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OAK RIDGE SCHOOL OF REACTOR TECHNOLOGY
STUDY OF A FUSED SALT BREEDER REACTOR
FOR POWER PRODUCTION

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
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OAK RIDGE SCHOOL OF REACTOR TECHNOLOGY

F. C. VonderLage, Director

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STUDY OF A FUSED SALT BREEDER REACTOR FOR POWER PRODUCTION

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September 1953

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PREFACE

The program which has evolved at the Oak Ridge School of Reactor Technology consists of two semesters of formal course instruction, followed by ten weeks given to reactor design studies. These studies provide each student an opportunity of applying to a specific but representative problem the principles and technology which the school attempts to impart. Individual student groups are chosen so as to include men of the various engineering professions and scientific fields in much the same pattern found in a typical reactor project.

This report is based on the study made by its authors while they were students in the 1952-53 session of ORSORT. It was made and the report prepared in ten weeks. Obviously those weeks were diligently spent. Even so it would be unreasonable to expect that the study reported here is either definitive or free of defect in judgement. The faculty and many, perhaps all of the students are convinced that the project well served its pedagogical purpose. The report is published for the value it has to those who are professionally engaged in the field.

As the authors have noted, several members of the Oak Ridge National Laboratory gave generously of their time and knowledge. The faculty joins with the authors in appreciation of that help. Most particularly is acknowledged the advice and inspiration which the group received from its consultants, A. M. Weinberg and W. M. Breazeale.

F. C. VonderLage

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for

The Faculty of ORSORT

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In the course of this study we have had considerable good advice and help from our consultant, Dr. A.M. Weinberg, Research Director, ORNL, and in his absence, Dr. W. M. Breazeale, Consultant, ORNL. The members of the staff of the Oak Ridge National Laboratory have given us a large amount of useful information, and have aided us in answering many questions. In particular, the following have given considerable time and effort: R. C. Briant, H. W. Savage, and H. F. Poppendiek, on engineering problems; W. D. Manly and C. J. Barton, Sr., on fused salt problems; S. Peterson, D. E. Ferguson, and W. G. Stockdale, on chemical problems; R. A. Charpie and E. C. Smith, Jr., on nuclear problems; and C. E. Clifford, Jr. on shielding problems. In addition, Mrs. Harriet Madden has been of unfailing help in our day to day trials, and in the preparation of this report.

The Authors

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327 004

TABLE OF CONTENTS

	<u>Page No.</u>
Acknowledgments	4
Table of Contents	5
Illustrations	6
Tables	7
Abstract	8
Chapter 1. PURPOSE AND RESULTS OF STUDY	9
1.1 Object of Study	9
1.2 General Results	9
1.3 General Description of the Reactor	11
Chapter 2. PRELIMINARY CONSIDERATIONS	19
2.1 Introduction	19
2.2 Advantages and Disadvantages of Fused Salts	19
2.3 Choice of Fuel	19
2.4 Choice of Fused Salt	20
2.5 Lithium 7	21
2.6 Choice of Reactor Type	21
2.7 Choice of Temperature and Materials	22
2.8 Methods of Reactor Control	24
Chapter 3. NUCLEAR STUDIES	25
3.1 Nuclear Calculations	25
3.2 Size of Reactor	27

	<u>Page No.</u>
3.3 Determination of UF_4 Concentration	29
3.4 Temperature Coefficient of Reactivity	32
3.5 High Energy Reactions	36
3.6 Equilibrium Purity of Li^7	37
3.7 U^{233} Resonance Integral	38
3.8 Photoneutrons	38
Chapter 4. CHEMICAL PROCESS DESIGN	41
4.1 Introduction	41
4.2 Neutron Economy Consideration	43
4.3 Selection of a Reference Chemical Processing Scheme	52
4.4 Description of the Process	53
4.5 Processing Optimization	58
4.6 Summary	67
Chapter 5. POWER PLANT AND HEAT TRANSFER	70
5.1 Power Cycle	70
5.2 Steam Generator and Heat Transport Systems	72
5.3 Control	76
5.4 Miscellaneous Requirements	76
Chapter 6. REACTOR MECHANICAL DESIGN	79
6.1 Reactor Structure	79
6.2 Heat Generation and Temperature Distribution in the Graphite Reflector	80

327 006

	<u>Page No.</u>
6.3 Thermal Stresses in the Graphite	82
6.4 Temperature Distribution in the Graphite Between the Heat Exchanger and the Edge of the Reflector	85
6.5 Boron Burn-up	87
6.6 Salt-Sodium Heat Exchangers	87
6.7 Salt Dump Tank System	89
6.8 Fission Gas Direct Collection	90
6.9 Within-Shield Atmosphere	90
6.10 Points of Salt Withdrawal from the Return to the Reactor in Salt Processing	90
6.11 Reactor Instrumentation and Control Equipment	91
6.12 Reactor Initial Preheating and Oxygen Removal	93
6.13 Initial Salt Charge to the Reactor	93
Chapter 7. SHIELDING	94
7.1 General Description of the Shield	94
7.2 Preliminary Calculations	94
7.3 Calculation of the Flux	96
7.4 Reactor Shield Calculation	97
7.5 Thermal Insulation and Heat Removal	100
7.6 Sodium Activation	101
7.7 Shielding of Sodium Piping	104
7.8 Breakdown of Shield Cost	105

	Page No.
Chapter 8. ECONOMIC ANALYSIS	107
8.1 Cost Summary	107
8.2 Fixed Charges	107
8.3 General Discussion	113
8.4 Inventory Material Prices	114
Chapter 9. RECOMMENDED RESEARCH	115
9.1 Summary	115
9.2 Nuclear Problems	115
9.3 Engineering Problems	117
9.4 Chemical Processing Problems	119
Appendix A Thorium Resonance Cross Sections	120
Appendix B Change of Resonance Integral with Temperature	122
Appendix C Thermal Neutron Cross Section	124
Appendix D Sample Multigroup Calculations	126
Appendix E Fused Salt Experimental Data	129
Appendix F References	134

327 008

ILLUSTRATIONS

<u>Fig. No.</u>	<u>Title</u>	<u>Page No.</u>
1.1	Cross Section Through Reactor	12
1.2	General Layout	13
3.1	Thorium Absorption Data	28
3.2	$\frac{\text{ThF}_4}{\text{UF}_4}$ and Breeding Ratio vs. Mole % UF_4	30
3.3	Uranium Inventory vs. Mole % UF_4	31
3.4	Gamma Flux in the BSF Reactor	39
4.1	Loss of Breeding Ratio vs 250 MW Unit Processing Period	47
4.2	Loss of Breeding Ratio vs 125 MW Unit Processing Period	48
4.3	Steady State Core Loading of Pa^{233} in 125 MW Unit vs Processing Period and Recovery Factor	50
4.4	Steady State Core Loading of Pa^{233} in 250 MW Unit vs Processing Period	51
4.5	Engineering Flow Sheet of Reference Chemical Processing Scheme	54
4.6	Materials Flow Sheet of Reference Chemical Processing Scheme for a 250 MW Unit with a 300 Day Processing Period	57
4.7	Total Cost of Chemical Processing vs Processing Period	66
6.1	Temperature Distribution Based on Model Used in Section 6.2	83
6.2	Shutdown Fission Product Heat After Infinite Operation at 614 MW	92
7.1	Reactor Shield Sketch	95

<u>Fig. No.</u>	<u>Title</u>	<u>Page No.</u>
7.2	Sodium Activation in Solid B_4C	103
D-1	Master Sheet	127
D-2	Calculation Sheet	128
E-1	Phase Diagram of $LiF-BeF_2-UF_4$	130
E-2	Phase Diagram of $LiF-BeF_2-ThF_4$	131
E-3	Viscosity Data for $LiF-BeF_2$ Fused Salt	132
E-4	Density Data for $LiF-BeF_2$ Fused Salt	133

327 010

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TABLES

<u>Table No.</u>	<u>Title</u>	<u>Page No.</u>
1.1	Heat and Power Plant Data	15
1.2	Nuclear Data	16
1.3	Chemistry Data	17
1.4	Shielding Data	17
1.5	Cost Summary	18
3.1	High Energy Neutron Reactions	37
4.1	Capitalization Costs of 250 MW and 125 MW Reactors' Chemical Plants	60
4.2	Annual Cost of Chemical Plants for Processing Periods of 150, 300, and 600 days	61
4.3	Electrical Output Costs	64
5.1	Plant Capability	71
5.2	General Flow Data	73
5.3	Heat Exchanger Schedule	74
5.4	Reactor Pump Requirements	75
8.1	Summary of Estimates of Investment Costs	108
8.2	Summary of Estimates of Fixed and Operating Costs	111
8.3	Estimate of Fixed Charge Rates	112

ABSTRACT

The possibility of using a fused salt breeder reactor as a heat source for a central station power plant is investigated. Calculations are made for a homogeneous U-233-Th breeding system in a fused LiF-BeF₂ salt, using separated salt and continuous reproduction of the fuel is obtained. Analysis of associated chemical and engineering problems is presented. Economic analysis shows a cost of 6.6 mills/kwh for 235 megawatts of net electrical output. A summary is included of the research needed to determine feasibility.

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Chapter 1

PURPOSE AND RESULTS OF STUDY

1.1 Object of the Study

This report is the result of a ten week study which was the final part of a one year course in the Oak Ridge School of Reactor Technology. The object of the study is to determine the possibility of a central station power plant using a fused salt thermal breeder (or converter) reactor as a heat source. This report is not as much a definite engineering design as it is an answer to the question of whether it makes sense to talk about fused salt reactors for power. Since not all of the necessary information was available, a part of the study was the determination of the research which needs to be done in order to make a final decision on the feasibility of the plant. In order to study the problem more easily, a particular design of reactor was chosen, and the analysis was made on the basis of this particular design. The design is summarized later in this chapter, and the choice of design is described in Chapter 2.

1.2 General Results

On the basis of this design, the use of a fused salt reactor in a central station power plant is both practical and economically justified, subject to the confirmation of the assumptions by a fair amount of research and testing. In this report estimates are made for two sizes of plant, 115 megawatts and 235 megawatts of net electric output, producing power at a cost of 8.2 mills/kwh and 6.6 mills/kwh, respectively. A breeding ratio of unity, taking into account all known losses, is calculated for the $\text{Li}^7\text{F} - \text{BeF}_2 - \text{ThF}_4 - \text{U}^{233}\text{F}_4$ fused salt system, as a homogeneous reactor.

Optimization of the reactor design is not attempted because of lack of time, but an approach to the optimum has been made by the use of careful consideration in the selection of design points.

Many important factors in the calculations are not known and have been estimated or assumed. Some of the more important factors that need to be determined experimentally are:

- a) ratio of capture to fission cross section for U^{233} as a function of energy.
- b) resonance integral of U^{233} , and better measurements of U^{233} and Th^{232} cross sections as functions of energy.
- c) corrosion rates of graphite and inconel in fused salts at temperature of 1300°F or higher.
- d) specific heat, thermal conductivity, viscosity, and density of the fused salt mixture.
- e) scale coefficients of sodium on inconel and fused salts on inconel.
- f) new chemical processing methods for removing fission products from fused salts directly.
- g) vapor pressure of PaF_5 in fused salt systems between 1000°F and 1600°F.

Many important factors have been studied experimentally, but the information has been published only recently or for a limited distribution. These include a variety of fused salt phase diagrams, some corrosion tests, and some nuclear cross section measurements. The constants used in this report are tabulated in the appendices.

1.3 General Description of the Reactor

The particular power plant system chosen for design study is shown in Figs. 1.1 and 1.2. Reference is made to these figures in the following discussion. A more complete analysis of the individual components of the system is given in the later chapters of this report.

