 98% TD without reducing the particle size. As a result of the trend toward higher burnups, longer operating cycles and more efficient fuel management, the burnable absorbers combined to the fuel has been proposed; incorporation of gadolinia into the pellets or coating of pellets with a thin layer of zirconium diborate.

The pellets are loaded into Zircaloy tubes. Spring or clip is inserted and then Zircaloy plugs are welded at each ends.

4.2.3 Metal Fuel Fabrication

4.2.3.1 Production of Uranium Metal

Though uranium metal has been produced since early fifties, the metal production activity has continued to decline to the present level. Uranium tetrafluoride (UF_4), so-called green salt, is an intermediate common to the production of both uranium metal and uranium oxide fuel, as shown in Fig.4.3 and Fig.4.4. However, higher grade of UF_4 is required for metal production (i.e. 96% to 98+% UF_4 content) in order to maintain acceptable yields in a reduction process;

Metal is produced by heating a mixture of UF_4 and magnesium turnings in a reduction reactor until the reaction occurs, producing molten uranium and magnesium fluoride slag which separate by gravity. The uranium metal production technology has been improved by the increased reactor batch size, the recycle of slag for liner use and improvements in firing cycle.[4d] Calcium could replace magnesium as the reductant with some increase in yield, but would probably be more costly.

In the production of metal fuel, the continuous casting process may be promising if the AVLIS process should be chosen for the enrichment of uranium.[4d]

4.2.3.2 Advanced Enrichment Technique

Worldwide attention has been focused on the atomic vapor laser isotope separation (AVLIS) process because of its potential to provide enrichment service at one-half the cost of existing enrichment technique like gaseous diffusion or gas centrifuge.

In the AVLIS process, the introduction of feed and withdrawal of product occurs in metal form rather than UF_6 . If the UF_6 production step could be skipped in the fuel cycle, about 15% of the enrichment cost of UF_6 based AVLIS process could be saved.[4d] An AVLIS facility would contain a material conversion section to prepare feed material for AVLIS process, the AVLIS instrumentation itself and a section for converting product materials to a form suitable for shipping, storage or conversion to a final product form. In principle [4e], the material to be enriched is vaporized and the vapor phase is then irradiated by tuned laser light. Then, the ionized isotope is collected by means of electric and magnetic fields. For example, at an AVLIS facility, the laser will be tuned to a U-235 resonance to produce U-235 ions that can be separated from more numerous, electrically neutral U-238 materials in the vapor. Optical methods are under development for the on-line monitoring the actual AVLIS process. Other recent researches include direct electrolytic reduction from yellow cake to metal for the AVLIS feed and solvent extraction purification of the metal product.

4.2.3.3 Fuel Element Production

With the respective recipe of fuel and cladding materials and their geometry, the type of bond between the fuel and cladding is chosen by considering the materials and their thermal properties. The principal types are: (1) mechanical bond, (2) liquid or gas bond and (3) metallurgical bond.

One of important roles of the bond is heat-transfer from fuel to cladding, eventually to the coolant. It may be accomplished by simple mechanical contact of fuel and cladding. The good thermal contact can be effected by filling a conducting gas or liquid in the gap. Conventionally, oxide fuels usually utilize helium gas as the bond because of its high thermal conductivity. However, it is known that liquid is better conductor than gas. Thus, the liquid sodium bonded fuels have been employed for the fast breeder. Moreover, some liquid metals such as lead-bismuth and lithium are being considered as the bonding material even for light water reactors.[4f]

The liquid metal bond is most effective to minimize the temperature drop between fuel and cladding. The bond space can be available to accommodate the potential fuel swelling. Thus, the slug can be removed from the cladding to be reprocessed. Also, it is most economical because it allows unmachined fuel slugs. On the other hand, it requires a precise bonding technique to exclude the acentric fuel slug in cladding and to assure the uniform wetting of the bond.

The metallurgical bond is advantageous in reducing the chance of fuel corrosion by the coolant due to the cladding breach. Irradiation could cause the destruction of the bond due to the formation of brittle transition zone or differences in thermal expansion. Therefore, the fuel and cladding materials should be compatible so that there are not many material combinations. The metallurgically bonded fuel elements are fabricated by coextrusion. The coextrusion process is most economically applied to long fuel elements involving a minimum number of end closures.

Liquid Metal-Bonded Fuel Elements The fuel element slug and cladding are separately fabricated, assembled and then bonded by liquid sodium. For the fabrication of metal fuel pins, injection casting process is favored due to its simplicity for remote operation and its high yield. The procedure of the injection casting are described schematically in Fig.4.6. The alloy is induction melted under a vacuum in a refractory-coated graphite crucible. The assembly of high-silica glass tubes is suspended with open end down, and electrically heated to accept the molten metal. At the casting temperature of the molten alloy, the open ends of the mold tubes are submerged into the crucible. As the vacuum chamber is switched to a pressurized helium tank, the molten alloy is injected into the molds by pressurizing the casting furnace. After the solidification of the cast metals, the glass molds are crushed away from the cast metals, which are cleaned afterwards and cut to length.

There have been some improvements involved in the injection casting process over 30 years.[4g];

- 1) Accurate control of the casting temperature, reducing impurity level, internal defects in the slugs and surface roughness.
- 2) Chemistry control, achieving one-step melting process.
- 3) homogeneity of the melt, lowering impurity levels.

Under inert atmosphere, the cladding is carefully filled with the measured amount of sodium and the uranium alloy slugs are inserted into it. The cladding containing the fuel and sodium is then heated up to melt the sodium and move the slug down, raising the sodium level above the top of the fuel slug. Finally, the end cap is welded on to the top of the cladding.

Metallurgically Bonded Fuel Elements A metallurgically bonded fuel element would give the rigidity required for the Mark III rods in EBR-II [4h]. Based on the compatibility between the fuel and cladding, a U-2 wt% Zr alloy fuel clad with Zircaloy was prescribed. The extrusion cladding was chosen because it could effect a metallurgical bond between the fuel and cladding. Fig.4.7 shows the extrusion billet and its constituent materials used for the Mark III blanket rod. The Zircaloy cladding was cut from pre-extruded Zircaloy tube, bright-etched in nitric hydrofluoric acid solution and completely rinsed and dried. Separately, the U-Zr alloy was a heat-treated casting, abraded with wet sand paper, degreased in carbon tetrachloride, rinsed in acetone and air dried. The Cu-10% Ni alloy used for internal nose plug and cut-off

Number of Castings at Single Operation - 25
 Diameter of Casting - 14.22 inches
 Length of Casting - 15 to 17 inches
 Length - to - Diameter Ratio - >100:1

Fig.4.6 Procedure of Injection Casting of Metal Fuel Pins [4h]

Fig.4.7 Composite Billet for Coextrusion of EBR-II Mark III Blanket Rod [4h]

Specifications			
Component	Material	Component	Material
Core	U-2 w/o Zr	Front end plug	16 gage Cu sheet punched and cupped
Cladding	Zircaloy-2	Rear end plug	Deoxidized Cu plate
Internal nose plug	Cu-10 w/o Ni	Evacuation tube	Seamless hard drawn Cu tube
Internal cut-off	Cu-10 w/o Ni	Extrusion cone	Mild steel
Container	Seamless hard drawn Cu tube	Extrusion die	18-4-1 steel

components were conventionally fabricated as the copper containers were.

After assembled into the container, the billet was evacuated to 5×10^{-5} mmHg or less, and heated to 430 C in a half hour and held at the temperature for an additional hour. Then, the billets were sealed off. The extrusion processed at 663 C for a reduction of area of 20 to 1. The copper container was pickled out using nitric acid and then cold-swaged to the desired size. The heat treatment consists of an isothermal quench from 780 to 500 C and subsequent air cooling. The heat treatment produced the more uniform interface between the fuel and cladding.

4.3 Reprocessing

At the time of discharge, the fuel comprises mostly uranium (some 96 wt.% but less enriched, 0.8 wt.% in U-235) and plutonium (about 1 wt.%), both of which are reusable. The remainder (about 3 wt%) consists of highly radioactive fission products and minor actinides (Np-237, Am-241/243 and Cm-242/244).

In order to separate U and Pu from the fission products in spent fuels, oxide or metal, there are two processes; PUREX process and pyrochemical process. Pyrochemical process is highly advantageous on reprocessing of metal fuels, while PUREX process is on that of oxide fuels. However, either process could be employed for either type of the fuels. PUREX process operates below 100 C. while pyroprocess works at higher temperatures.(around 500 C). Former produces large volume of wastes while latter produces small.

4.3.1 PUREX process

PUREX process is defined as aqueous process with solvent extraction. Though several wet processes has been tested historically, the PUREX process has displaced all others at the present. The process consists of several steps such as primary decontamination, plutonium partitioning, uranium purification and aqueous waste processing. These steps [4b] are shown schematically in Fig.4.8.

Fig.4.8 PUREX Process for Oxide Fuel [2c]

After the dissolution process, there is an aqueous solution of uranyl nitrate, plutonium(IV) nitrate, nitric acid, nitrate compounds of neptunium, americium, curium and non-volatile fission products. The non-volatile F.P. are separated as much as 99 to 99.9% from the uranium and plutonium in early stage of the process so that its removal simplifies shielding, maintenance and protects solvent degradation from radiolysis. The decon system consists of extracting section and scrubbing section. In extraction section, U and Pu are extracted from the dissolver solution with 30 vol.% TBP in kerosene. Am and Cm are predominantly trivalent, have low distribution coefficients like the rare-earth F.P.s and leave the extraction section in the aqueous raffinate. Np in hexavalent state is partly extracted into organic extract and Np in pentavalent state remains in the aqueous. Centrifugal contactors or pulse columns reveal better protection from solvent degradation than the mixer-settlers because the former has low holdup in the extraction section.

After the primary decon, the U and Pu are extracted into a organic solvent(TBP), Pu is reduced to the trivalent state, in which it is inextractable by TBP, while U is left in the extractable hexavalent condition. Under the existing technology, Pu is reduced chemically, which has the disadvantage that chemicals to be added increase the waste volume significantly. Thus, electrochemical reduction methods were developed and demonstrated at KfK, Germany. [41]

Uranium leaving the partitioning step in the organic phase is extracted back into the aqueous phase by nitric acid. Then, it can be purified by one or more additional cycles of solvent extraction by TBP, while Pu can be kept in the extractable trivalent state.

4.3.2 Pyrochemical Process

Pyrochemical Process is defined as a dry process with electrotransport of transuranics.[4j] The pyrochemical process consists of two chemical processing steps; electrorefining and halide slagging. Both process steps are combined to effect 1) fission product removal from the core element and 2) plutonium transfer from the blanket to the core element. (Fig.4.9) Basically, the electrorefining step is used to concentrate U & Pu from the core element (Fig.4.10). The halide slagging step is intended to recover Pu from the blanket element. (Fig.4.11)

In the electrorefining step, the fuel alloy can be dissolved either by direct dissolution or by anodic dissolution. In direct dissolution, chopped fuel pins contained in a porous cylindrical basket are lowered into a cadmium anode, the lower portion of the electrorefining cell. (Fig.4.10) Above the cadmium is a molten salt consisting of salts with high chemical stability. (e.g. LiCl, KCl, BaCl₂ etc.) Under the operating temperature of 500 C, the fuel materials are dissolved from the cladding with the addition of oxidizing reagent cadmium chloride salt. The U, Pu, Zr and soluble fission products are distributed between the cadmium anode and the

Fig.4.9 Pyrochemical Process for IFR Core and Blanket [4]

salt electrolyte based on their solubility and chemical potential. Alkali metals, alkaline earth, rare earths and iodine fission products favor the electrolyte; U & Pu are distributed to both phases; and the remaining noble metal fission products and the insoluble cladding stay in the cadmium anode. In anodic dissolution, chopped fuel pins in the basket are lowered into the electrolyte, then an electric current is applied to the basket as anode and the cadmium pool as cathode in order to dissolve the chopped fuel pins. This is the reverse of a normal electrorefining process.

After the dissolution of the spent fuel, the normal electrotransport process is then carried out to separate the heavy metals from the fission products. With the properly applied voltage across the electrodes, U, Pu and other actinides are transferred from the cadmium pool to the cathode. The fission gases such as Xe and Kr are released to the cell chamber, from which they are pumped out and compressed into storage cylinders. The cathode deposits are subsequently removed from the cell.

In the salt/metal separation step, the deposits are melted in its own BeO crucible to consolidate the heavy metals as product ingots. The remaining salt and cadmium from the product is recycled back to the core electrorefining cell. After the alloy composition is adjusted, the ingots becomes the feed to the injection casting process, which is described in sec.4.2.3.3.

In the halide slagging step, the blanket element is melted at 1250 C, in contact with a molten salt containing UCl_3 , thereby

Fig.4.10 Electrorefining for Recovery of U & Pu [4j]

Fig.4.11 Halide Slagging for Recovery of Pu from Blanket U [4j]

oxidizing Pu and transferring it to the salt. The Pu-rich salt phase, the upper portion of the slags (Fig.4.8), is fed to the core electrorefining cell to concentrate the core element. The Pu-depleted metal phase is recast into the blanket element.

Principal wastes from the pyrochemical process are the chloric salt electrolyte and cadmium anode metal, fuel cladding, fission gases and TRU-contaminated wastes from fuel melting operations. It has been suggested that the high level wastes could be converted to metal alloys or glass, a potential waste form to be sent to a geological repository.

4.4 Waste Management

As mentioned earlier, oxide fuel favors PUREX process, while metal fuel favors pyrochemical process. The pyroprocess treating oxide fuels is under development, which is not described here.[41] On the other hand, the pyroprocess has an economic advantage in treating metal fuels over PUREX process. The process waste volumes are compared between the two processes.(Table 4.1) There are several reasons behind the superiority of the pyroprocess;

(1) The pyroprocess plant has smaller dimensions due to the fewer steps in the processes and consequent lower capital cost.

(2) The pyroprocess produces smaller volume of waste.

(3) In the pyroprocess, the high level waste is immobilized into a concrete matrix rather than a glass matrix, reducing the treatment cost.

(4) In the pyroprocess, the cell gas is purified by a simpler method.

(5) Several radionuclides such as C-14, I-129, Ru and T can be easily contained in the pyroprocess.

(6) Long-lived actinides are separated from waste stream in the pyroprocess, reducing the radiological lifetime of the resultant high level waste from millions of years to hundred years.

(Fig.4.12)

Table 4.1 Comparison of Process Waste Volumes in L/MWy(th) [4k]

Fuel Type:	Purex		Pyrochemical
	Oxide	Metal	Metal
<u>High Level - TRU</u>			
Glass Matrix	1.6	5.1	--
Concrete-Cladding Matrix	1.8	1.5	--
Reduction Salt in Concrete	--	3.0	--
Metal Matrix	--	--	4.0
Off-Gas Filters	0.1	0.1	<0.1
Misc. Wastes	0.3	0.4	0.1
<u>Intermediate Level - nonTRU</u>			
Salt-Hardware in Concrete	--	--	12.0
Assembly Hardware	12.0	10.0	0.0(a)
Iodine Containment	1.0	1.0	0.0(b)
<u>Low Level - nonTRU</u>			
Absorber Beds	0.7	0.7	0.0(c)
Fission Gases	0.4	0.4	0.4
Tritium Containment	13.0	13.0	0.1
Carbon-14 Containment	0.1	0.1	0.0(d)
Off-Gas Filters	0.3	0.3	<0.1

- a. Assembly hardware in concrete-salt matrix.
b. Iodine in concrete-salt matrix.
c. Negligible quantity produced.
d. Carbon-14 in concrete-salt or metal matrix.

Fig.4.12 Radiological Risk Factors of Fission Products and Actinides in PWR Spent Fuels [4m]

4.4.1 PUREX Wastes

The PUREX wastes consist of mainly (1) tritium-contaminated low level waste, (2) assembly hardware, (3) halide salts, sodium and corrosion products such as ZrO_2 . The corrosion products and sodium bonding could be immobilized into a glass matrix, which is high level waste. The halide salts and cladding hulls could be encased in a concrete matrix.

The PUREX process produces considerably more low level wastes such as tritium and more high level glass and concrete matrix, leading to much larger volume of wastes. Without additional work of actinide separation, the radiological risk of the high level waste would stay high up to millions of years, requiring the costly site characterization and severe licensing requirements of the geologic repository.

4.4.2 Pyrochemical Process Wastes

The process wastes consist of solid wastes and fission gasses. The solid wastes are (1) metal and salt from electrorefining cells, (2) assembly hardware and plenum cladding and (3) ceramic crucibles. Also, the fission gases are mostly the Xe, Kr and T, which are released when the cladding is chopped or when the fuel is dissolved in the electrorefining cells.

These primary wastes could be treated in the ways adequate to their own properties.[4k] Table 4.2 and Fig.4.13 illustrate specifications and treatment methods of the solid wastes from the pyroprocess of IFR. The actinide-removed salt and assembly hardware are classified as non-TRU intermediate level waste, which

Table 4.2 Solid Wastes from the Pyroprocess of IFR [4k]

Average Reactor Electrical Output: 1000 MW
 Total Reactor Thermal Output: 2880 MW/y
 Time in Reactor: 4 y; Cooling Time: 1 y
 Processing days/year: 260

<u>Electrorefining Waste - Metal</u>		<u>Electrorefining Waste - Salt</u>	
Cd + Fission Products	17 kg	Salt + Fission Products	15 kg
Alloy Zr	8 kg		
Cladding (fuel section)	20 kg		
Anode Baskets	1 kg		
Activity	130 kCi	Activity	160 kCi
Decay Heat	640 W	Decay Heat	500 W
<u>Steel</u>		<u>Ceramic Wastes</u>	
Assembly Hardware	120 kg	Crucibles	4 kg
Cladding (plenum)	22 kg	Fume Traps	2 kg
		Pin-casting Molds	15 kg

Fig.4.13 Treatment Methods of Solid Wastes from Pyroprocess [4k]

could be encased in a concrete matrix. In this case, the cement mixed with ion-absorbing material could show a lower leach rate in groundwater. The metal and ceramic crucibles belong to TRU high level waste, which could be immobilized into a metal matrix and packaged in a corrosion-resistant metal container. Cadmium encapsulated in the metal waste could reduce both the leach rate of the long-lived radioisotopes and the corrosion rate of the matrix to the groundwater. Also, fission gases could be removed from the process cell and placed in a steel container.

The radiological risk of HLW could be significantly reduced, because most of actinides accompany naturally with the plutonium in the pyroprocess.

5. THESIS PROPOSAL

Integral Fast Reactor (IFR) require an initial core loading of heavy metals before the actinide recycling process begins. Up to this time, two sources of heavy metals have been considered for the startup core of IFR's. One source would be the current stockpile of oxide fuel discharged from PWR's. This requires separation of the transuranic elements and current reprocessing technologies are immature and cumbersome for the oxide. The other source that has been considered is the current defense inventory of enriched uranium and surplus plutonium. This source would be difficult to gain access to and political pressure not to use up our defense stockpile could prove unsurmountable.