The over-all plan of the plant (Fig. 1.2) is similar to a conventional steam power plant, except that the reactor and steam boiler take the place of the coal fired boiler (with its coal, ash, and air handling equipment), and a chemical processing building is added. Provision is made for a second reactor to be built after ten years, which is the estimated life of the first reactor (except for fuel, which depreciates negligibly). Two sizes of power station have been chosen for analysis: 125 MW and 250 MW of electricity (gross capability). The estimated capital investment for these is $\$35.9 \times 10^6$ and $\$56.2 \times 10^6$, respectively, and the cost of electricity is estimated at 8.2 mills/kwh and 6.6 mills/kwh, respectively.

The homogeneous reactor core (Fig. 1.1) is a fused salt solution of $U^{233}F_4$ and ThF_4 in $Li^7F - BeF_2$. This solution serves as fuel, moderator, and fertile region for breeding. The normal average operating temperature of the core is $1400^\circ F$. Since the temperature coefficient of reactivity is negative ($-1.36 \times 10^{-4}/^\circ C$) automatic control of the reactor is assumed (see Chapter 2). Stacked graphite blocks are used both as container for the fused salt and as reflector for the neutrons. The fused salt is pumped through passages in the graphite, and passes through heat exchangers which are located within the graphite mass surrounding the reactor core.

12

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FSBR Power Plant — Cross Section through Reactor

FIG. 1.1
ORSORT
FSBR POWER PLANT
CROSS SECTION
THROUGH REACTOR

1/8 IN = 1 FT

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These heat exchangers are made of inconel and the heat is transferred from the fused salt to liquid sodium. Provision is made for heat exchanger replacement, and the estimated life of the exchangers is one year. There are four primary heat exchangers for the 125 MW plant and eight for the 250 MW plant.

The liquid sodium is pumped through the superheater and reheater in parallel, and then through the boiler, and steam is delivered to the throttle of the turbine at 1450 psi, 1000°F. The turbine in each case, namely 125 MW or 250 MW capability, is a type and size which has been built commercially.

The chemical process is a combination of fluoride volatility and water dissolution in which the uranium, protactinium, lithium, and beryllium are separated from the bulk of the fission products and thorium. A continuous processing scheme is used so that the entire reactor is processed every 600 days for the 125 MW plant and every 300 days for the 250 MW plant. The long processing times are possible because the high thermal efficiency of the plant requires less fuel burn-up than an aqueous homogeneous reactor. The fission product gases (xenon, etc.,) and part of the protactinium are removed from the core by sweeping with helium at the free surfaces of the fused salt.

The reactor is surrounded by a heavy shield which attenuates both gamma rays and the neutron flux, and acts as a thermal insulator for the reactor. A two-foot thick concrete shield surrounds the sodium system, including the main steam boiler, superheater, and reheater. The sodium pumps

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are shielded so that they can be drained and replaced individually without danger to personnel.

Provision is made in the design for preheating the reactor during startup, for dumping the fuel, and for emergency cooling. Tables 1.1 to 1.5 list some of the more important design results.

Table 1.1

Heat and Power Plant Data

Electrical output, gross	(MW)	125	250
" " net	(MW)	115	235
Efficiency of steam plant	(%)	37.1%	38.1%
Heat transferred in primary heat exchangers	(Btu/hr) (MW)	1045 x 10 ⁶ 306	2090 x 10 ⁶ 612
Heat generated in reactor	(MW)	310	616
Total steam output	lbs/hr	860,000	1,720,000
Sodium flow	cfm	3265	6530
Fused salt flow	cfm	1495	2990
Fused salt total volume	ft ³	787	927

Temperatures and pressures are given in Fig. 5.1. More specific data are given in Table 5.3.

Table 1.2

Nuclear Data

Salt composition (mole %):	Li ⁷ F	48.6	
	BeF ₂	48.6	
	ThF ₄	2.73	
	U ²³³ F ₄	.11	
	Li ⁶ F		
	(Start)	.024	
	(Equil.)	.004	
Critical mass, core	(KgU ²³³)	240.9	
Critical volume, core	(L)	17,260	
Critical radius (reflected)	(cm)	160	
		<u>125 MW</u>	<u>250 MW</u>
Total mass of U ²³³	(kg)	311	367
Total volume of salt	(L)	22,260	26,260
Average power density	$\frac{\text{kw}}{\text{L}}$	13.74	23.4
Average specific power	$\frac{\text{kw}}{\text{g}}$	.984	1.67
Thermal absorptions	%		23
Thermal fissions	%		35
Breeding ratio (equilibrium)			1.00
Temperature coefficient of reactivity			-1.36 x 10 ⁻⁴

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Table 1.3

Chemistry Data

	<u>125 MW</u>	<u>250 MW</u>
Average processing time (days)		
Bulk fission products	600	300
Volatile FP and 50% of Pa	13	13
Gases (Xe, etc.) and 50% of Pa	.01	.01
Breeding ratio loss from FP and Pa absorption, Chemical losses	.04	.045
Chemical process holdup		
Fused salt (L)	100	100
U ²³³ (kg)	1	1
Pa ²³³ (kg)	12.3	24.4
Chemical processing cost \$/gmU	1.80	1.20
(including capital investment) mills/kwh	.31	.25

Table 1.4

Shielding Data

	<u>Material</u>	<u>Thickness</u>
Main shield (reactor)	Fire clay	2 in.
	Water pipes	1/2 in.
	Sand	8 ft
	Concrete	2-1/2 ft
Secondary shield (sodium system)	Concrete	2-1/2 ft

Table 1.5

Cost Summary

Electrical output MW	115	235
Capital investment	\$35.9 x 10 ⁶	\$56.2 x 10 ⁶
Investment/kw	\$312	\$239
Fixed charges at 15% (80% load factor) mills/kwh	6.4	4.9
Labor and supervision mills/kwh	.8	.8
Maintenance mills/kwh	.6	.6
Chemical processing mills/kwh	<u>.4</u>	<u>.3</u>
Cost of electricity mills/kwh	8.2	6.6

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Chapter 2

PRELIMINARY CONSIDERATIONS

2.1 Introduction

In order to obtain cost figures for a fused salt reactor in a short time, a reactor design was chosen early in the course of study. This design was somewhat arbitrary, but whenever possible both dollar and neutron economy were considered in the choice of components. This chapter describes the basis of design.

2.2 Advantages and Disadvantages of Fused Salts

A fused salt reactor has both advantages and disadvantages when compared with other types of reactors. The low vapor pressure of fused salts allows the use of near atmospheric pressures in the reactor even at high temperature. A fused salt fuel permits continuous removal of fission product gases, and continuous chemical processing of the fuel is possible. On the other hand, the melting points of fused salts are high, and the heat transfer properties are poor compared to liquid metals. Fused salts do not have as good moderating properties as aqueous solutions, but the higher operating temperatures give a better thermal efficiency, so that a fused salt reactor compares favorably with an aqueous reactor.

2.3 Choice of Fuel

The original statement of the problem postulated a thermal reactor, so that only the U^{233} -Thorium fuel-breeding system was considered. Since U^{233} is not now available in quantity, this presupposes that enough will be made in other reactors so that this reactor could be started. If it is not possible to do this, the reactor could be started using U^{235} to make the

U^{233} from Thorium. This would be a more expensive method, since the neutrons formed per neutron absorbed is less for U^{235} than for U^{233} . In general, the design is for the second nuclear power plant of this kind, i.e., it is assumed that the necessary research and development have been done.

2.4 Choice of Fused Salt

Three types of fused salts were considered: hydroxides, chlorides, and fluorides. The solubility of uranium in hydroxides is not sufficient to be satisfactory, although the moderating properties are good. The chloride moderating properties are poor, and in addition, chlorine would require isotopic separation. Fluorine has fair moderating properties and a low nuclear cross section. From these considerations, the fused fluoride system looked more favorable and hence was chosen.

A number of fused fluorides have been studied in connection with the A.E.C. aircraft reactor program, and phase diagrams have been prepared. (1)

Of the salts studied, the $Li^7F - BeF_2$ system has the best moderating properties and the smallest nuclear cross section, although it has the disadvantage of requiring separated Li^7 . Preliminary calculations indicated that this salt has moderating properties nearly as good as graphite. In addition, both UF_4 and ThF_4 are soluble in this salt in useful concentrations (see Figs. E-1 and E-2, Appendix E), and the eutectic melt has the lowest freezing point of all the fused fluorides considered. Since the heat transfer properties of this salt were estimated to be comparable to or better than other fused fluorides, and also since the melting point is comparatively low, (1,4) the $Li^7F - BeF_2$ fused salt system was chosen for study.

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2.5 Lithium 7

The use of a lithium salt in the reactor requires isotopically pure Li^7 , because Li^6 has a large thermal neutron cross section. Since natural lithium is 92.5% Li^7 , and since the mass numbers are so small, it should require only a small isotope separation plant to obtain fairly high purity Li^7 . For this reactor it is assumed that 99.95% Li^7 (same thermal neutron cross section as sodium) could be obtained at an estimated cost of \$200 to \$600 per pound. Further purification is achieved by burning the Li^6 in the reactor. There is an equilibrium concentration of Li^6 in the reactor, when the burned Li^6 balances the Li^6 formed from the (n, α) reaction on beryllium (see section 3.6). This equilibrium is for 99.992% Li^7 in the lithium.

2.6 Choice of Reactor Type

Breeding reactors can be grouped into four designs: homogeneous one or two region and heterogeneous one or two region. The two region reactor has the disadvantages of increased complexity of design and the use of an absorbing media between the core and breeding blanket. It does have advantages in that the core may be made smaller to increase neutron leakage to the blanket and perhaps a quicker and more efficient method of the Pa^{233} removal can be devised. The one region reactor, although having a larger core to reduce neutron leakage to a minimum, would have the advantage of using only one chemical processing to remove both fission products and the Pa^{233} . It would also preclude the necessity of utilizing two sets of heat exchangers to remove heat from the core and blanket, as in the two region design.

Since both UF_4 and ThF_4 can be added to the $Li^7F - BeF_2$ melt, breeding is accomplished without the use of an external breeding blanket. This simplifies the design, with resultant lower costs of construction and operation.

One advantage of a heterogeneous reactor is that it frequently permits use of a better moderator and a consequent reduction in the fuel inventory. From previous fused salt reactor designs a heterogeneous reactor appeared to be a requirement for low fuel inventory. However, simple one group calculations showed that a homogeneous one region reactor using the $Li^7F - BeF_2$ salt might be nearly comparable and indeed, further and more refined multigroup calculations proved this to be true. The reduced initial investment in Li may offset the increase in construction cost for a heterogeneous core, but because of lack of time this optimization is not carried to a conclusion in this report.

2.7 Choice of Temperatures and Materials

The minimum fused salt temperature was determined from the measured phase diagrams (Figs. E-1 and E-2 in Appendix E). The freezing point of the 50% Li^7F - 50% BeF_2 salt with small percentages of UF_4 and ThF_4 in it is about 400°C (750°F.). The lowest temperature allowed in the fused salt part of the system was arbitrarily chosen to be 100°F higher than this to avoid the danger of freezing the salt. Since the viscosity of the fused salt is high at lower temperatures, pumping problems indicate that the average temperature of the fused salt should be about 1400°F (see Chapter 5).

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Corrosion tests^(2,6,34) of fused salts have indicated that nickel and inconel (of the common metals) are corroded least by the salts at temperatures above 1000°F. High mass transfer of Ni, and low tensile and creep strength at elevated temperatures prohibit the use of Ni. Corrosion of inconel at temperatures above 1300°F is fairly high and may limit its usefulness (see section 6.6). Molybdenum might be a good container material for fused salts, but no tests have been run, and in addition the fabrication of molybdenum is somewhat difficult. No definitive tests have been made on the corrosion of graphite in fused salts. However, fused salts have been heated to temperatures above 1500°F in graphite crucibles with no noticeable corrosion effects.⁽¹⁾ On this basis graphite was chosen as the main container for the fused salts, with the provision that corrosion tests, particularly loop tests in a reactor, be made (see Chap. 9). Because of fabrication problems, inconel was chosen for pump and heat exchanger construction. The temperature of the inconel is kept below 1400°F. (see section 7.5) to minimize corrosion. Provision is made in the reactor design for replacement of inconel parts which are in high temperature regions of the reactor.