However, there is a third option. This would be to use spent fuel from metal fueled PWRs. Metal fuel from these evolutionary PWR's could be easily reprocessed to provide the heavy metals necessary for the initial core loading of IFR's. In Addition, the depleted Uranium metal from the AVLIS enrichment process of PWR fuel could be utilized as the makeup of the blankets for the IFR. Moreover, using the actinides recovered from PWR metal fuel as a feed for the IFR would greatly reduce the volume of wastes generated from the PWR.

Fig. 5.1 IFR Fuel Cycle Combined with Metal-Fueled PWR

The economics of metal fueled PWR's are analyzed on the basis of two viewpoints; One is the economic viability of the metal fueled PWR itself in the nuclear industry. The other is the secondary benefit of the metal fueled PWR on the IFR fuel cycle. Since the IFR pyroprocess favors metal fuel rather than oxide fuel, a metal fuel would be less costly. On the other hand, this would require some modifications in an IFR pyroprocess facility to accept the spent fuel from the PWR and reprocess it. The mass flow of actinides and fission products in each stream of the modified IFR fuel cycle would be evaluated to characterize the possible design changes in the processes. Modifications would be specified as much as possible and estimated as an increment to the total fuel cost of the IFR, while the initial core from spent metal fuel would result in its savings.

High content (50 wt%) of zirconium in the PWR metal fuel would be a potential valuable inventory which could be loaded into the IFR core rather than being disposed of. At the present, an electrorefining technique can remove about half the alloyed zirconium from the metal waste. Thus, the pyroprocess needs to be investigated to improve the recovery ratio of zirconium content, otherwise destined to the waste.

REFERENCES

- 2a) "World List of Nuclear Power Plants", Nuclear News, Vol.34(2) 72 (1991)
- 2b) J.H. Rust, "Nuclear Power Plant Engineering", S.W. Holland Company, 1979.
- 2c) S. Glasstone and A. Sesonske, "Nuclear Reactor Engineering", 3rd Ed., Van Nostrand Reinhold Co., 1967.
- 2d) R. Vijuk and H. Bruschi, "AP600 offers a simpler way to greater safety operability and maintainability", Nucl. Eng. Int. Vol.33(411), 22-28 (1988)
- 2e) S.N. Tower, T.L. Schulz and R.P. Vijuk, "Passive and Simplified Features for the Advanced Westinghouse 600 MWe PWR", Nucl. Eng. Design, Vol.109, 147-154 (1988).
- 2f) K.E. Stahlkopf, J.C. Devine and W.R. Sugnet, "US ALWR programme sets out utility requirements for the future", Nucl. Eng. Int. Vol.33(411), 16-19 (1988)
- 2g) D. Lancaster, "A Proposal for Evaluation of the Use of Metal Fuel in PWRs", Georgia Tech (1989)
- 2h) D. Lancaster, "A Proposal for Advanced Analysis of the Use of Metal Fuel in PWRs", Georgia Tech (1991)
- 2i) C.E. Till & Y.I. Chang, "The Integral Fast Reactor", Trans. Am. Nucl. Soc., Vol.60, 449 (1989)
- 2j) J. Cartron, "New Interest in Passive Reactor Designs", EPRI Journal, Apr./May (1989)
- 2k) C.E. Till & Y.I. Chang, "The integral Fast Reactor", Advances in Nuclear Science and Technology, Vol.20, Plenum Press, 1988.
- 3a) K.R. Kummerer, "Metallic Fuel for Fast Reactors", Kerntechnik, Vol.52(1), 21-27 (1988)
- 3b) L.C. Walters, B.R. Seidel & J.H. Kittel, "Performance of Metallic Fuels & Blankets in LMFBR", Nucl. Tech., Vol.65, 179-231 (1984)
- 3c) H.A. Pray & W.E. Berry, "Corrosion of Uranium Alloys in High Temperature Water", BMI-874 (1957)
- 3d) M.V. Nevitt, "Fast Reactor Fuel Alloys; Retrospective and Prospective Views", J. Nucl. Mat., 165, 1-8 (1989)

- 3e) L.C.Walters et.al., "High Reliability Fuel in the U.S.", Int. Conf. on Reliable Fuels for LMR, Sept.7-11, 1986, Tucson, Arizona.
- 3f) W.G.Steele, A.R.Wazzan and D.Okrent, "Steady-State Fission Gas Behavior in U-Pu-Zr Metal Fuel Elements", Nucl. Eng. Design, Vol.113, 289-95 (1989)
- 3g) T.Kobayashi et.al., "Development of the SESAME Metallic Fuel Performance Code", Nucl. Tech., Vol.89, 183-193 (1990)
- 3h) M.J.Feldman et.al., "Experience in Remotely Fabricating EBR-II Fuel", Trans. Am. Nucl. Soc., Vol.30, 309 (1978)
- 3i) J.P.Becca, et.al., "Performance of EBR-II Driver Fuel", Trans. Am. Nucl. Soc., Vol.12, 90 (1969)
- 3j) R.E.Einziger & B.R.Seidel, "Irradiation Performance of Metallic Driver Fuel in EBR-II to High Burnup", Nucl. Tech., Vol.50, 25 (1980)
- 3k) N.J.Olson, C.M.Walter & W.N.Beck, "Statistical and Metallurgical Analysis of Experimental MarkIa Driver Fuel Element Cladding Failure in EBR-II", Nucl. Tech., Vol.28, 134 (1976)
- 3l) B.R.Seidel & R.E.Einziger, "In-Reactor Cladding Breach of EBR-II Driver Fuel Element", Radiation Effects in Breeder Reactor Structural Materials, Metallurgical Soc. of AIME, N.Y. (1977)
- 3m) B.R.Seidel & L.C. Walters, "EBR-II Metallic Driver Fuel- A Live Option", J. Eng. Power, Vol.103, 612 (1981)
- 3n) C.M.Walters, N.J.Olson & G.L.Hofman, "EBR-II Driver Fuel Qualification and Performance", J. Nucl. Mat., Vol.19, 181 (1973)
- 3o) G.L.Hofman, "Irradiation Behavior of Experimental Mark II EBR-II Driver Fuel", Nucl. Tech., Vol.47, 7 (1980)
- 3p) W.N.Beck, et.al., "The Irradiation Behavior of High Burnup U-Pu Alloy Prototype Fuel Elements", ANL-7388 (1968)
- 3q) J.P.Howe & et.al., "Effect of Composition and Heat Treatment on the Irradiation Performance of U-Zr Alloys", Reactor Sci. & Tech. Vol.4(1), (1954).
- 3r) J.L.Ethridge & et.al., "Fast Flux Facility Core System", Trans. Am. Nucl. Soc., Vol.60, 255 (1989)
- 3s) R.G.Pahl, D.L.Porter, et.al., "Experimental Studies of U-Pu-Zr Fast Fuel Pins in the EBR-II", Met. Trans. A. Vol.21A, 1863-70 (1990)

- 3t) R.G.Pahl, C.E.Lahm & et. al., "Recent Irradiation Tests of U-Pu-Zr Metal Fuel Elements", Int. Conf. on Reliable Fuels for LMRs, Sept.7-11,1986, Tucson, Arizona.
- 3u) B.R.Seidel & G.L.Hofman, et.al., "Experience with EBR-II Driver Fuel", ibid, p2-107.
- 3v) G.L.Hofman, et.al., "Chemical Interaction of Metallic Fuel with Austenitic and Ferritic Stainless Steel Cladding", ibid,p4-121.
- 3w) H.Blanck, "Specification and Characterization of Dense Fuels for LMFBRs", J. Nucl. Mat., Vol.153, 171-177 (1988)
- 3x) B.R.T.Frost, "Nuclear Fuel Elements: Design, Fabrication and Performance", Pergamon Press, Oxford, 1982.
- 3y) C.R.Tripton, ed., "Reactor Handbook: Vol.I Material", Interscience Publishing Co., 1960.
- 3z) The Staff of N-Reactor Project, "Hanford's New Production Reactor", Nucl. Eng. Design, Vol.4, 299-310 (1966)

- 4a) R.N.Hill & D.C.Wade, et.al., "Physics Studies of Higher Actinide Consumption in an LMR", Conf-900418 (1990)
- 4b) M.Benedict et.al., "Nuclear Chemical Engineering", 2nd ed., 1981.
- 4c) S.E.Ion, " Process Development in Nuclear Fuel Fabrication", Nucl. Engr. Vol.31(4), 95-101 (1990)
- 4d) J.H.Pashley, et.al., "Some Fuel Cycle Implications of Metal-Based Uranium Enrichment Processes", Symposium of AIChE, 1982.
- 4e) R.C.Stern & J.A.Paisner, "Atomic Vapor Laser Isotope Separation", AIP Conf. Proc., No.146, 397-400 (1986)
- 4f) J.S.Tulenکو & R.F.Wright, "An Innovative Fuel Design Concept for Improved Light Water Reactor Performance and Safety", Trans. Am. Nucl. Soc., Vol.62, 91 (1990)
- 4g) D.B.Tracey, et.al., "Improvements in the Fabrication of Metallic Fuels", Trans. Am. Nucl. Soc., Vol.60, 314 (1989)
- 4h) H.H.Hausner, "Nuclear Fuel Elements", Reinhold Publishing Co., 1959.
- 4i) K.Ebert, "Reprocessing of Spent Nuclear Fuel: Status and Prospect", Nucl. Energy, Vol.27(6), 361-5 (1988)
- 4j) L.Burris et.al., "The Application of Electrorefining for Recovery and Purification of Fuel Discharged from IFR", Annual AIChE Meeting, Nov.2-9,1986, Miami, Florida.

- 4k) R.H.Rigg & T.R.Johnson et.al., "A Waste Treatment Process for a Pyroprocess Fuel Reprocessing Facility", presented at SPECTRUM'87.
- 4l) Y.I.Chang, " Integral Fast Reactor Fuel Cycle", Trans. Am. Nucl. Soc., Vol.62, 73 (1990)
- 4m) Y.I.Chang, " Fission Energy: The Integral Fast Reactor", CONF-8909263-1 (1990)

APPENDIX C

ECONOMIC OPTIMIZATION OF PRESSURIZED WATER REACTOR CORES

(Thesis Proposal)

Holger J. Pfeifer

Nuclear Engineering Program
G. W. Woodruff School of Mechanical Engineering
Georgia Institute of Technology

Report Submitted in Partial Fulfillment
of the Ph.D. Qualifying Examination
at Georgia Institute of Technology

April 1992

Contents

1	Introduction	1
1.1	Motivation	1
1.2	Scope of Work	4
1.3	Report Organization	5
2	Early PWR Optimization	7
2.1	Primary System Temperature and Pressure	7
2.2	Core Design Methods	8
2.2.1	Optimal Height to Diameter	8
2.2.2	Hydrogen to Uranium Optimization	9
2.2.3	Fuel power density and design dimensions	13
2.2.4	Evolution to 17x17 assembly	15
3	Simplified Fuel Cycle Cost Model	19
3.1	Intro to Linear Reactivity Model	19
3.2	Batch Effect	20
3.3	Refuel Enrichment inclusion into the LRM	25
4	Modern Optimization Models	27
4.1	PWR	27
4.2	Liquid Metal Cooled Reactor	31

5 Preliminary Analysis	32
6 Nonlinear Optimization	36
6.1 General Overview of Nonlinear Optimization	36
6.2 Solution Algorithms	37
6.2.1 Unconstrained optimization	38
6.2.2 Lagrangian multipliers	38
6.2.3 Penalty and Barrier Function	38
6.2.4 Feasible Direction Methods	40
7 Thesis Proposal	41

List of Figures

2.1	Minimum DNBR as a function of coolant temperature rise: power density $200 \frac{W}{cm^2}$ and L is the rod length	10
2.2	Coolant pressure drop as a function of rod length: power den- sity = $200 \frac{W}{cm^2}$ and coolant temperature rise = 60 °F	10
2.3	Initial Conversion ratio for a critical reactor (pin Dia. = .400 in)	11
2.4	Lifetime for a very large reactor (pin Dia. = .400 in)	11
2.5	Burnup, A, and reactivity as functions of H/U ratio	12
2.6	Fuel cycle cost for a representative PWR against enrichment and discharge burnup	14
2.7	Characteristic boiling curve (B-D jump at critical heat flux) .	16
2.8	Influence of fuel assembly type on PWR fuel pin corrosion . .	17
2.9	Internal gas pressure in PWR fuel rods	17
3.1	Reactivity as a function of burnup and ^{235}U enrichment for a representative PWR	20
3.2	Increase in discharge burnup with number of staggered reload batches (fixed reload enrichment)	22
3.3	Core behavior predicted by linear reactivity model for fixed reload enrichment	23
3.4	Qualitative dependence of LRM parameters on enrichment . .	26

4.1	Total energy cost vs. fuel rod Diameter at optimum configuration for CIRENE reactor	28
4.2	"Total" energy cost vs. fuel rod outer diameter for different spacings, channel active heights and steam pressures at a constant radial flux shape factor	29
4.3	Overall coolant reactivity vs. channel active height for different radial power shape factors and spacings at constant pin diameter and pressure	29
5.1	Total costs for a oxide fueled PWR design	34
5.2	Total costs for a Uranium metal fueled PWR design	35

List of Tables

2.1	History of Westinghouse PWR development	18
3.1	Summary of LRM Results for Steady-State Cores	24
5.1	Design data and costs for a oxide fueled PWR design study . .	34
5.2	Design data and costs for a Uranium metal fueled PWR design study	35

Chapter 1

Introduction

1.1 Motivation

New reactors currently being developed and introduced by major reactor vendors around the world are scheduled to replace nuclear units designed in the 1960's. Examples for these new reactor concepts include Westinghouse's AP600, CE's System80+ and the ABB PIUS design. In order to establish consistency among the new designs the Electric Power Research Institute (EPRI) has developed Advanced Light Water Reactor (ALWR) design groundrules. All future LWR's have to meet EPRI's three fundamental design objectives [1]:

- Technical Excellence -
The plant should be safe, efficient, operable, maintainable and compatible with the environment.
- Economic Advantage -
The ALWR must be economically competitive with other power generation options, considering both life-cycle and first costs.
- Investment Protection -

The ALWR must have:

- very low susceptibility to major accidents
- assured licensability
- predictable and controllable construction schedule
- predictable operating cost and plant availability

These objectives in turn lead to a specific set of ALWR design principles:

- Reactor Safety -
A strong emphasis on human factors and on safety system design.
- Simplicity -
Main emphasis on plant simplification from the standpoint of the operator. Additional concentration areas for plant simplicity include safety, construction, operation and maintenance.
- Margin -
Provide a plant with substantially improved margins compared to existing plants
- Reliance on Proven Technology -
Program adhering to "if it's not broken, don't fix it" ethic [1] .
- Standardization -
Standardization of a series of future units to reduce construction times and costs.

The design principle requiring proven technology has lead to new reactor concepts incorporating major components from existing designs. Both AP600 and System80+, the front runners in the Pressurized Water Reactor (PWR) area, rely on existing fuel assemblies which were designed in the 1960's and 1970's. AP600 achieves safety and operating margin goals by reducing power

density and increasing attention to safety system design. The reduction in power density is achieved by increasing the number fuel assemblies from that which is normally required for 600 MW(e). Studies on the effects of varying pin pitch and diameter were not reported.

Changes in the technical and political environments should be pivotal to nuclear fuel/core design. A quick list of the major changes in the environment since fuel assemblies were last modified follows:

- No reprocessing (a once-through fuel cycle) -

Reprocessing for commercial nuclear fuel is no longer permitted in the United States. Due to no positive cost component at the end of the fuel life, longer residence times are advantageous.

- Price changes in ore and enrichment (SWU) costs
- Increased safety concerns -

Desire for passive cooling should influence core design.

- Lower uncertainties in both correlations and data -

Reduced uncertainties and advanced programs dealing with uncertainties in both data and correlations have improved design accuracy.

- Increases in computer speed and storage capacity -

Computing capacity during early core optimization studies (1960's, 1970's) were minute when compared to the power available from today's mainframes, workstations or even personal computers. Microelectronics and silicon chips have placed significant computing power in the hands of most researchers.

- New nonlinear models in Operations Research -

These models were developed in late 70's and hence were not available for the current generation of fuel assembly design.

1.2 Scope of Work

The changes in technology and environment since current PWR cores were designed demands an investigation into the economic benefits obtained by a core reoptimization. The last set of reactor core optimization studies were published in the 70's and dealt primarily with maximizing uranium ore utilization. Utilization is not an appropriate figure of merit given the current low uranium ore prices.

In order to make the core design tractable, simplifying assumptions must be made. First, it will be assumed that the secondary system optimization is independent of the primary system optimization. Second, the appropriate figure of merit is total power cost in mills per kwh for the combined capital, fuel cycle and maintenance charges. Safety will not be used as a primary figure of merit, rather attempts will be made to maintain consistent safety margins.

The core analysis will vary fuel pin diameter, hydrogen to uranium ratio (H/U), and core height with a fixed secondary system to arrive at a new:

- Fuel Pin Diameter
- Pin Pitch
- Number of Fuel Assemblies
- Core Height
- Core Vessel
- Containment Size
- Safety Systems/ Safety Margins

1.3 Report Organization

This report will review core optimization literature for primarily PWR's. Some preliminary analysis has been performed and will be presented. Non-linear optimization will be reviewed and a Ph.D. thesis proposal presented.

Chapter 2 concerns itself with early PWR optimization. First, it describes the development of primary system pressure and temperatures. Second, an overview of the design and development of early PWR cores is given. The overview ranges from establishing initial core height to diameter ratios to core power densities and design dimensions. This section also includes the development of 17x17 fuel assemblies and a chart of Westinghouse's fuel development.

Chapter 3 deals with the fuel cycle impacts on core design. The linear reactivity model (LRM) is introduced and the batch effect is derived.

Modern optimization models are presented in Chapter 4. The chapter will cover both PWR as well as LMFBR optimization models. Since it is currently used in PWR design the variation of parameter method is presented first. More advanced methods such as nonlinear simultaneous optimization are presented for PWR's. A discussion of LMFBR's is included due to more recent developments in optimization techniques.

Preliminary analysis is reported in Chapter 5 and is centered around studies performed at Georgia Institute of Technology on the optimization of a 600MW(e) PWR core. The analysis was based on a single parameter study and provided only limited observations.

In order to efficiently solve core optimization problems, a modern non-linear optimization algorithms will be needed. Chapter 6 provides the background of nonlinear optimization along with applications to the nuclear industry not related to core design.

Last, Chapter 7 gives an overview of the eventual Ph.D. thesis. It describes the objective and constraints that will be applied to the optimization

CHAPTER 1. INTRODUCTION

6

problem. In addition the proposal will point out how the eventual design will be verified.

Chapter 2

Early PWR Optimization

Early design optimization (1950's to 1970's) is relevant to the next generation reactors since major design features have been adopted by next generation units. One of the first items to be set was the primary system temperature and pressure. Section 2.1 discusses this optimization. The core height, number of assemblies, pin pitch and diameter optimization will be discussed in Section 2.2.