The direct heating of the turbine steam in the reactor is impractical. Induced radioactivity in the steam would contaminate the turbine, making the servicing of the turbine difficult and requiring shielding around the turbine. The circulation of fused salt through the boiler, superheater, and reheater would require too much external hold-up of the fuel. High pressure steam inside 1500°F graphite is inherently dangerous. For these reasons a secondary

heat transfer fluid was chosen. Mercury, NaK, and sodium were considered. The high pressure needed for mercury leads to some of the same objections as for steam in the reactor. NaK would work well, but sodium has better heat transfer properties and a higher boiling point. Thus sodium was chosen as the secondary heat transfer fluid.

The operating temperature of this reactor is high enough to permit use of modern high temperature, high pressure steam equipment for electrical generation. Since 125 MW and 250 MW capability turbines have been built, these two electrical outputs were used as the basis for the heat calculations. The use of gas turbines was not considered, since no single machine approaching this size has been built, and since the use of a number of gas turbines would make the physical size of the plant almost impossibly large.

2.8 Methods of Reactor Control

The preliminary design assumed the reactor to be self-controlling through the negative temperature coefficient. Calculation of the temperature coefficient gave a value of -1.36×10^{-4} (see section 3.4) which should be of sufficient magnitude for inherent stable self-control. However time did not permit an investigation of reactor kinetics to determine control characteristics.

Continuous chemical processing provides the method for concentration control. Uranium and thorium are continuously introduced into the reactor to replace that which is burned up, thus maintaining the concentration of these materials to give the desired mean operating temperature.

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Chapter 3

NUCLEAR STUDIES

3.1 Nuclear Calculations

The nuclear calculations assumed a bare spherical homogeneous model, $\text{Li}^7\text{F} - \text{BeF}_2 - \text{ThF}_4$ fused salt carrier, and U^{233} fuel. The fission product poisons and Pa^{233} absorptions were neglected in the nuclear calculations, since they were considered along with chemical processing losses (see Chap. 4). Two region calculations were not made because of lack of time.

The first type of calculations were of the four-factor formula variety⁽⁷⁾ and were used only to obtain approximate values of UF_4 concentration and reactor size for the homogeneous case. These calculations were known to be inaccurate since the fast effect is greater than one and the resonance escape probability is considerably less than one, giving an epithermal reactor for which the four-factor formula is invalid. The "age" of the neutrons in the $\text{Li}^7\text{F} - \text{BeF}_2$ salt calculated by a numerical integration over the fission spectrum and including the first and last collisions,⁽⁸⁾ is 457 cm^2 . This compares favorably with the age in graphite, 350 cm^2 .

Later calculations were of the multigroup type⁽¹³⁾ which were perhaps the most accurate that could be applied to this reactor. The first multigroup calculations consisted of 20 lethargy groups; this was later decreased to 10 groups which gave very similar results and expedited the work.

The accuracy of these calculations is certainly no greater than that of the assumptions made concerning the nuclear properties of U^{233} , Th^{232} ,

and Pa^{233} . The assumptions were necessary to obtain cross sections to be used in the multigroup calculations and their accuracy has a considerable effect on the calculated concentrations of fissionable and fertile materials, and even on the conclusions as to the feasibility of a U^{233} breeder. These assumptions must be investigated before the reactor can be designed with any assurance. A list and discussion of the more important assumptions made at the start of the calculations follows:

1. The assumption which can radically effect the calculations and perhaps the feasibility of the breeding U^{233} reactor is that $\alpha, \left(\frac{\sigma_a}{\sigma_f}\right)$, for U^{233} is a constant with energy and equal to the only measured value .098 at .025 ev.⁽⁹⁾ However, in the case of U^{235} the value for α is known to increase with increasing energy, a trend which would detract from the possible breeding ratio in high temperature U^{233} reactors. An attempt to evaluate the variation of α with energy was upset when the measured fission cross section was found to be larger than the total cross section.
2. The absorption cross section was taken to 1.098 times the fission cross section⁽⁹⁾ since fission cross section measurements are more accurate than transmission measurements.
3. The value of $\eta, \left(\frac{\nu \Sigma_f}{\Sigma_a}\right)$, is taken to be 2.30 and ν to be 2.53. Measurements indicate this as $2.32 \pm .03$ with ν
 $= 2.55^{(15)}$.

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4. The cross section for Th^{232} was adjusted (Fig. 3.1) so as to equal the measured value of the resonance integral. The integrated area under the σ_a curve (from E. C. Smith⁽¹⁶⁾) is less than 20 barns whereas the measured resonance integral is 81.3 barns.⁽¹⁷⁾ Arguments for the method of adjusting thorium cross sections are presented in Appendix A.
5. The Li was assumed to be 99.99% Li^7 which is the expected composition after 5 years of operation assuming 99.95% Li^7 at startup.
6. For expediency in calculating reactors of many different U and Th concentration, the concentrations of LiF and BeF_2 were taken as 50 mole percent each. The slight error introduced is in favor of higher breeding ratio.

A sample 10 group calculation is shown in Appendix D.

3.2 Size of Reactor

The diameter of the reactor was more or less arbitrarily chosen from nuclear, economic and engineering reasons. Some of the advantages and disadvantages of large reactors are the following:

Advantages of large reactors:

1. Enhances breeding ratio by reducing leakage.
2. Allows smaller UF_4 concentrations to obtain critical mass, which reduces U holdup in heat exchangers.

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Fig 3.1
THORIUM ABSORPTION DATA

Figure 3.1

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3. Large mass of fused salt allows absorption of larger amounts of heat with the same temperature rise, providing greater stability for fluctuating power output.

Disadvantages of large reactors:

1. Increases amount of fused salt, with special emphasis on Li^7 .
2. Increases size and weight of structural members, shielding, storage facilities, etc.
3. Increases chemical processing on per gram U basis.

Any attempt to make an optimization of reactor size, however, was hopelessly tied up in variable costs of Li^7 and U^{233} , so that the neutron leakage and heat capacity were used as determining factors and the reactor diameter was set between 10 and 11 feet.

Calculations were then made using a bare radius of 200 cm assuming that a 50 cm graphite reflector would reduce the reflected radius to 160 cm. This would result in a reactor 10'-3" diameter. The results of these calculations are shown in Fig. 3.2, a plot of breeding ratio vs. UF_4 concentration for a fixed reactor diameter. Fig. 3.3 shows the critical mass and total U inventory vs UF_4 concentration.

3.3 Determination of UF_4 Concentration.

The combined chemical losses and neutron losses to fission product poisons were arbitrarily chosen to be 5%, which seemed a reasonable selection

Fig 3.2

Figure 3.2

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Fig 3.3

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on the basis of chemical processing costs. This required that the breeding ratio be 1.05 in order to produce as much fuel as is burned. Looking at Figs. 3.2 and 3.3 it can be seen that the UF_4 concentration should be .11 mole percent to give the desired breeding ratio and that resulting critical mass is 241 kg. The total fuel inventory in the reactor is then 369 kg. for a 250 MW electrical output or 311 kg for 125 MW electrical output.

3.4 Temperature Coefficient of Reactivity

Calculation of the temperature coefficient utilized the asymptotic resonance reactor method as described by A. M. Weinberg.⁽¹⁰⁾ The characteristic equation for this method is

$$K_f e^{-B^2 \tau_f} + K_s \frac{e^{-B^2 \tau_s}}{1 + L^2 B^2} = 1$$

where K_f and K_s are the fast and thermal multiplication constants, and τ_f and τ_s are the ages of neutrons which cause fast and thermal fissions. Since the main resonance peaks of U^{233} occur at lower energies than the main resonance peaks of Th^{232} an approximation may be obtained by lumping the resonance integrals at the energies of the main peaks. Then

$$K_f \cong \eta_1 f_1 p_1$$

$$\text{and } K_s \cong \eta_2 f_2 p_2$$

where η_1 and η_2 are neutrons emitted per neutron absorbed in U^{233}

and $\eta_1 = \eta_2$.

$$f_1 = \frac{\Sigma_a}{\Sigma_T} \int_{E_{\text{thermal}}}^{E_0} \sigma_a(u) \frac{dE}{E}$$

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$$f_2 = \left(\frac{\Sigma_a^U}{\Sigma_a^{\text{Total}}} \right)_{\text{Thermal}}$$

$$p_1 = e^{-\left\{ \frac{N_{\text{Th}}}{\xi \Sigma_T} \int_{E_{\text{thermal}}}^{E_0} \sigma_{a\text{eff}}(\text{Th}) \frac{dE}{E} + \frac{1}{\xi \Sigma_T} \int_{E_{\text{thermal}}}^{E_0} \Sigma_a(\text{Li}^6 + \text{Be}) \frac{dE}{E} \right\}}$$

$$p_2 = p_1 e^{-\frac{Nu}{\xi \Sigma_T} \int_{E_{\text{thermal}}}^{E_0} \sigma_a(U) \frac{dE}{E}} = p_1 e^{-f_1}$$

For the just critical reactor, the simplified critical equation becomes

$$1 = K_{\text{eff}} = \eta f_1 p_1 e^{-B^2 \tau_f} + \frac{\eta f_2 p_2 e^{-B^2 \tau_s}}{1 + L^2 B^2}$$

For a small change in temperature, dT , there is a corresponding change in K_{eff}

$$dK = \eta f_1 p_1 e^{-B^2 \tau_f} \left\{ \frac{df_1}{f_1} + \frac{dp_1}{p_1} + \frac{de^{-B^2 \tau_f}}{e^{-B^2 \tau_f}} \right\} + \frac{\eta f_2 p_2 e^{-B^2 \tau_s}}{1 + L^2 B^2} \left\{ \frac{df_2}{f_2} + \frac{dp_2}{p_2} + \frac{de^{-B^2 \tau_s}}{e^{-B^2 \tau_s}} + d \left(\frac{1}{1 + L^2 B^2} \right) \right\}$$

The changes in the terms of the second portion of the equation can be considered individually:

$$\frac{df_1}{f_1} = d \left[\frac{Nu}{\xi \Sigma_T} \int_{E_{\text{th}}}^{E_0} \sigma_a(U) \frac{dE}{E} + \frac{Nu}{\xi \Sigma_T} d \left[\int_{E_{\text{th}}}^{E_0} \sigma_a(U) \frac{dE}{E} \right] \right]$$

For the low concentrations of U in this reactor $\int \sigma_a(U) \frac{dE}{E}$ does not change with temperature. (8) The density change does not affect the ratio $\frac{Nu}{\Sigma_T}$.