2.1 Primary System Temperature and Pressure

Primary system temperatures and pressure pin point plant thermodynamic efficiency. Thermodynamic efficiency substantially dictates whether the plant will be economically competitive or not. Therefore, the initial step of nuclear power plant optimization is the determination of general thermal-hydraulic system characteristics. The primary system pressure for early PWR's such as Yankee Rowe hovered around 2000 psia, not too different from modern PWR's which operate at 2250 psia. Primary system pressure is limited by construction material and cost consideration to 2500 psia [2].

A safety/accident margin requires a maximum of 2000 psia design pressure at the time of the Yankee Rowe design. At 2000 psia the saturation temperature is 636°F. If saturation temperature is reached, bulk boiling will ensue, resulting in decreased heat transfer and reactivity imbalances in the core. These are highly undesirable characteristic in a PWR. The inlet temperature is established by tradeoffs between increased thermodynamic efficiency, increased DNB and increased flow rate.

Due to improved pressure vessel construction and decreases in pressure margins, modern reactors operate at 2250 psia. The pressure increase results in an elevated saturation temperature which may be employed to add safety margin or to increase efficiency [3].

2.2 Core Design Methods

The main variables designers control are core height, pin pitch and diameter and number of assemblies. The core height and number of assemblies can be correlated by a height to diameter ratio. The hydrogen to uranium (H/U) ratio relates the pin diameter to the pitch. Total mass flow rate is normally fixed by the core power and inlet/outlet temperatures.

2.2.1 Optimal Height to Diameter

Simple math can show that the height to diameter ratio for a cylindrical reactor should be near 1.0 to minimize neutron leakage per unit volume.

$$S = (2 \times \pi \times R \times H) + (2 \times \pi \times R^2) \quad (2.1a)$$

$$V = \pi \times R^2 \times H = \text{constant} \quad (2.1b)$$

$$H/D = \text{variable} \quad (2.1c)$$

It is possible to solve for the surface area in terms of H/D and volume (V). Differentiating this equation with respect to volume and setting it equal to zero results in:

$$H/D = 1.0 \quad (2.1d)$$

Shifting the H/D ratio away from 1.0 may be desired for thermal hydraulic reasons. Assuming the total core mass flow rate is constant, increasing H/D for a given pin size and H/U ratio will proportionally increase the channel flow rate. As seen in Figure 2.1 the raised channel flow rate positively affects the departure from nucleate boiling ratio (DNBR). DNBR equals the ratio of the actual to the critical heat flux. At the same time the increased H/D ratio will raise the core pressure drop (Figure 2.2). An increased pressure drop requires higher pumping power and reduces the potential for natural circulation cooling [4]. Tradeoffs between DNBR, neutronics, and pressure drop lead to PWR designs with H/D 's greater than 1.0. Designers opting for larger boiling margins over increased pumping power and neutron leakage.

2.2.2 Hydrogen to Uranium Optimization

The hydrogen to uranium (H/U) ratio mainly influences the fuel cycle cost and the moderator temperature coefficient (MTC). An optimal H/U ratio can be determined from a fuel cycle economic study as long as no MTC limitations are violated. The addition of thermal-hydraulic considerations slightly varies the optimal H/U ratio.

The optimum H/U ratio depends heavily on the fuel cycle assumptions made. During the 1950's and 60's plutonium presented a significant positive value. For a fixed enrichment low H/U ratios were desired to increase plutonium production (Figure 2.3). A low H/U ratio tends to decrease fuel reactivity which in turn lowers discharge burnup (Figure 2.4).

The H/U ratio directly determines the moderator temperature coefficient

Figure 2.1: Minimum DNBR as a function of coolant temperature rise: power density $200 \frac{\text{W}}{\text{cm}^2}$ and L is the rod length [4]

Figure 2.2: Coolant pressure drop as a function of rod length; power density = $200 \frac{\text{W}}{\text{cm}^2}$ and coolant temperature rise = 60 °F[4]

Figure 2.3: Initial Conversion ratio for a critical reactor (pin Dia. = .400 in) [5]

Figure 2.4: Lifetime for a very large reactor (pin Dia. = .400 in) [5]

Figure 2.5: Burnup, A (slope constant for reactivity equation), and reactivity(at zero burnup) as functions of H/U ratio [6]

(MTC). For safe PWR operation a negative MTC is required at full power. Figure 2.5 displays the H/U ratios impact on MTC and maximum discharge burnup.

Figure 2.1 and Figure 2.2 illustrate the H/U ratios impact on thermal-hydraulic properties. As can be observed in Figure 2.1, an increase in H/U ratio decreases DNBR margin. The decreased DNBR results from a decreased channel flow velocity produced by the increased channel flow area. The decreased channel flow velocity also reduces core pressure drop (Figure 2.2), a beneficial effect. These effects tend to compensate each, with an increased DNBR favored during the safety analysis. Reverse arguments are true for a potential decrease in H/U.

2.2.3 Fuel power density and design dimensions

Fuel power density and pin dimensions are controlled by a combination of economic considerations and thermal-hydraulic constraints. Historically, economic analysis of the fuel cycle provides a target discharge burnup (30,000 MWD/MTU). Thermal-hydraulic constraints were then used to size the fuel pin.

Fuel power density is a function of several fuel cycle parameters:

$$PowDensity = \frac{BU_{cyc} \frac{1W}{10^6 MW} \frac{10^6 gm}{1 MT} \times \rho \times .88 \frac{g UO_2}{g L}}{Cycle Length (days) \times CF} \quad (2.2)$$

$$Bu_{cyc} = \text{cycle burnup} \left(\frac{MWD}{MTU} \right)$$

$$\rho = UO_2 \text{ density} \left(\frac{gm}{cm^3} \right)$$

$$Cycle Length = \text{Cycle length in days}$$

$$CF = \text{capacity factor - system dependent constant}$$

$$PowDen = \text{Power Density in fuel} \left(\frac{W}{cm^3 UO_2} \right)$$

In order to calculate optimum power density, fuel cycle length and cycle burnups have to be found. Cycle burnups are determined by a power cost minimization. The historical target discharge burnup of 30,000 MWD/MTU resulted from a fuel cycle cost minimization under the assumption that recovered plutonium has positive market value. Figure 2.6 graphically indicates approximately 30,000 MWD/MTU as "optimal" [7]. 12 or 18 month are typical cycle lengths since outages have to be located in low demand seasons (spring or fall). Substituting in 12 month cycle, 3 batch core and 30,000 MWD/MTU discharge burnup into Equation 2.2 yields a desired power density of $335 \frac{W}{cm^3 UO_2}$. Given large safety margins and the desire to keep the pin size and the number of pins technologically and economically feasible, this power density was too high to achieve in early designs. Early 60's designs averaged around $230 \frac{W}{cm^3}$. Increasing the number of pins would have involved raising the stainless steel (SS) cladding volume. SS is a material with a rel-

Figure 2.6: Fuel cycle cost for a representative PWR against enrichment and discharge burnup[7]

atively large parasitic absorption cross sections at thermal neutron energies. Translating fuel power density into core power density yields a desired core power density of 72 kw/l for Yankee Rowe ($H/U = 3.7$). To achieve a higher power density (104.7 kw/l) modern designs have increased the number of pins per assembly to circumvent the high linear heat rate and pin surface heat flux problems. The greater number of pins is allowed due to Zircaloy cladding and improved manufacturing technology.

Two major thermal-hydraulic considerations limit the power density permitted in the fuel:

- DNB limit - a limit on the maximum heat flux permitted (W-3 correlation). Upon reaching a DNBR of 1.0, a radical drop in heat transfer rate occurs, resulting in a large jump in fuel centerline temperature (Figure 2.7).
- Fuel Centerline melt - a limit on the maximum linear heat generation rate (LHGR) (linear heat rate $\propto$ pin centerline temperature).

By applying sufficient safety margin to the limiting factors a "design to limit" approach will yield a feasible pin radius for minimum number of pins in the core (reducing costs), assuming optimal H/D and H/U ratios have been established [9].

2.2.4 Evolution to 17x17 assembly

Westinghouse changed its assembly design from 15x15 to 17x17 to raise core power density under regulatory pressure. The principle behind the 17x17 design is the retention of both assembly area and H/U ratio while improving fuel performance. The greater number of pins per assembly reduces the surface heat flux and linear heat rate and correspondingly increases DNBR and decreases fuel centerline temperature. The 17x17 assembly allowed plants to

Figure 2.7: Characteristic boiling curve (B-D jump at critical heat flux)[8]

maintain high core power density while maintaining emergency core cooling and vessel safety margins [10].

The potential for a core power increase is not the only advantage obtained from the 17x17 design. Plants desiring higher discharge burnups and extra safety margin also receive benefits from the higher number of pins. The 17x17 reduces corrosion and fuel pin internal pressure by a decrease in LHGR (seen in Figure 2.8 and Figure 2.9 respectively).

Table 2.1 summarizing changes and advances made in PWR design by Westinghouse [12].

Figure 2.8: Pressurized water reactor cladding waterside corrosion. Influence of fuel assembly type (same mass flow, system pressure, coolant inlet/outlet temperature and power history)[11]

Figure 2.9: Internal gas pressure in PWR fuel rods (- plenum length 5%, - plenum length 10%) [11]

Table 2.1: History of Westinghouse PWR development [12]

Plant Name	Yankee Rowe	SELNI	San Onofre	Zorita	Zion	Beaver Valley	Trojan
Start Const.	1957	1961	1961	1965	1968	1970	
Commercial Op.	1961	1965	1967	1968	1973	1975	1975
Power(th)	485	875	1317	510	3250	2652	3111
Designer	West.	West.	West.	West.	West.	West.	West.
Core Height(cm)	233	265	301	213	366	366	366
H/D	1.28	1.06	1.09	1.31	1.09	1.20	1.08
Fuel Rod Dia.(in)	.31		.122		.122	.122	.374
Pellet Dia.(in)	.294	.35-.37	.383		.3659	.3659	.3225
Pitch (in)	.122	.513	.557	.568	.563	.563	.496
Assembly	76	157	157	69	193	153	193
# of pins/Ass.	6x6	15x15?	11x14	14x11	15x15	15x15	17x17
Cycle (month)	12	12	12	12-14	11-11	11-14	18
Cladding/Fuel	SS	SS	SS	Zr	Zr	Zr	Zr
Fuel	UO ₂	UO ₂	UO ₂	UO ₂	UO ₂	UO ₂	UO ₂
Pressure (psi)	2000	2020	2485	2100	2250	2250	2250
Avg. Heat Flux $\frac{B}{cm^2}$	33.7	40.7	45.2	49.0	65.6	65.6	59.86
Avg. Power (kw/l)	72	61.1	71.6	77.0	101.7	101.7	103.5
Avg. Power (kw/kgU)	23.2	21.0	23.5	31.3	37.6	37.6	38.36
$F_{\Delta H}$	2.56	2.11		1.90	1.60	1.58	1.55
F_Q	4.19	3.25		3.20	2.79	2.72	2.50

Chapter 3

Simplified Fuel Cycle Cost Model

Fuel cycle costs integrated over the life of the plant are a significant percentage of the total power cost. Plant design can have a major impact on these costs. The extensive neutronic modeling required to estimate fuel cycle costs in the past made the simultaneous analysis of plant and fuel cycle costs nearly impossible. This chapter will introduce a simplified fuel cycle cost model developed in the late 1970's.

3.1 Intro to Linear Reactivity Model

The linear reactivity model establishes a functional relationship between fuel discharge burnup and the reactivity of the fuel. The relationship is linear in nature for uranium oxide fuel over the enrichment range encountered in commercial PWR's (see Equation 3.1 and Figure 3.1)[6]. Uranium depletion, and plutonium and fission product buildup combine to produce the linear reactivity curve.

Figure 3.1: Reactivity as a function of burnup and ^{235}U enrichment for a representative PWR [6]

$$\rho = \rho_0 - AB \quad (3.1)$$

ρ = reactivity at any given burnup

ρ_0 = reactivity at zero burnup

A = slope constant

B = burnup

For a one batch core with the final reactivity set to zero a discharge burnup can be calculated using Equation 3.1 (B_1 represents discharge burnup for a one batch core):

$$B_1 = \frac{\rho_0}{A} \quad (3.2)$$

3.2 Batch Effect

By employing the linear reactivity model, the effects of discharge burnup and cycle reactivity due to a variation in the number of fuel batches can be judged.

For the analysis of a simple batch effect model, certain assumption must

be made:

- Identical lattice structure
- Fixed refuel enrichment
- Equal power sharing between assemblies

Equal power sharing results in the mean reactivity of the core equaling the average batch reactivity.

$$\rho_s = \frac{1}{n} \sum_{j=1}^n \rho_j \quad (3.3)$$

For an n batch core, individual batch reactivity versus burnup curves can be written from Equation 3.1.

$$\begin{aligned} \rho_1 &= \rho_0 - AB_c \\ \rho_2 &= \rho_0 - 2AB_c \\ \rho_3 &= \rho_0 - 3AB_c \\ &\vdots \\ \rho_n &= \rho_0 - nAB_c \end{aligned}$$

The average reactivity for a core with the previous burnup profile yields

$$\rho_s = \frac{1}{n} \left[n\rho_0 - \frac{n(n+1)}{2} AB_c \right] = 0 \quad (3.4)$$

(employing $\sum_{j=1}^n j = \frac{n(n+1)}{2}$)

which in turn leads to Equation 3.5 and Figure 3.2 when solved for discharge burnup.

$$B_d = nB_c = \left(\frac{2n}{n+1} \right) \left(\frac{\rho_0}{A} \right) = \left(\frac{2n}{n+1} \right) B_1 \quad (3.5)$$

4 of 4

Figure 3.2: Increase in discharge burnup with number of staggered reload batches (fixed reload enrichment) [6]

B_d = discharge burnup

ρ_0 = reactivity at zero burnup

n = number of batches in core

The batches effect on discharge and cycle burnup is further illustrated in Figure 3.3. Table 3.1 provides burnup and reactivity information dependent on the constraint chosen: fixed discharge burnup, cycle burnup or initial fuel reactivity.

The assumptions used to produce Table 3.1 must be relaxed to produce a core design oriented relationship. Lattice parameters, particularly pin diameter and H/U ratio, have a significant effect on initial fuel reactivity and the slope constant A (Figure 2.5). Power sharing also affect discharge burnup by appropriately weighting the core average reactivity. While the inclusion of power sharing increases accuracy, only minor variation in discharge burnup results [6].

Figure 3.3: Core behavior predicted by linear reactivity model for fixed reload enrichment[6]

Parameter	Constraint		
	Fixed ρ_0	Fixed B_d	Fixed B_c
Discharge burnup, B_d	$\left(\frac{2n}{n+1}\right) B_1$	B_1	nB_1
Cycle burnup, $B_c = B_d/n$	$\left(\frac{2}{n+1}\right) B_1$	B_1/n	B_1
Core average burnup or FP inventory	BOC	$\left(\frac{n-1}{n+1}\right) B_1$	$\left(\frac{n-1}{2n}\right) B_1$
	EOC	B_1	$\left(\frac{n+1}{2n}\right) B_1$
	Cycle average	$\left(\frac{n}{n+1}\right) B_1$	$\frac{1}{2} B_1$
Core average cycle $\Delta\rho$ or control poison	BOC	$\left(\frac{2}{n+1}\right) \rho_0$	$\frac{1}{n} \rho_{01}$
	Cycle average	$\left(\frac{1}{n+1}\right) \rho_0$	$\frac{1}{2n} \rho_{01}$
Reactivity intercept, ρ_{00}	ρ_0	$\left(\frac{n+1}{2n}\right) \rho_{01}$	$\left(\frac{n+1}{2}\right) \rho_{01}$

Table 3.1: Summary of LRM Results for Steady-State Cores [6]

3.3 Refuel Enrichment inclusion into the LRM

Thus far, the LRM has dealt with fuel reactivity but not refuel enrichment. During cost analysis and core design the refuel enrichment is needed, requiring a modified LRM. In addition to enrichment costs, refuel enrichment also directly controls the amount of ore to be purchased. As can be observed in Figure 3.4 refuel enrichment is linearly related to ρ_0/A and A and virtually linear with respect to ρ_0 for constant lattice parameters.

By substituting $\mathcal{J}(x_p - x_0)$ for ρ_0/A in Equation 3.5, Equation 3.6 is derived. Equation 3.6 can be directly inserted into fuel cost calculations upon suitable variable substitution.

$$B_d = \left(\frac{2n}{n+1} \right) \mathcal{J}(x_p - x_0) \quad (3.6)$$

- $\mathcal{J}$ = variable dependent on lattice structure and power sharing
- x_p = refuel enrichment
- x_0 = lattice dependent variable

Figure 3.4: Qualitative dependence of LRM parameters on enrichment [6]

Chapter 4

Modern Optimization Models

Nuclear optimization in the 1980's dealt principally with improving the nuclear fuel cycle not core design. The refuel studies targeting reduced fuel cycle costs and increased safety margins. Theoretical advances have been made in core design methodology, but they have not been applied to actual designs. These advances center on the treatment of uncertainties using probabilistic function and nonlinear optimization models which solve all constraints simultaneously.

4.1 PWR

Variation of parameters has been employed widely for the evaluation of new reactor designs (ex. [4], [13], [14]). Extensive use of parameter variation was made during the design of a hybrid pressurized water reactor (CIRENE) in France in the early 1970's to estimate both the cost per kilowatt hour and preliminary construction costs [15] [16] (Figure 4.1 and Figure 4.2). The CIRENE report also included a technical impact study to evaluate safety aspects of the different design alternatives. Figure 4.3 portrays the influence of different design alternatives on coolant reactivity, an important safety

Figure 4.1: "Total" energy cost vs. fuel rod outer Diameter at $P_d = 48 \text{ kg/cm}^2$ in correspondence of optimal configurations with respect to channel active height, spacing and radial power shape factor [16]

consideration.

An approach similar to CIRENE's taken at the Georgia Institute of Technology during the evaluation of Zirconium/Uranium alloy fueled PWR's [17] revealed severe short comings in the variation of parameter method. Since not all constraints were satisfied and optimized simultaneously, the study left doubt whether the impact of interacting "independent" parameters was adequately accounted for. Additionally, a thorough search of all plant parameters would not be economically feasible due to the large time and man hour investments required. Variation of parameters is also not compatible with uncertainties in correlations and input variables. Only with great difficulty can error bands can be applied to a solution. More advanced models, such as the ones described next and those included in Chapter 6, circumvent this problem by providing solutions in either probabilistic form or with boundaries on independent parameters under which the solution holds.