Thus

$$\frac{df_1}{dp_1} = 0$$

$$= -p_1 d \left[\frac{N_{th}}{\xi \Sigma_t} \int_{E_{th}}^{E_0} \sigma_{a\text{eff}}(Th) \frac{dE}{E} + \frac{1}{\xi \Sigma_t} \int_{E_{th}}^{E_0} \Sigma_a(Li^6 + Be) \frac{dE}{E} \right]$$

Since the ratio $\frac{N}{\Sigma}$ does not change with temperature, and since the Li^6 and Be integrated absorptions do not change with temperature

$$dp_1 = -p_1 \frac{N_{th}}{\xi \Sigma_t} d \left[\int_{E_{th}}^{E_0} \sigma_{a\text{eff}}(Th) \frac{dE}{E} \right]$$

The effective resonance integral does change with temperature due to Doppler broadening of the peaks. Assuming that the only change is in the largest peak at 23 ev, and using the derivation of Appendix B,

$$d \left[\int_{\text{peak}} \sigma_{a\text{eff}}(Th) \frac{dE}{E} \right] = \left[\int_{\text{peak}} \sigma_{a\text{eff}}(Th) \frac{dE}{E} \right] \times \left[\frac{\frac{N_{Th} \sigma_{a\text{max}}}{\Sigma_s}}{1 + \frac{N_{Th} \sigma_{a\text{max}}}{\Sigma_s}} \right] \times \frac{\Gamma_{\text{Doppler}}}{\Gamma} \times \frac{dT}{2T}$$

$$\text{at } T = 1400^\circ\text{F} = 1860^\circ\text{R}, \quad \Gamma_{\text{Doppler}} = 0.2 \text{ ev}$$

$$\text{so that } \Gamma = \Gamma_\gamma + \Gamma_{\text{Doppler}} \approx 0.3 \text{ ev}$$

$$\text{also } 1 + \frac{N_{Th} \sigma_{a\text{max}}}{\Sigma_s} \approx \frac{\int_{\text{peak}} \sigma_{a_0} \frac{dE}{E}}{\int_{\text{peak}} \sigma_{a\text{eff}} \frac{dE}{E}} = \frac{56 \text{ barns}}{17.9 \text{ barns}} = 3.13 \text{ (see Fig. 3.1)}$$

$$\text{thus } d \left[\int_{\text{peak}} \sigma_{a\text{eff}}(Th) \frac{dE}{E} \right] = .00224 \text{ dT barns}$$

$$\text{and } dp_1 = -44.7 \times 10^{-6} p_1 dT$$

$$\begin{aligned}
 \underline{d \left[e^{-B^2 \tau_f} \right]} &= -e^{-B^2 \tau_f} \times B^2 d \left[\int_0^{u_f} \frac{D}{\xi \Sigma_s} du \right] \\
 &= e^{-B^2 \tau_f} \times B^2 \tau_f \times 2 \frac{d\rho}{\rho} \\
 &= \frac{2}{\rho} \times \frac{d\rho}{dT} B^2 \tau_f e^{-B^2 \tau_f} dT
 \end{aligned}$$

from the density data in Appendix E (Fig. E-4)

$$\rho_{1400^\circ\text{F}} = 2.037$$

$$\frac{d\rho}{dT} = -230 \times 10^{-6}/^\circ\text{F}$$

for $B^2 = 2.45 \times 10^{-4}/\text{cm}^2$ and $\tau_f = 400 \text{ cm}^2$

$$\underline{d \left[e^{-B^2 \tau_f} \right]} = -22.2 \times 10^{-6} e^{-B^2 \tau_f} dT$$

$$\underline{\frac{df_2}{f_2}} = d \left[\frac{\Sigma_a(U)}{\Sigma_a(\text{Total})} \right]_{\text{thermal}} = 0 \text{ if all thermal cross sections}$$

are proportional to $\frac{1}{v}$.

$$\begin{aligned}
 dp_2 &= e^{-f_1} dp_1 - p_1 e^{-f_1} df_1 \\
 &= \underline{-44.7 \times 10^{-6} p_2 dT}
 \end{aligned}$$

$$\underline{d \left[e^{-B^2 \tau_s} \right]} = \frac{2}{\rho} \frac{d\rho}{dT} B^2 \tau_s e^{-B^2 \tau_s} dT$$

for $\tau_s = 457 \text{ cm}^2$

$$\underline{d \left[e^{-B^2 \tau_s} \right]} = -25.4 \times 10^{-6} e^{-B^2 \tau_s} dT$$

$$d\left[\frac{1}{1+L^2B^2}\right] = -\frac{B^2 d(L^2)}{(1+L^2B^2)^2} = +\frac{L^2B^2}{(1+L^2B^2)^2} \left[\frac{2}{\rho} \frac{d\rho}{dT} dT + \frac{d\sigma_a}{\sigma_a} \right]$$

if $\sigma_a \sim \frac{1}{v} \sim T^{-1/2}$

then $\frac{d\sigma_a}{\sigma_a} = -\frac{dT}{2T}$

so that $d\left[\frac{1}{1+L^2B^2}\right] = \frac{L^2B^2}{(1+L^2B^2)^2} \left[\frac{2}{\rho} \times \frac{d\rho}{dT} - \frac{1}{2T} \right] dT$

for $L^2 = \frac{D}{\Sigma_a} = \frac{1.00}{1554 \times 10^{-5}} = 66.3 \text{ cm}^2$

$$d\left[\frac{1}{1+L^2B^2}\right] = -8.0 \times 10^{-6} \frac{1}{1+L^2B^2} dT$$

Now the change in the simplified critical equation can be written:

$$\begin{aligned} \frac{dK}{dT} = & K_f e^{-B^2 \tau_f} \left[0 - 44.7 \times 10^{-6} - 22.2 \times 10^{-6} \right] \\ & + K_s \frac{e^{-B^2 \tau_s}}{1+L^2B^2} \left[0 - 44.7 \times 10^{-6} - 25.4 \times 10^{-6} - 8.0 \times 10^{-6} \right] \end{aligned}$$

The values $K_f e^{-B^2 \tau_f} = .715$

and $K_s \frac{e^{-B^2 \tau_s}}{1+L^2B^2} = .352$

were found in preliminary reactor calculations.

Thus $\frac{dK}{dT} = -0.75 \times 10^{-4}/^{\circ}\text{F} = -1.36 \times 10^{-4}/^{\circ}\text{C}$

3.5 High Energy Reactions

A calculation was made of several possible high energy neutron reactions. A multigroup method was used which effectively integrated the

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cross section⁽¹¹⁾ over the fission spectrum. The results are given in table 3.1.

Table 3.1
High Energy Neutron Reactions

Reaction	Product Nucleus	Absorptions
		U Absorptions
$\text{Be}^9(\text{n}, \alpha)$	Li^6	.025
$\text{Fl}^9(\text{n}, \alpha)$	O^{16}	.04
$\text{Fl}^9(\text{n}, \text{p})$	F^{19}	.05
$\text{Be}^9(\text{n}, 2\text{n})$	He^4	.0035
$\text{Th}(\text{n}, \text{f})$		

The extra neutron from the $\text{Be}^9(\text{n}, 2\text{n})$ reaction compensates in part for the other losses. The cross section for this reaction is really not known, and should be measured fairly carefully. The $\text{Be}(\text{n}, \alpha)$ absorptions are considered in the multigroup calculations (see Appendix D). The thorium fissions are also considered in the multigroup calculations, and are assumed to have the same number of neutrons per fission as U^{233} .

3.6 Equilibrium Purity of Li^7

The purification of Li^7 in the reactor by the burn-out of Li^6 is complicated by the formation of Li^6 from the $\text{Be}(\text{n}, \alpha)$ reaction. The equilibrium purity of Li^7 was calculated to be 99.992% Li^7 . The nuclear calculations assumed a purity of 99.990% Li^7 . This purity was estimated to be obtained in the reactor with the expenditure of 56.3 kg of U^{233} for

the small power plant, and 66.5 kg for the large plant. This amount of fuel is added to the initial inventory costs (see Chap. 8).

3.7 U^{233} Resonance Integral

The area under fission cross section curve $\times \frac{dE}{E}$ for U^{233}

(BNL-170) was integrated with a result of 770.3 barns for the fission resonance integral. When multiplied by 1.098 $\left(\frac{\sigma_a}{\sigma_f}\right)$ which was assumed constant with increasing energy) this gave a value of 845.8 barns for the absorption resonance integral. The resonance absorption integral as calculated previously⁽⁸⁾ was 442 barns. The lack of agreement between these two values indicates the need for experimental research. The values calculated in this study are used in the calculations.

3.8 Photoneutrons

Since the reactor contains a large amount of beryllium, and also has a large high energy gamma ray flux, a bonus of neutrons is obtained from the $Be(\gamma, n)$ reaction. The threshold energy for this reaction is 1.63 Mev, and the cross section as a function of energy has been measured.⁽⁴³⁾ However, the gamma flux spectrum in a reactor is not well known. Some measurements of gamma flux outside a reactor core have been made in the Bulk Shielding Facility at ORNL. A guess, based on these measurements, is made to determine the flux inside the core (Fig. 3.4) of the BSF.

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Figure 3.4

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In order to make an estimate of the gamma flux in the fused salt reactor, it is assumed that the ratio of the gamma fluxes in the two reactors is equal to the ratio of the power densities. Then the estimated flux is integrated over the cross section to give the number of neutrons formed by the $\text{Be}(n, \gamma)$ reaction. The result of this calculation gives:

$$\frac{\text{Neutrons from beryllium}}{\text{Neutrons from fission}} = 0.00027$$

This is small enough to be negligible in the nuclear calculations. However it does effectively add 10% to the number of delayed neutrons.

Chapter 4

CHEMICAL PROCESS DESIGN

4.1 Introduction

The reactor has 616 MW or 310 MW heat power output (250 MW or 125 MW electrical output respectively). The required chemical plant must be capable of performing the following operations:

- 1) Removal of fission products from the salt
- 2) Recovery of Pa^{233} for decay to U^{233}
- 3) Recovery of U^{233} and Li^7
- 4) Remake of fused salt solution of proper concentration for the reactor.

In some respects, the demands made on this chemical plant are new in the atomic energy field. For instance, the plant requirements are unique in that the expensive fused salt solvent must be stripped of fission products and then recovered completely if costs are not to be excessive. The presence of uranium, thorium, and protoactinium greatly complicates the problem.

Effort has been directed toward extraction of fission products from liquid metals using fused salt as the extracting medium.⁽²²⁾ The reverse of this mechanism has not yet been demonstrated. There is a suggestion that fluoride volatility is applicable for the decontamination of the uranium compound⁽²³⁾ but if the expensive fused salt solvent must be recovered, this is of little help. A metathesis reaction, whereby uranium as an oxide exchanges with the rare earths has been postulated on thermodynamic grounds,⁽²³⁾ but no experimental data have been obtained. An

electrolysis is interesting, but little is known of decomposition potentials of the fused salt components in this combination at various electrodes.

Carrying of fission products by a fog of metal from a liquid metal electrode would be worthy of consideration if more were known of carrying properties, solubility, reactivity, etc., of metals in fused salts. High temperature distillation of all components may be possible.

There is little information regarding the state of existence of the fused salt components at high temperature. Precipitation of the fission product fluorides might take place. Complexing of Li, Th, Pa, and other components, including fission products, undoubtedly occurs. Reduction of the rare earths and other fission products is difficult to predict under these conditions.

Those data of reasonable certainty pertaining to fused salt systems are as follows:

- 1) Xe, Kr, H₃, and volatile fission products at a temperature of 1500°F will exist in the atmosphere over the core solution at the pump impeller inlets.
- 2) The boiling point of PaF₅ at one atmosphere is 1340°F. ⁽²⁴⁾
- 3) UF₆ and the volatile fission product fluorides are separable by fractionation.
- 4) The fused salt will solidify as Li₂BeF₄ and BeF₂. ⁽²⁵⁾
- 5) BeF₂ is completely soluble in any amount of water, and hydrolyzes.
- 6) Li₂BeF₄ is soluble in water at 30°C to the extent of 1.32 g/l. ⁽²⁶⁾

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7) Be is recoverable from water solution by recrystallization as

8) Li and Be are recoverable from water as $\text{Li}_2\text{BeF}_4 \cdot \text{H}_2\text{O}$. (26)

9) ThF_4 and the non-volatile fission product fluorides are insoluble in water.

In addition, because of the fluorine deficiency in the system, the equilibrium valence of the uranium will probably be less than four. Thus, fluorination of the melt in a gas-liquid phase reaction at temperatures near 1300°F will oxidize the U and possibly other components to higher states.

An approach to the selection of the proper processing scheme and the operating parameters is along the following lines:

- 1) Analysis of fission product and heavy element buildup in the reactor as a function of chemical processing.
- 2) Selection and description of a reference chemical processing scheme satisfying the nuclear requirements.
- 3) Selection of the operating parameters with respect to cost.