In the late 1960's and early 1970's, advanced linear and nonlinear programming, combined with advances in computer technology enabled designers to consider more complex problems. Simultaneous consideration of both

Figure 4.2: "Total" energy cost vs. fuel rod outer diameter for different spacings, channel active heights and steam pressures at radial flux shape factor $F_R = .825$ [16]

Figure 4.3: Overall coolant reactivity vs. channel active height for different radial power shape factors and spacings at pin diameter $D_o = 17\text{mm}$ [16]

the thermal-hydraulic and neutron constraints were becoming possible. As early as 1964 nonlinear codes were shown to be useful in the analysis of simultaneous constraints in the nuclear industry [18]. However, two major problems were encountered by the analysis: error propagation by the solution algorithm and convergence. L.S. Popyrin attempted to apply nonlinear optimization to a nuclear power plant as a unit in 1978 but did not complete the task primarily due to limited resources [19]. Popyrin identified three major problems with his analysis:

- Inadequately determined data and constraints
- Interdependent Data
- Limited computer resources for automation of the procedure

Even though not all the problems encountered by Popyrin have been completely solved sufficient progress has been made to reevaluate nonlinear optimization potential for nuclear analysis.

Another optimization approach that has become economically feasible due to the boosted computational powers is a Monte Carlo approach. A Monte Carlo analysis of fluctuation in variables such as interest rates and construction times, provides an excellent insight into how to best "optimize" construction schedules and what capital cost [20] can be expected.

As stated previously, current optimization deals primarily with retrieving maximum output from a given basic design. A probabilistic risk assessment (PRA), of the reactor, represents a substantial advance in optimizing reactor output and minimizing power cost over previous methodologies. Operating margins and safety systems can be optimized to any given probability of failure using the PRA technique. A model employing nonlinear programming as the decision tool during probabilistic risk assessment was developed at the University of California in 1986 by X.P. Yang [21]. Not meant for core design Yang's model dealt with the optimization of safety systems. An exceptionally

significant aspect of Yang's work was the inclusion of a sensitivity analysis. The sensitivity analysis estimated the changes in independent variables required to significantly impact the design. The nonlinear constrained problem was reduced to a lower order to speed processing time and was solved using Lagrangian multipliers.

4.2 Liquid Metal Cooled Reactor

While optimization studies for PWR were grounded in retaining "early" core design characteristics, Liquid Metal Fast Breeder Reactor (LMFBR) designers employed more advanced tools. GE's COROPT code [22] is an excellent example of the progress possible in core optimization. Written in 1976, COROPT incorporates basic thermal-hydraulic, neutronic and structural design models. The COROPT user can select an applicable objective function and design constraints and the code will optimize the objective function and solve for the independent variables. As discussed by R. J. McCandless, COROPT employs a "simple, constrained, directed, sequential search optimization algorithm" [23], similar but more efficient than variation of parameters. However, COROPT algorithms still requires several thousand iterations to converge, thereby making computational effort lengthy. The code has been shown to produce reliable result when compared with more detailed analysis tools. Therefore, a similar approach to PWR design with advanced optimization tools should be possible.

Chapter 5

Preliminary Analysis

Preliminary analysis performed while studying the feasibility of a Zirconium-Uranium metal fueled pressurized water reactor showed a strong dependence of minimum cost on core power density. Due to the batch effect presented in Chapter 3 the fuel cycle favors large batch numbers to reduce reload enrichment. The batch effect thus determines an "optimal" power density from a fuel cycle costs perspective when combined with capital costs. Initial analysis of UO_2 fueled PWR confirms the LRM batch effect and the overriding influence of fuel cycle cost on total plant cost. While preliminary studies indicate savings by a change in fuel diameter, similar financial benefits could be obtained by an increased number of fuel assemblies at constant core power. Reduced power density again increases the number of batches, thus reducing costs. To a certain point the elevated construction costs associated with the lower power density can be offset by reduced fuel cycle costs. The AP600 power density of 75kw/l is located near the optimal power density derived at Georgia Tech. Sample data for both metal and oxide fueled cores are shown below in both tabular and graphical formats. Designs displayed were derived by an iterative procedure. The number of fuel pins were increased for a constant H/D ratio until a target DNBR was met. UO_2 fueled core

data are shown in Table 5.1 and Figure 5.1 while Table 5.2 and Figure 5.2 represent Uranium alloy data.

AP600									
pin dia (mm)	6.60	7.62	8.64	9.14	9.50	10.16	11.18	12.19	13.21
Pitch (mm)	8.42	9.89	11.35	12.09	12.60	13.55	15.02	16.49	17.96
Length (m)	2.46	2.91	3.33	3.53	2.90	3.90	4.24	4.56	4.85
Core Volume (liters)	7483	12290	18488	21969	24522	29632	38141	47379	57134
Number of fuel pins	38786	39196	39035	38642	38261	37515	36140	34657	33105
Pressure Drop (kPa)	319.9	166.7	104.2	87.6	79.1	67.9	57.7	52.2	49.2
Incremental Press Drop (kPa/m)	129.9	57.4	31.3	24.8	27.3	17.4	13.6	11.5	10.2
Power Density (kW/liter)	258.3	157.3	104.6	88.0	78.8	65.2	50.7	40.8	33.8
MTU Core Total	18.3	31.0	47.7	57.2	64.3	78.4	102.1	128.0	155.5
Cycle Burnup (MWD/MTU)	57703	34071	22153	18477	16461	13498	10365	8265	6800
No. Batches Per Core	0.78	1.32	2.03	2.44	2.73	3.33	4.34	5.44	6.62
Initial Enrichment (%)	8.82	4.83	3.34	2.95	2.75	2.47	2.21	2.04	1.93
No. of Rods Refueled	49735	29677	19216	15866	13996	11253	8324	6365	5003
Refuel Enrichment (%)	6.71	5.39	4.73	4.53	4.41	4.25	4.08	3.96	3.88
Linear Power (kW/m)	19.7	16.5	14.5	13.8	13.5	12.9	12.3	11.9	11.7

Costs as a Function of Fuel Pin Diameter

AP600									
pin dia (mm)	6.60	7.62	8.64	9.14	9.50	10.16	11.18	12.19	13.21
Initial core (\$)	6.19E+07	5.48E+07	5.63E+07	5.90E+07	6.14E+07	6.67E+07	7.68E+07	8.86E+07	1.02E+08
Refueling (Levelized \$)	4.64E+08	3.67E+08	3.18E+08	3.03E+08	2.95E+08	2.83E+08	2.70E+08	2.62E+08	2.56E+08
Vessel (\$)	2.43E+06	4.00E+06	6.02E+06	7.15E+06	7.98E+06	9.64E+06	1.24E+07	1.54E+07	1.86E+07
Containment	3.23E+07	3.58E+07	4.05E+07	4.31E+07	4.51E+07	4.91E+07	5.57E+07	6.29E+07	7.06E+07
Pump Capital (\$)	7.54E+05	3.93E+05	2.46E+05	2.06E+05	1.86E+05	1.60E+05	1.36E+05	1.23E+05	1.16E+05
Pump Operating (\$)	2.17E+07	1.13E+07	7.06E+06	5.94E+06	5.36E+06	4.60E+06	3.91E+06	3.54E+06	3.34E+06
Total (\$)	5.83E+08	4.73E+08	4.28E+08	4.19E+08	4.15E+08	4.13E+08	4.19E+08	4.33E+08	4.50E+08

Table 5.1: Design data and costs for a oxide fueled PWR design study [24]

Figure 5.1: Total costs for a oxide fueled PWR design [24]

	AP-600								
pin dia (mm)	9.5	7.1	8.1	8.4	8.6	9.1	9.5	10.2	10.9
Pitch (mm)	12.6	9.8	11.4	11.7	12.1	12.9	13.5	14.5	15.6
Core Volume (liters)	24522	16305	24382	26645	28917	33719	37245	44003	52219
Pressure Drop (kpa)	79.1	104.7	71.6	66.8	62.9	57.2	54.2	50.4	47.9
Number of fuel pins	38280	48175	46900	46512	46016	45011	44275	42792	41058
Length (m)	3.66	3.19	3.65	3.76	3.87	4.07	4.21	4.45	4.71
Power (W/gram-U)	29.3	41.6	27.3	24.9	22.8	19.4	17.5	14.7	12.3
MTU Core Total	64.3	45.3	69.0	75.7	82.5	96.8	107.4	127.8	152.6
Cycle Burnup (MWD/M)	12346	17525	11495	10477	9619	8194	7387	6210	5197
No. Batches Per Cor	3.6	2.6	3.9	4.3	4.7	5.5	6.1	7.2	8.7
Initial Enrichment	2.37	2.85	2.30	2.22	2.15	2.04	1.98	1.89	1.82
No. of Rods Refueled	10502	18761	11981	10829	9836	8196	7268	5905	4742
Refuel Enrichment (	4.19	4.47	4.14	4.08	4.03	3.96	3.91	3.84	3.79
Linear Power (kW/m)	13.5	12.3	11.0	10.8	10.6	10.3	10.1	9.9	9.8

Table 5.2: Design data and costs for a Uranium metal fueled PWR design study [17]

Figure 5.2: Total costs for a Uranium metal fueled PWR design [17].

Chapter 6

Nonlinear Optimization

Due to the large number of nonlinear components encountered in nuclear optimization a suitable nonlinear solution algorithm is required. Nonlinear implying that the functions involve higher than first order terms not separable from each other. Nonlinear optimization is an outgrowth of operation research and linear programming and concerns itself with optimizing an object function with a limited number of constraints, where any or all of the components can be nonlinear.