Nuclear cross section data are given in Appendix C. Definitions of symbols throughout this chapter are given at the end of the chapter.

4.2 Neutron Economy Consideration.

To determine the nuclear characteristics of the reactor as affected by chemical processing, the method of Halperin and Stoughton⁽²¹⁾ is adopted with modifications. This method assesses a loss of neutrons per neutron

absorption by U^{233} as a function of processing time, removal factor, chemical losses, and flux. This, in effect, measures the loss in breeding ratio, depending on the choice of parameters.

In a reactor with a breeding ratio of unity, the following identities hold:

- 1) one fission product neutron absorption $\equiv$ one neutron lost
- 2) one atom of U^{233} lost in chemical processing $\equiv$ one neutron lost
- 3) one Pa^{233} neutron absorption $\equiv$ two neutrons lost
- 4) one atom of Pa^{233} lost in chemical processing $\equiv$ one neutron lost

Modifications of the Halperin and Stoughton values for the fission product cross sections and fission yields for a U^{233} reactor with a thermal energy of .089 ev give:

		<u>Fission Yield</u>		<u>Neutron Capture Cross Section</u>	
		<u>Symbol</u>	<u>Value</u>	<u>Symbol</u>	<u>Value</u>
Removed from reactor as rare gases			0.40		
High cross section rare earths	y_r		0.009 ⁽²⁰⁾	σ_r	85,000 b
Remaining fission products	y_a		1.6	σ_a	25 b

In addition, the chemical processing is considered continuous, and one-half of the Pa^{233} is distilled from the reactor core as rapidly as it is formed. This latter assumption is an estimate to account for the equilibrium existing in the reactor between PaF_5 vapor and complexed Pa in the melt.

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The expressions for the steady state reactor system loadings are listed below:

$$\frac{N_a}{N_{23}} = \frac{y_a \sigma_f^{23} \phi \tau}{\sigma_a \frac{\phi}{3} \tau + R_a}$$

$$\frac{N_r}{N_{23}} = \frac{y_r \sigma_f^{23} \phi \tau}{\sigma_r \frac{\phi}{3} \tau + R_r}$$

$$\frac{N_{13}}{N_{23}} = \frac{\frac{1}{2} \alpha \sigma_c^{02} \phi \tau}{\lambda_{13} \tau + \sigma_c^{13} \frac{\phi}{3} \tau + R_{13}}$$

Alpha (α) is a synthetic constant describing the ratio N_{02}/N_{23} , taken equal to 80 to satisfy the relation in a reactor with unity breeding ratio:

$$N_{23} (\sigma_a^{23} \phi) = N_{02} (\sigma_c^{02} \phi)$$

$$\alpha = \frac{N_{02}}{N_{23}} \approx \frac{\sigma_a^{23} \phi}{\sigma_c^{02} \phi} \cong 80$$

The fission product absorptions and Pa^{233} absorptions occur principally at thermal energies. The ratio of total to thermal flux is taken to be approximately three to one. Hence, with the constant α reasonably describing the relation between Th^{233} and U^{233} absorptions as though in a thermal reactor (see above), and with $\frac{\phi T}{3}$ approximating the thermal flux, the reactor is postulated to be thermal for this analysis and thermal cross sections ($E_{th} = .089 \text{ ev}$) are used.

The resulting equation expressing the neutron loss per U^{233} absorption (the loss in breeding ratio), L , with explanation of terms is as follows:

$$L = \frac{\ell_c^{23}}{\sigma_a^{23}} + \left[\frac{y_a \sigma_a(\phi) \tau}{\sigma_a \phi \tau + R_a} + \frac{y_r \sigma_r(\phi) \tau}{\sigma_r \phi \tau + R_r} \right] \frac{\sigma_f^{23}}{\sigma_a^{23}} + \frac{\frac{1}{2} \sigma_c^{02} \ell_{13}}{\sigma_a^{23}} \left[\frac{1}{\lambda_{13} \tau + \sigma_c^{13} \phi \tau + R_{13}} + 1 \right]$$

$\left(\begin{matrix} U^{233} \\ \text{chem.} \\ \text{losses} \end{matrix} \right)$ (losses to fission products) $(Pa^{233} \text{ chemical losses})$
 $+ \frac{\sigma_c^{13} \sigma_a^{02} \phi \tau}{3 \sigma_a^{23} (\lambda_{13} \tau + \sigma_c^{13} \phi \tau + R_{13})}$
 $(Pa^{233} \text{ absorption losses})$

Graphs of breeding ratio loss vs. processing period for the above equation are shown in Figs. 4.1 and 4.2. Termwise contributions to the total curves are also represented. A total flux of 10^{14} and a thermal flux of 3.3×10^{13} for the 250 MW unit and a total flux of 5×10^{13} and a thermal flux of 1.7×10^{13} for the 125 MW unit are used as averages over the reactor system. The reactor system includes the core and heat exchange section. Recovery factors of 0.9 for the fission products were modestly assumed on the assurance that probably higher decontamination factors can be obtained. A recovery factor of 1.0 for the Pa^{233} , and chemical loss fraction of 0.001 for the U^{233} and Pa^{233} , seem not too optimistic for a potential processing system.

Although Figures 4.1 and 4.2 give reasonably accurate estimates of the loss in breeding ratio, as a function of processing period, they do not define an optimum processing period. Since the cost of chemical processing is generally high, much longer processing periods are necessitated.

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The chemical plant is designed on this basis, and a total cost curve as a function of processing period is presented later to substantiate this view (Fig. 4.7).

Graphs of the steady state reactor system loadings of Pa^{233} as a function of processing period, and recovery factors for both electrical outputs are presented in Figs. 4.3 and 4.4. With the anticipated processing periods, a half life for equilibrium concentration of Pa^{233} is about 23 days, and the half life for the growth of group "a" fission products is less than one year. Group "r" fission products build up to a small steady state concentration independent of longer processing periods since they are essentially burned out as rapidly as they are formed.

The steady state equations of uranium isotope concentration are:

$$\frac{N_{24}}{N_{23}} = \frac{\sigma_c^{23}}{\sigma_c^{24}} \left[1 + \frac{a \sigma_c^{02} \lambda_{14}}{6 \sigma_c^{13} (\lambda_{14} + R_{14})} \left(\frac{\sigma_c^{13} \phi}{\lambda_{13} + \sigma_c^{13} \frac{\phi}{3} + R_{13}} + \frac{\sigma_c^{03} \phi}{\lambda_{03}} \right) \right]$$

$$\frac{N_{25}}{N_{24}} = \frac{\sigma_c^{24}}{\sigma_c^{25}}$$

$$\frac{N_{26}}{N_{25}} = \frac{\sigma_c^{25}}{\sigma_c^{26}}$$

For processing periods of 0 and ∞ the uranium isotopes build up to the following steady state values for both the 250 MW unit and the 125 MW unit.

	$\tau = 0$	$\tau = \infty$
N_{24}/N_{23}	0.685	0.728
N_{25}/N_{23}	0.094	0.100
N_{26}/N_{23}	<u>0.497</u>	<u>0.529</u>
Total	1.276	1.357
N_U/N_{23}	2.276	2.357

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FIGURE 4.4
STEADY STATE CORE LOADING
OF PA233 IN 250 MW UNIT
VS.
PROCESSING PERIOD

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The processing period and the recovery factor for Pa^{233} and Pa^{234} have a minor effect on the heavy isotope steady state concentration. Ninety-five percent of the equilibrium values are attained after about 20 years.

4.3 Selection of a Reference Chemical Processing Scheme

On the basis of the data presented in the introduction, a reference chemical processing scheme has been adopted. This scheme is not the method proposed to serve this reactor. Its shortcomings, both engineering and costwise, are probably manifold. Rather, this processing scheme is to serve as a basis for reasonable estimation of the cost of chemical processing entailed in the cost of electrical power from the reactor.

Undoubtedly the processing can be done more cheaply by utilizing some of the methods postulated in the introduction and discounted because of the dearth of information available. For example, a simple extraction technique, selectively removing the rare earths and fission products, will be much cheaper than any complex scheme that is presented here. Such a technique can surely be developed when the problem has captured the amount of attention given other such problems.

The method adopted here is essentially a water dissolution of the fused salt solvent after removal of the uranium component. The fission product fluorides are generally insoluble in water, as is the ThF_4 . After the insoluble fraction has been filtered out, the salts must be recovered from the solution. They are then recombined with the proper amount of make-up thorium and with the removed uranium, made up to proper concentration of U^{233} by addition of the Pa^{233} decay product. This melt is fed back into the reactor.

For cost reasons, no attempt is made to recover the thorium.

The following sections will discuss the equipment used, and quantities handled, and will present a more detailed description of the process.

4.4 Description of the Process.

Engineering Flow Sheet (see Fig. 4.5)

Gas Separation

Gases are removed from each of the heat exchanger impellers by sweeping with helium. The fused salt is pumped continuously from the bottom of one heat exchanger to a cyclone type gas separator, A. The gases from the separator are combined with those from the impellers and sent to a water cooled condenser, B, where all except the rare gases are liquified. The rare gases are passed through a refrigerated charcoal box, C, where xenon and krypton are adsorbed; the helium is recycled to the inlets of the heat exchanger pumps. The major portion of the fused salt from the gas separator is returned immediately to the reactor.

Protoactinium Decay

The liquid from the condenser, B, is pumped to the protoactinium decay tank, D, where fluorine is added and where UF_6 and the volatile fission product fluorides are distilled continuously into condenser E, leaving the PaF_5 to decay in D. The liquid from E is pumped to distillation column F where the fission product fluorides more volatile than UF_6 are distilled off and sent to waste storage. From F, the bottoms liquid is pumped to a second distillation column, H, where UF_6 is distilled off. The UF_6 is

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condensed in J while the fission product fluorides from the bottoms of H are sent to waste storage.

Fused Salt Decontamination

The remainder of the fused salt, which is not returned to the reactor from the gas separator, is pumped to a fluorination column, K, where the uranium is oxidized to UF_6 at high temperature and distilled from the salt. The UF_6 is then condensed in L and combined with the UF_6 from condenser J.

The fused salt from K passes onto a water cooled belt, M, where it freezes and falls into a dissolver, N, where water is added. The solution of lithium and beryllium with the insoluble fission product fluorides and the ThF_4 is sent to waste storage, O, where it is filtered in a sand-filled asphalt pit. The filtrate is collected in a sump and returned to the reaction vessel, P. The contaminated thorium is expected to build up in the asphalt pit over a period of five years.

Salt Recovery

Ammonium bifluoride is added to the solution in P. The solution from P is evaporated under reduced pressure in Q. Low pressure is maintained by means of a barometric condenser and steam ejector, R. It is expected that HF will be distilled with the water from the evaporator, and, that at the reduced temperature resulting from the reduced pressure, NH_3 will not be distilled. The water solution of HF from the hot well of the barometric condenser is pumped to distillation column S to remove HF. The water from S is sent to a surge tank, T, for recycling to the dissolver, N.

In the evaporator, Q, Li_2BeF_4 and $(\text{NH}_4)_2\text{BeF}_4$ crystallize and the slurry is sent to a spray drier, V, and a rotary drier, W, to remove water and NH_3 .

Salt Remake

The UF_6 from condensers J and L is reduced to UF_4 with hydrogen in unit process X. The HF from S and X is combined with the NH_3 from the driers to remake $\text{NH}_4\text{F} \cdot \text{HF}$ at Y. The salt from W, the UF_4 from X, and make-up ThF_4 are placed in the remake tank, Z, where the entire mass is melted and returned to the reactor.

Material Flow and Equipment (See Fig. 4.6)

Material flow rates and equipment construction and size are described here for a 300 day processing period with the 250 MW unit. A complete picture of the material flow rates appears in Fig. 4.6. A brief description of the equipment follows:

1) Pa^{233} decay and recovery side

All equipment is small, pressurized, and constructed of nickel or inconel.

2) Fused salt decontamination

The equipment is larger because of the limited solubility of Li_2BeF_4 in water. For example, the dissolver, reagent tank, and water holdup tank are of 3000 gallon capacity each. Material of construction is predominantly steel. In general, this equipment is readily available and of commercial types.

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3) Waste storage

58

The asphalt pit is covered and is located at a safe distance from the plant. It contains 3,600 cubic feet of sand. A header distributes the slurry feed evenly over the sand area. Tests on asphalt indicate that it may sustain radiation and thermal effects fairly well under these conditions.

A layout of equipment indicates that almost all the equipment can be fitted into a process cell of 37 ft x 34 ft x 22 ft interior dimensions. This layout allows 2 ft around each unit for accessories. The $\text{NH}_4\text{F}\cdot\text{HF}$ remake system and UF_6 reduction are not included in the cell.

4.5 Processing Optimization

A cost estimate covering the reference processing scheme has been carried out as follows for both the 250 MW and 125 MW units for processing periods of 150, 300, and 600 days:

1) Installed costs of the following have been estimated:

- a) Equipment
- b) Processing cell
- c) Special equipment
- d) Radioactive waste disposal facilities
- e) Laboratory and equipment
- f) Piping
- g) Instruments

2) Pre-operational costs have been estimated.

3) Material inventory costs have been estimated.

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TABLE 4.1

CAPITALIZATION COSTS OF 250 MW and 125 MW REACTORS' CHEMICAL PLANTS

 $\tau = 150$ Days

	Dep'n Time (yrs)	Direct Charges	Indirect Charges	Total Charges	Dep'n Rate
Process Section					
Cell	30	229,000	80,000	307,000	10,300
Equipment	5	68,000	23,500	91,500	18,400
Spec. Equipment	5	75,000	26,300	101,300	20,300
Piping	5	256,000	89,600	346,000	69,300
Inst. & Controls	5	110,000	38,500	148,000	29,600
Crane	30	10,000	3,500	13,500	4,500
Waste Disposal					
Gaseous	10	10,000	3,500	13,500	1,300
Liquid	5	3,000	1,000	4,000	800
Solid	5	16,000	5,600	21,600	4,300
Laboratory & Equip.	15	138,000	48,300	186,000	12,400
Preoperational	10	12,000	4,200	16,200	1,600
Materials Investments	30	90,000	31,000	121,000	4,000
Total					
Average Dep'n Rate					12.9%

 $\tau = 300$ Days

Process Section					
Cell		180,000	63,000	243,000	8,100
Equipment		52,000	17,800	68,800	13,800
Spec. Equipment		50,000	17,500	67,500	13,500
Piping		182,000	64,000	246,000	49,200
Inst. & Controls		78,000	27,300	105,300	21,100
Crane		10,000	3,500	13,500	450
Waste Disposal					
Gaseous		10,000	3,500	13,500	1,350
Liquid		3,000			
Solid		8,000			
Laboratory & Equipment		105,000			
Preoperational		10,000	3,500	13,500	1,350
Material Investments		45,000	15,800	60,800	2,000
Total				988,300	123,310
Average Dep'n Rate					12.5%

$\tau = 600$ Days

Dep'n Time (yrs)	Direct Charges	Indirect Charges	Total Charges	Dep'n Rate
Process Section				
Cell	145,000	51,000	196,000	6,500
Equipment	40,000	14,000	54,000	10,800
Spec. Equipment	40,000	14,000	54,000	10,800
Piping	144,000	51,000	195,000	39,000
Inst. & Controls	62,000	21,700	83,700	16,700
Crane	10,000	3,500	13,500	400
Waste Disposal				
Gaseous	10,000	3,500	13,500	1,400
Liquid	3,000	1,000	4,000	800
Solid	4,000	1,400	5,400	1,100
Laboratory & Equip.	85,000	30,000	115,000	7,700
Preoperational	8,000	2,800	10,800	1,100
Materials Investments	22,000	7,700	29,700	1,000
Total			774,600	97,300
Average Dep'n Rate			12.6%	

60

61

Table 4.2

Annual Cost of Chemical Plants for Processing Periods of 150, 300, and 600 Days

	125 MW Unit			250 MW Unit		
	<u>150 Days</u>	<u>300 Days</u>	<u>600 Days</u>	<u>150 Days</u>	<u>300 Days</u>	<u>600 Days</u>
Fixed Costs	332,000	235,000	184,000	332,000	235,000	184,000
Operating Costs:						
Raw materials	306,000	153,000	76,500	364,000	182,000	91,000
Unit Processes	29,000	14,500	7,300	34,600	17,300	8,700
Services	4,600	2,300	1,200	4,600	2,300	1,200
Labor	52,500	35,000	35,000	52,500	35,000	35,000
Maintenance	<u>68,600</u>	<u>49,400</u>	<u>38,700</u>	<u>68,600</u>	<u>49,400</u>	<u>38,700</u>
Total	792,700	489,200	342,700	856,300	521,000	358,600

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- 4) Indirect costs have been applied to each item.
- 5) A depreciation time to the nearest five years has been assigned to each item.
- 6) An average annual depreciation rate has been calculated.
- 7) Annual fixed costs have been estimated.
- 8) Annual operating costs have been estimated in terms of:
 - a) Raw materials
 - b) Unit processes
 - c) Services
 - d) Labor
 - e) Maintenance
- 9) The cost per gram of U^{233} processed and the cost per kwh of electricity have been calculated.

The capitalization cost of both the 250 MW unit and the 125 MW unit chemical processing plants is presented, as a function of the processing, in Table 4.1. Estimates of specific depreciation times to the nearest five years and a calculated average depreciation rate is provided for each processing period.

Annual costs for the 250 MW unit and the 125 MW unit chemical plants are given in Table 4.2.

Sample capitalization costs for a processing period of 300 days are obtained from assumptions and calculations as follows:

62

Process section:

Cell - 37' x 34' x 22' interior dimensions with 7' thick walls and top, and 3' thick base, two short sides and one long side; 1412 cu. yd. of concrete at \$125/cu. yd. Dimensions are determined by an equipment layout allowing 2' around each piece for piping and special equipment

Equipment - estimated as installed in each case (\$1000/pump)

Special equipment - estimated by comparison with equipment cost.

Piping - total installed equipment cost x 1.8

Instrumentation - total installed equipment cost x 0.77⁽²⁷⁾

Process waste disposal:

Gaseous - refrigerated charcoal bed estimated as unit

Liquid - estimated in comparison with solid waste disposal cost

Solid - sand-filled asphalt pit of 27,000 gal. solid capacity and sump at \$0.30/gal.

Laboratory section:

Building and equipment - at \$75/cu. yd. of cell concrete volume (1412 cu. yd.), giving a total cost of processing structure of \$200/cu. yd.

Pre-operational costs:

An estimate based on size and complexity of the plant

Materials investments:

Estimated as one day's holdup of materials:

Li⁷ 35.7 lb at \$200/lb

63

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BeF ₂	240 lb at \$15/lb
NH ₄ F·HF	290 lb at \$0.50/lb
U ²³³	1.22 Kg at \$25,000/Kg *

Annual costs for the 250 MW unit with a processing period of 300 days are obtained as follows:

Fixed costs:

The fixed charges are subdivided as follows:

9 1/4%	return on investment and income taxes
12 1/2%	average depreciation rate
<u>2%</u>	insurance and local taxes
23 3/4%	Total fixed charges

Operating costs:

Raw materials:

ThF ₄	24,200 lb Th at \$7.50/lb Th
F ₂ , NH ₄ F·HF	costs are negligible

Unit processes:

UF ₆ Reduction	1330 Kg at \$1.30/Kg
NH ₄ F·HF Remake	107,000 lb at \$0.15/lb

Services:

Steam	5,110,000 lb at \$0.20/1000 lb
Electricity	73,000 kwh at \$0.005/kwh
Demineralized water	1,100,000 gal. at \$0.08/100 gal

Labor:

17,500 man-hours at \$2.00/man-hour

Cost of manufacture of U-233 in converter assumed to be 1.25 x cost of U-235 burned in converter. Unclassified minimum value of \$20/gm of U-235⁽⁴⁵⁾ used for this report.

64

65

Maintenance:

Capital investment x 5%(27)

Capitalization costs for the 150 day and 600 day processing periods were estimated similarly as were the annual costs for both units for these processing periods.

Both the capitalization and annual costs vary only with the processing period since the plant size and cost and the operating costs depend upon the volume throughput of fused salt.

The cost of processing per gram of U^{233} throughput and the cost per kwh of electricity naturally differ between the two units because of the different core system loadings of U^{233} and the different electrical outputs respectively. These costs have been calculated from the annual costs and summarized in Table 4.3.

Table 4.3

Electrical Output Costs

	125 MW			250 MW		
Processing period (days)	<u>150</u>	<u>300</u>	<u>600</u>	<u>150</u>	<u>300</u>	<u>600</u>
Cost/gram U^{233} processed	\$1.05	\$1.29	\$1.81	\$0.96	\$1.17	\$1.61
Cost/kwh of electricity (mills)	.72	.45	.31	.39	.25	.16

In order to select a reference processing period it is desirable to estimate the total cost in mills/kwh of chemical processing. This approach evaluates, approximately, the cost of neutrons and of atoms of U^{233} and Pa^{233} lost as a function of the processing period. The same expressions used in the development of loss of breeding ratio vs.

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processing period (section 4.2) for both units are applicable to this cost analysis. An additional term expressing the cost of chemical processing/kwh is included, however. All terms are suitably modified by appropriate constants to convert from loss of breeding ratio to mills/kwh of electricity. The cost of chemical processing per gram of U^{233} throughput, C, is available by interpolation between the calculated costs as shown in Table 4.3. The value of U^{233} , V, is taken as \$25 per gram. The expression of total cost, M, in mills/kwh of electricity is then:

$$M = \frac{233 \times 10^3 V}{3ebA} \left[\frac{y_r \sigma_r(\phi/3) \tau}{\sigma_r(\phi/3) \tau + R_r} + \frac{y_a \sigma_a(\phi/3) \tau}{\sigma_a(\phi/3) \tau + R_a} \right] + \frac{233 \times 10^3 V}{3ebA} \left[\frac{\alpha \sigma_c^{02} \sigma_c^{13} \phi(\tau/\sigma_f^{23})}{\lambda_{13} \tau + \sigma_c^{13} \phi/13 + R_{13}} \right] +$$

(fission product absorptions) (Pa^{233} absorptions)

$$+ \frac{233 \times 10^3 V \alpha \sigma_c^{02} / 13}{2ebA \sigma_f^{23}} \left[\frac{1}{\lambda_{13} \tau + \sigma_c^{13} \phi(\tau/3) + R_{13}} + 1 \right] + \frac{233 \times 10^3 V}{2ebA \sigma_f^{23} \phi \tau} + \frac{233 \times 10^3 C}{ebA \sigma_f^{23} \phi \tau}$$

(Pa^{233} chemical losses) (U^{233} chem. losses) (processing cost)

Fig. 4.7 is a graph of the total cost as a function of processing period for both the 250 MW and 125 MW units. The recovery factor of group "a" fission products is again taken as 0.9. A chemical loss fraction of .001 for U^{233} and Pa^{233} is again assumed.

From the graph, Fig. 4.7, it is apparent that the arbitrarily chosen processing periods of 300 days for the 250 MW unit and 600 days for the 125 MW unit are reasonable. Breeding ratio losses for these periods range between 4% and 5% (see Figs. 4.1 and 4.2). It should be emphasized, however, that these periods do not optimize the reactor core composition

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FIGURE 4.7
TOTAL COST OF CHEMICAL PROCESSING
V.S. PROCESSING PERIOD

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with chemical processing. Instead a breeding ratio loss of approximately 4 1/2% is arbitrarily suffered and the core composition is calculated on this basis.