6.1 General Overview of Nonlinear Optimization

Nonlinear programming or optimization solves Equation 6.1 using a variety of computer implementable algorithms.

maximize or minimize

$$F = f(x_1, \dots, x_n) \quad (6.1a)$$

subject to

$$g_i(x_1, \dots, x_n) \{ \leq, =, \geq \} b_i; i = 1, \dots, m \quad (6.1b)$$

$$l_i \leq x_i \leq u_i \quad (6.1c)$$

where

- x_i = independent variable
- F = objective function
- $g_i(x)$ = objective function
- b_i = constrained on function $g_i(x)$
- l_i = lower bound on variable x_i
- u_i = upper bound on variable x_i

These equations may be solved by either *direct methods* or *indirect methods*. Where indirect methods deal with the behavior of a function, such as it's derivative, while direct methods rely on comparisons of the object functions value between different points.

Digital computer enhancement has been the driving force behind the increased use of optimization analysis and many of the new optimization algorithms presented in Section 6.2. Optimization methods are ideal for computer implementation since they are very repetitive and can often be expressed in matrix form.

6.2 Solution Algorithms

Core optimization problems have nonlinear constraints requiring the use of a suitable nonlinear algorithm. Nonlinear optimization algorithms have been developed for constrained and unconstrained problems. The algorithms range from Lagrange multipliers to feasible direction methods. Lagrange multipliers transform the constrained problem to one that is unconstrained. Feasible direction algorithms solve the constrained problem "directly". Each methodology has its advantages and disadvantages as described later in this section.

6.2.1 Unconstrained optimization

Unconstrained nonlinear problems can be solved by a variety of methods [25]. Some of these methods date back into the 1950's with most methods developed in the early 1960's and 1970's:

- Line Searches (One Dimensional)
 - Without Using Derivatives (ex. Golden Section, Fibonacci method [1953])
 - With Derivatives (ex. Bisection Search Method, Newton's Method)
- Multidimensional Searches
 - Without Derivatives (ex. Cyclic Coordinate, Hooke and Jeeves [1961] & Rosenbrooks Methods [1960])
 - With Derivatives (ex. Steepest Descent)
- Conjugate Directions (ex. Davidon-Fletcher-Powell [1963], Fletcher and Reeves [1964] & Zangwill [1967])

6.2.2 Lagrangian multipliers

The use of Lagrange multipliers adds the constraints to the objective function using Lagrange multipliers. A two step iterative optimization process then solves for the optimum multiplier value μ and minimizes the objective function. The extra computational effort required by the addition of the unknown Lagrange multiplier and the for a continuously differentiable function makes this approach not suitable for many applications.

6.2.3 Penalty and Barrier Function

Similar to Lagrange multipliers the penalty and barrier functions algorithm converts a constrained optimization problem to an unconstrained one.

Even though penalty and barrier methods are grouped together they have distinct individual characteristics.

Characteristics of Penalty Functions [25]:

- Deals with both inequality and equality constraints
- Functions need to be continuous
- Forces a positive penalty for points outside the feasible region and no penalty for points inside
- Algorithm can start from outside feasible region
- May require small step size possibly resulting in slow convergence and premature termination

Characteristics of Barrier Function [25]:

- Only handles inequality constraints
- Function required to be continuous
- Function incurs penalty as boundary is approached
- Starting point is required to be *inside* feasible region
- Explicit check on feasibility at every step required

While the use of penalty function for solving nonlinear problems dates back into the mid 50's early 60's (Camp [1955] and Pietrgowski [1962]) barrier function approaches were first proposed in 1961 and not thoroughly investigated until the late 60's [25].

6.2.4 Feasible Direction Methods

Methods of feasible directions deal with the problem of how to move from one feasible point to another while improving the objective value. A variety of methods exist to locate a feasible direction and the distance in which to move in the 'improving' direction [25]. In all these methods the feasibility of the primal problem is maintained.

- Method of Zoutendijk [1960]
- Gradient Projection Method of Rosen (modified 1961 for nonlinear constraints)
- Reduced Gradient Method of Wolfe [1963] (generalized to nonlinear constraints by Abadie and Carpentier [1969])
- Convex Simplex Algorithm of Zangwill [1967] (deals only with linear constraints)

Chapter 7

Thesis Proposal

The primary purpose of the project will be to construct a working optimization model for the cost analysis of a pressurized light water reactor core. Secondary systems will not be incorporated into the model due to the development goals of simplicity and speed and the assumptions that core power, inlet and outlet temperatures and total mass flow rates will remain constant. Eliminating secondary systems effects should result in a good comparison base for the actual core design. Except for the pressure vessel, the only other excore components considered to influence the optimum design will be the primary system coolant pumps and the containment vessel. Safety system costs will be addressed and any benefits obtained by the newly designed core noted. Financial benefits obtained by improved safety systems will be included in the thesis. The model derived will incorporate all necessary mechanical and neutronic parameters necessary for the initial design analysis and will optimize the core for a given set of design parameters. Care must be taken that all functional relationships and dependencies are retained while excluding any variables not appreciably effecting the final result. Nuclear core optimization parameters and relationships are very interdependent and sensitive to minute changes thus making the previous statement vital to the

production of valid results. Functions of temperature at various locations in the core, a DNBR constrained and an assurance of mechanical integrity (stress/strain limitations) along with a reactivity model to estimate loading enrichment and fuel cycle costs will be incorporated into the model. These constraints must hold under both steady state and transient conditions. Certain assumptions will be made to simplify the design and no attempt will be made to optimize fuel assembly loading. The figure of merit employed will be mills per kwh for the combined capital and fuel cycle cost. The model should be machine portable (written in F77) as well as have the option for individual constraint and parameter refinement. It will be basically nonlinear in nature and will employ a good convergence and minimal error propagation algorithm. The resulting design will be tested during both steady state and transient conditions with COBRA/PC and RELAP5/MOD3 in order to insure that no large scale errors exist in the derived model.

Bibliography

- [1] John C. Devine. U.S. ALWR - The EPRI Program in Review. Technical Report 87-JPGC-NE-5. Electric Power Research Institute. Palo Alto, California. 1987.
- [2] E. M. Mabuce and K. A. Hub. A study of favorable design values for a reactor operated for power generation only. Technical Report AEP-99. Atomic Electric project. St. Louis. 1953.
- [3] G. H. Farbam and J. H. Jacobs. PWR's - how they've grown to today's size. *Power*, pages 69-71, April 1968.
- [4] A. R. Radkowsky et al. Optimization of Once-Through Uranium Cycle for Pressurized Water Reactors. *Nuclear Science and Engineering*, 75:265-274. 1980.
- [5] Alexander Sesonske. *Nuclear Power Plant Design Analysis*. United States Atomic Energy Commission. 1973.
- [6] M. J. Driscoll, T. J. Downar, and E. E. Pilat. *The Linear Reactivity Model for Nuclear Fuel Management*. American Nuclear Society La Grange Park, Illinois USA. 1990.
- [7] M. Benedict, T. H. Pigford, and H. W. Levi. *Nuclear Chemical Engineering*. McGraw Hill Book Company. 1981.

- [8] James H. Rust. *Nuclear Power Plant Engineering*. S.W. Holland Company, 1979.
- [9] Brian Frost. *Nuclear Fuel Elements*. 1987.
- [10] W. Marshall. *Nuclear Power Technology*, volume 1, page 221. Carendon Press. 1983.
- [11] F. Wunderlich, F.-W. Aisch, and L. Heins. Fuel mechanical design as a boundary condition for fuel management optimization. *Kerntechnik*. 52(4):234-237. 1988.
- [12] IEAC. Directory of Nuclear Reactors.
- [13] J. R. Seeker. Optimum axial blanket design for PWRs. *American Nuclear Society*. 1983.
- [14] E.K. Fukita and M.J. Driscoll. Design and fuel cycle optimization for a once through PWR cycle. Technical report, 1978.
- [15] S. Bozzola and C. Lombard. Optimization studies for CIRENE reactor system Part 1: Basic criteria and calculation procedures. *Energia Nucleare*. 19(4):236-250. 1972.
- [16] S. Bozzola, C. Lombardi, C. Moro, and G. B. Zorzoli. Optimization studies for CIRENE reactor system Part 2: Technical and economic assessment. *Energia Nucleare*. 19(5):295-316. 1972.
- [17] Dale Lancaster et al. A technique for the Selection of the Fuel Pin Diameter for a Uranium/Zirconium Alloy Fueled Pressurized Water Reactor. *Nuclear Technology*. July 1992.
- [18] B. M. Nel and H. Fenech. Physics and Engineering Design Optimization of Nuclear Power Plant Using Gradient Methods. Technical report, International Conference on Peaceful Uses of Atomic Energy, 1964.

- [19] L. S. Poyrin. Methods of Mathematical Modeling and Optimization of Nuclear Power Plants. *Soviet Atomic Energy*, 40(1):166-173, 1978.
- [20] J. B. Hodge. Monte Carlo Simulation used for Nuclear Power Plant Economic Evaluation. *Energia Nucleare*, 23(4):207-211, 1976.
- [21] X. P. Yang. Optimal safety goal allocation for nuclear power plants. *Reliability Engineering & System Safety*, 25(3):257-258, 1989.
- [22] COROPT Users Manual. Technical Report GEAP 14127, General Electric Company, October 1976.
- [23] R. J. McCandless et al. LMFBR Core Design Optimization Methods. *Trans. Am. Nuc. Soc.*, 24:378-379, 1976.
- [24] Dale B. Lancaster and Holger Pfeifer. Partial Optimization of the Fuel Pin Diameter for a Passive PWR. In *American Nuclear Society Conference Proceedings*, 1991.
- [25] M. S. Bazzara and C. M. Shetty. *Nonlinear programming: theory and algorithms*. Wiley: New York, 1979.

**DATE
FILMED**

1 / 26 / 94

END