4.6 Summary

The operating characteristics of the reactor system as determined by the chemical processing are summarized below.

	<u>Plant Output</u>	
	125 MW	250 MW
Type of process	Water dissolution of fused salt for bulk fission product removal; dry fluoride volatilization of volatile fission products; continuous.	
Processing period for bulk of fission products	600 days	300 days
Volatile fission products	13 days	13 days
Pa ²³³	13 days	13 days
Breeding ratio loss to fission products, Pa absorption, chemical processing losses	4%	4 1/2%
Chemical plant capitalization cost	\$ 775,000	\$ 1,000,000
Chemical plant operating cost, annual cost	343,000	521,000
Cost per gram of U ²³³ processed	\$ 1.80	\$ 1.20
Cost per kwh of electricity (100% load factor)	0.31 mills	0.25 mills

69

DEFINITION OF SYMBOLS FOR CHAPTER 4

τ	= processing period - time required to process the entire reactor system
R_{13}	= fraction of Pa^{233} recovered in the chemical process
R_a	= fraction of group "a" removed in the chemical process
R_r	= fraction of group "r" removed in the chemical process
ℓ_c^{23}	= chemical loss fraction of U^{233}
ℓ_c^{13}	= chemical loss fraction of Pa^{233}
ϕ	= neutron flux
σ	= cross section in barns (subscripts and superscripts specify type)
y_r	= fission yield of group "r"
y_a	= fission yield of group "a"
a	= atomic ratio of Th^{232} to U^{233} in reactor system
L	= loss in breeding ratio
M	= total cost of chemical processing (mills/kwh)
e	= thermal efficiency of plant
b	= heat energy produced per fission = 8.96×10^{-18} kwh/fission
C	= cost of chemical processing (dollars/gram of U^{233} processed)
V	= value of U^{233} (dollars/gram)
N_a	= atoms or moles of group "a" in the reactor system
N_r	= atoms or moles of group "r" in the reactor system

70

N_{13} = atoms or moles of Pa^{233} in the reactor system
 N_{23} = atoms or moles of U^{233} in the reactor system
 N_{02} = atoms or moles of Th^{232} in the reactor system
 N_{24} = atoms or moles of U^{234} in the reactor system
 N_{25} = atoms or moles of U^{235} in the reactor system
 N_{26} = atoms or moles of U^{236} in the reactor system

71

Chapter 5POWER PLANT AND HEAT TRANSFER5.1 Power Cycle

Two sizes of plant have been considered. The first is based on the use of a single-shaft turbo-generator having a nominal rating of 100 MW and a maximum capability of 125 MW (115 MW net). The second employs a cross-compound (two-shaft) turbo-generator nominally rated at 200 MW and having a maximum capability of 250 MW (235 MW net), which approaches the size of the largest unit yet built. Both units operate on the same cycle and under the same conditions.

The power capabilities of the plants are shown in Table 5.1. The use of fused salts in the reactor in the temperature range between 1300°F and 1500°F, as discussed in Chap. 2, makes possible the adoption of a power cycle of high efficiency employing steam at high pressure and temperature. The steam conditions chosen are 1450 psi and 1000°F at the throttle, with reheat to 1000°F at 400 psi, and 1 1/2 inches of mercury absolute pressure at the exhaust. Regenerative feed water heating to 440°F is utilized, bringing the thermal efficiency of the turbine cycle proper above 42%. After allowance for the small heat loss from the reactor and for auxiliary power, the heat input from the burn-up of U²³³ is on the order of 8800 Btu per net kwh of electrical output, a net efficiency of 38.1% and 37.1% for the large and small units respectively.

This turbo-generator unit is of a type which is fully developed and is in use today in some of the larger power stations, and in these large

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72

Table 5.1Plant Capability

Turbine Capability	MW	<u>125</u>	<u>250</u>
Net output	MW	115	235
Auxiliary power	MW	<u>10</u>	<u>15</u>
Gross generation	MW	125	250
Cycle efficiency	%	42.1	42.1
Turbine heat input	MW	297	594
Misc. losses at 3%	MW	<u>9</u>	<u>18</u>
Reactor output	MW	306	612
Reactor heat loss	MW	<u>4</u>	<u>4</u>
Fission heat production	MW	310	616
Net efficiency (net output to fission input)	%	37.1	38.1

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DIAGRAM OF SODIUM FLOW
FSBR POWER PLANT
FIG. 5.1

74
sizes the steam conditions selected are believed to entail minimum or near minimum plant investment cost. It is of importance to note that the intermediate fluid which transports heat from the reactor, in this case liquid sodium, not only permits design of the reactor primary heat exchangers to meet very special conditions but also makes possible the use of simple equipment with precise control of both superheat and reheat temperatures, a situation which is in sharp contrast to the problems involved in the superheating and reheating of steam in coal-fired plants.

5.2 Steam Generator and Heat Transport System

The heat transport and transfer system as discussed below is divided into unit system such that four units are required for the 125 MW turbine and eight for the 250 MW turbine, the reactor core and shield and the unit systems being identical for both cases. The operation of the boiler drum, however, is closely associated with the rate of steam production, so that a longer drum will be required for the larger plant. The steam and heat flows are shown in Table 5.2, heat exchanger data are shown in Table 5.3, and reactor pump requirements in Table 5.4.

The discussion which follows covers the plant of 235 MW net capability which has eight unit heat transport systems. Thus the 250 MW steam generator consists of eight parallel groups of heat exchangers connected to a single steam drum 66 inches in diameter and 50 feet or more in length (see Fig. 1.1). In each heat exchanger group there are three separate shells, one for evaporating, one for superheating, and the third for reheating. In each case the heat exchangers are single-pass, counterflow

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Table 5.2

General Flow Data

	<u>Total Flow</u>		* Flow per heat transfer circuit
	<u>125 MW Unit</u>	<u>250 MW Unit</u>	
<u>Steam Flow</u>			
Boiler and Superheater			
Turbine throttle	835,000 lb/hr	1,670,000 lb/hr	
Auxiliaries	8,000 lb/hr	16,000 lb/hr	
Miscellaneous and heat losses in sodium and steam system	17,000	34,000	
Total	860,000 lb/hr	1,720,000 lb/hr	215,000 lb/hr
Reheater	750,000 lb/hr	1,500,000 lb/hr	187,500 lb/hr
<u>Reactor Power</u>			
Boiler	637x10 ⁶ Btu/hr	1274x10 ⁶ Btu/hr	159.0x10 ⁶ Btu/hr
Superheater	282x10 ⁶ Btu/hr	564x10 ⁶ Btu/hr	70.5x10 ⁶ Btu/hr
Reheater	126x10 ⁶ Btu/hr	252x10 ⁶ Btu/hr	31.5x10 ⁶ Btu/hr
Total output	1045x10 ⁶ Btu/hr	2090x10 ⁶ Btu/hr	261.0x10 ⁶ Btu/hr
	306 MW	612 MW	
Reactor heat loss	4 MW	4 MW	
Total reactor input	310 MW	616 MW	
<u>Sodium Flow</u>			
Primary heat exchanger	9,940,000 lb/hr	19,880,000 lb/hr	2,485,000 lb/hr
Boiler	8,680,000 lb/hr	17,340,000 lb/hr	2,170,000 lb/hr
Superheater	6,000,000 lb/hr	12,000,000 lb/hr	1,500,000 lb/hr
Reheater	2,690,000 lb/hr	5,380,000 lb/hr	673,000 lb/hr
Main sodium bypass	1,260,000 lb/hr	2,520,000 lb/hr	315,000 lb/hr
<u>Salt Flow</u>			
Primary heat exchanger	11,760,000 lb/hr	23,520,000 lb/hr	2,940,000 lb/hr

* Four circuits for 125 MW unit, eight circuits for 250 MW unit

76

77

Table 5.3

HEAT EXCHANGER SCHEDULE

Heat Exchanger	Heat Transfer Btu hr x 10 ⁶	Heated Fluid (inside tube fluid)																				Symbols								
		Fluid	Mass Flow lb sec	Temperature		Specific Volume		Volume of Flow		Enthalpy		Tube Size		Number of Tubes	Total Flow Area ft ²	Length of Tubes ft	Total Transfer Surface ft ²	Velocity		Viscosity lb ft-sec	Reynolds Number x 10 ³			Spec. Heat Btu lb-°F	Thermal Cond. Btu hr-ft-°F	Prandtl Number c _p k	Heat Transfer Coefficient Btu hr-ft ² -°F	Friction Factor	Pressure Drop lb in ²	
				In °F	Out °F	In ft ³ lb	Out ft ³ lb	In ft ³ sec	Out ft ³ sec	In Btu lb	Out Btu lb	Outer Dia. Inch	Wall Thick Inch					In ft sec	Out ft sec											
Boiler	159	water Steam mixture	239 59.7 299		605 605	0.024 0.255 0.024	0.024 0.255 0.070	7.2 20.9		422 1163		2.000 0.100		40 0.709	31 647	10.1 29.5														c _p - specific heat D - hydraulic diameter f - friction factor g - 32.2 ft/sec ² k - thermal conductivity l - length Nu - Nusselt number Pe - Prandtl Number Re - Reynolds Number v - velocity - density - viscosity
Superheater	70.5	steam	59.7	605 1000	0.255	0.503	15.2 30.0	1163 1491	0.625 0.049	100 328	100 198	37x10 ⁻⁶	468	0.70 0.095	1.097	¹ 1780	0.016	41.6												
Reheater	31.5	steam	52	686 1000	1.627	2.279	84.6 118.7	1355 1523	1.000 0.065	100 388	204 286	23x10 ⁻⁶	397	0.52 0.045	0.957	¹ 2590	0.017	11.6												
Pri. Heat Exch.	261	sodium	690	850 1200	1/52/5	1/49.5	13.2 13.9 (13.6 ave.)				1.000 0.065	218 0.900	18.1 1034	14.6 15.5 (15.1 ave)	14x10 ⁻⁵	397	0.30 37.7	0.00401	² 8380	0.017	5.3									

Heat Exchanger	Cooled Fluid (outside tube fluid)																				Log Mean Temperature Difference					
	Fluid	Mass Flow lb sec	Temperature		Specific Volume		Volume of Flow		Tube Area ft ²	Shell Area ft ²	Flow Area ft ²	Velocity ft sec	Equivalent Hydraulic Diameter ft	Viscosity lb ft-sec	Reynolds Number x10 ³	Spec. Heat Btu lb-°F	Thermal Cond. Btu hr-ft-°F	Prandtl Number c _p k	Heat Transfer Coefficient Btu hr-ft-°F	Friction Factor			Pressure Drop lb in ²	Tube Wall Resistance Btu hr-ft ² -°F x10 ⁴	Overall Heat Transfer Coefficient	
			In °F	Out °F	In ft ³ lb	Out ft ³ lb	In ft ³ sec	Out ft ³ sec																	Calc. Value	Value Used
Boiler	Sodium	604	1044 800	1/50.8 1/52.8	11.9 (11.6 ave.)	11.4	0.873	1.850 0.977	11.85	0.147	16 x 10 ⁻⁵	567	0.30 40	0.0043	² 5300	0.016	2.5	5.84		800	301					
Superheater	Sodium	417	1200 1044	1/49.5 1/50.8	8.43 (8.33 ave.)	8.22	0.213	1.145 0.932	8.97	0.166	13 x 10 ⁻⁵	572	0.30 37	0.0039	² 3100	0.016	0.84	2.95	850	700	304					
Reheater	Sodium	187	1200 1044	1/49.5 1/50.8	3.78 (3.73 ave.)	3.68	0.545	1.128 0.583	6.43	0.077	13 x 10 ⁻⁵	189.5	0.30 37	0.0039	² 5700	0.020	0.86	3.86	342	300	271					
Pri. Heat Exch.	Salt	816	1484 1316	(1/131 ave.)	(6.22 ave.)			0.823	7.56	0.047	77 x 10 ⁻⁴	6	0.53 1.5	0.102	³ 1440	0.040	12.1	3.62	834	688	367					

1. $N_u = 0.023 R_e^{0.8} P_r^{0.4}$

2. $N_u = 7 + 0.025 P_e^{0.8}$

3. Graphical solution from page 196 of "Heat Transfer" by McAdams.

Source of Data:

Physical Properties - - Steam

Sodium

Fused Salt . See

Specific Heat = 0.53 Btu/lb - °F
Thermal Conductivity = 1.5 Btu/hr-ft-°FEstimated by H. Poppendiek
Private communication

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78

Table 5.4Reactor Pump Requirements

Type Fluid	Pumps	Flow Rate lb/sec	Head psi	Liquid H.P.	Kw Input	Total Input
Salt	8	806	20	32	67	536
Sodium	8	728	40	142	235	1880

These requirements appear to be less than the equivalent power for forced and induced draft fans, pulverizers, etc., for a coal-fired plant.

79

type, with the hot fluid (sodium) on the outside of the tubes. The sodium passes through the superheater and reheater shells in parallel, then the two streams join and pass through the evaporating section.

5.3 Control

Control of steam conditions is effected in the sodium circuits as follows: 1) by-passing the entire group to regulate steam pressure; 2) by-passing the paralleled superheater and reheater to control relative evaporating and superheating; and 3) positioning a valve in the reheater supply to proportion the sodium flow between the superheater and reheater, the latter being designed with less resistance than the former when the flow is properly divided.

5.4 Miscellaneous Requirements

The steam drum must be designed in accordance with the need for feedwater regulation, steam and water separation and distribution, and the limitations of carry-over to the turbine. These requirements dictate the use of a drum as developed for the present-day high pressure, high capacity steam generator; and indeed the entire boiler water circulating system must also be considered an integral part to be fully co-ordinated with the functions of the drum.

The sodium flow to each of the eight heat exchanger groups is individually and separately supplied from one of the eight primary heat exchangers located in the graphite structure of the reactor, a pump being located in the return line from the boiler to the primary heat exchanger. Each sodium circuit contains five main valves, namely, the three regulating

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valves discussed above (which are never fully closed), one shut-off valve to permit draining to the dump tanks in case of emergency or other outage requirement, and one shut-off valve to permit the sodium to be heated while starting without passing it through the steam generating heat exchangers. A number of small valves between helium and sodium lines will also be needed. Four shielded compartments (Fig. 1.2) are provided below the boiler heat exchanger room, and in each there are located the two sodium pumps which serve the two circuits immediately above. In event of serious trouble with any pump, both circuits in that compartment must be drained for decontamination, after which the faulty pump may be cut out of the line and replaced with a spare pump. During such an outage, the electric capacity will be reduced about 25%, but no serious operating difficulties are anticipated. Although the primary heat exchangers are not drainable on the sodium side, the fused salt temperature is limited to a value less than that which will cause the sodium to boil.

Small leakage of water or steam into the sodium in the heat exchangers is not expected to cause serious difficulty. Detection may be accomplished by providing a gas-collecting chamber and offtake in the sodium return line. Build-up of NaOH in the sodium system should not be difficult to follow, and replacement or purification of the sodium can be undertaken as may appear necessary. The influx of large amounts of water or steam, such as would occur from a major tube failure in one of the heat exchangers, would cause dangerous increase in pressure in that particular sodium system. While this possibility seems remote, suitable pressure-relieving devices must nevertheless be provided.

81

It is expected that provision can be made in the reactor reflector to minimize activation of the sodium, so that gamma radiation from the sodium system will be small. It is hoped that under these conditions the decomposition of water in the boiler section will be negligible and that the small quantity of oxygen released will be effectively removed by the scavenging sodium sulfite, which is a standard part of the boiler water treatment at these pressures. It is also to be observed that while the heat exchanger surfaces will be composed of inconel, which is highly corrosion-resistant, the boiler drum and internal parts must be made of the usual carbon steel if prohibitive cost is to be avoided.

When the power load at any level is abruptly reduced to zero, or near zero, it is necessary to remove delayed heat. Since it is impractical to by-pass high temperature steam directly to the condenser, a sodium-to-air heat exchanger with fans is provided in the main sodium by-pass line of two of the sodium systems.

Selection of inconel for use with fused salts and liquid sodium is discussed in section 2.7.

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Chapter 6REACTOR MECHANICAL DESIGN6.1 Reactor Structure

To provide maximum reactor life, the core, primary heat exchangers, salt circulation pumps, and fission gas collection chambers are placed in cavities in a graphite structure. This in turn is enclosed by an inconel tank. This arrangement is shown in Fig. 1.1. Each heat exchanger and associated salt pump is a combined replaceable unit. The hottest moving salt is in contact either with graphite or with replaceable inconel.

By the use of suitable cooling facilities both during reactor operation and shutdown conditions with salt in the reactor, the inconel tank will be cooled to about 1200°F. At this temperature, stagnant salt in contact with the inconel probably will not seriously corrode the tank.

The reactor graphite structure consists of unbonded graphite blocks of shapes similar to those required for stonework cisterns, arches, and domes. This provides stability both when empty and cold and under the buoyancy condition when filled with salt at normal operating temperature. Such shaped blocks would be produced by molding and subsequent machining. Suitable keying of adjacent blocks may be required under the extremes of temperature conditions encountered.

To permit minimum penetration of fissionable and fertile material into the blocks, the reactor first will be filled with salt containing no uranium or thorium. From this standpoint the employment of high density graphite may be desirable.

83

6.2 Heat Generation and Temperature Distribution in the Graphite Reflector

The satisfactory behavior of graphite at elevated temperatures has been observed in many common pyro- and electro-metallurgical practices. Its proven excellent resistance to spalling and thermal shock indicates that these problems should not be serious in this reactor. Order of magnitude estimations and calculations give a qualitative picture of the heat generation and temperature distribution in the graphite and an indication of the thermal stresses induced.

The following model is used in these calculations: consider a horizontal diameter of the spherical reactor passing through a heat exchanger; let the dimensions along this diameter be 5 feet core radius, 1.7 feet (50 cm) of graphite reflector, 2.5 feet thickness of heat exchanger, and 0.8 feet of graphite (these are the nominal dimensions of the reactor). Assume that the amount of heat generated in the 1.7 feet thick reflector shell, mainly due to gamma and neutron attenuation, is 3% of the total energy in the core, or 3% of 600 MW = 18 MW, distributed uniformly throughout the reflector. (This 3% figure is undoubtedly a very liberal one: the total amount of heat generated in the graphite balls surrounding the MTR is 420 kw at 45 MW power level, (42) approximately 1%; the estimated total heat production in the graphite reflector of the NRU reactor is of the order of 0.6 MW at 200 MW power level, (40) about 0.3%.) It was felt that if using this pessimistic value of 3% gave a fairly satisfactory maximum temperature in the graphite, i.e., a maximum temperature appreciably removed from the temperature at which the strength of graphite starts to fall off, approximately 4500°F (2500°C), and also gave some indication of reasonable thermal stresses, there would be justification for

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84

feeling that the graphite will behave satisfactorily at the lower and more probable heat generation in the reflector at operating conditions.

The heat generation per cubic foot, S, in the reflector is

$$S = \frac{\text{total heat}}{\text{reflector volume}} = \frac{0.03 \times 600 \text{ MW}}{\frac{4}{3}\pi(6.7^3 - 5^3) \text{ ft}^3} = 2.44 \times 10^4 \text{ watts/ft}^3$$

$$= 0.585 \times 10^4 \text{ cal/sec-ft}^3 = 0.86 \text{ watts/cm}^3 = 0.205 \text{ cal/sec cm}^3$$

The thermal conductivity of graphite in the 1600-2000°F range varies between the values of 0.07 and 0.18 cal/sec-cm-°C for nominally the same grade of graphite.⁽³⁰⁾ The value of $k = 0.15 \text{ cal/sec cm}^\circ\text{C} = 2.54 \text{ cal/sec ft}^\circ\text{F}$ was chosen. If this were a detailed analysis of the temperature distribution, this range of values of k would make it impossible to pinpoint a true temperature distribution even with the correct heat generation function.

The differential equation, in spherical coordinates, of the temperature distribution is:

$$\frac{d^2T}{dr^2} + \frac{2}{r} \frac{dT}{dr} = - \frac{S}{k},$$

at steady state with the boundary conditions:

- a. Temperature at the core-reflector interface = 1400°F = mean temperature of the fuel, i.e., $T(5 \text{ ft}) = 1400^\circ\text{F}$
- b. Temperature at reflector-heat exchanger core interface, along the horizontal diameter passing through the heat exchanger = 1400°F = $T(6.7 \text{ ft})$.

Solving the above equation by conventional methods, one obtains:

$$T = -Sr^2/6k + A + B/r.$$

85

Since $S/k = 5.85 \times 10^3 / 2.54 = 0.23 \times 10^4 \text{ } ^\circ\text{F}/\text{ft}^2$, this equation becomes:

$$T = -0.384 \times 10^3 r^2 + A + B/r.$$

Applying the boundary conditions, one obtains:

$$T = -0.384 \times 10^3 r^2 + 41.12 \times 10^3 - 150.6 \times 10^3/r$$

for r in feet, T in $^\circ\text{F}$, which is plotted in Fig. 6.1.

The above assumes that the graphite reflector is essentially a solid piece, since no consideration was taken of the heat transfer coefficient between the blocks of graphite. This is not a bad assumption, as far as heat flow is concerned, for measurements have shown that the heat transfer coefficient between two blocks of AGOT graphite under contact pressure of 15.7 psi is about 0.16 cgs units, as compared to about 0.4 cgs units for the thermal conductivity⁽³³⁾, and increases with increase in contact pressure. The contact pressure between the blocks of graphite in this reactor, because of thermal expansion, will certainly be greater than 15 psi.

The calculated temperature distribution has a maximum of 2260°F at $r = 5.8$ feet (see Fig. 6.1). This temperature is far below the maximum allowable temperature of about 4500°F .

6.3 Thermal Stresses in the Graphite

The maximum stress in a heated material is of the order of

$$\sigma_{\max} = \frac{nE \alpha \Delta T}{1 - \alpha \mu} \quad (29)$$

where n has a value depending on the amount of constraint, $0 < n < 1$, generally,

E = Young's modulus

α = coefficient of thermal expansion

327 088

87

ΔT = difference in temperature between two points in the material

μ = Poisson's ratio

a = value which depends on whether constraints exist in one, two, or three coordinate directions

<u>Constraints</u>	<u>a</u>
1	0
2	1
3	2

The values of E and a for graphite vary over a wide range for nominally the same grade of graphite. a ranges between 4×10^{-6} and $10 \times 10^{-6}/^{\circ}\text{C}$, equivalent to 2.2×10^{-6} and $5.4 \times 10^{-6}/^{\circ}\text{F}$, at 2000°F (1100°C). E ranges between 0.4×10^6 and 1.4×10^6 psi at this temperature.⁽³⁰⁾ Using average values of a and E , $3.8 \times 10^{-6}/^{\circ}\text{F}$ and 0.9×10^6 psi respectively, the stress equation becomes:

$$\sigma_{\max} = \frac{n3.8 \times 0.9\Delta T}{1 - a\mu} = \frac{n3.4\Delta T}{1 - a\mu}$$

Poisson's ratio for graphite is about 0.2.⁽³²⁾ If the worst case is considered, i.e., $n = 1$ and $a = 2$ (3 constraints), the above equation becomes:

$$\sigma_{\max} = \frac{3.4\Delta T}{1 - 2 \times 0.2} = 5.7\Delta T$$

For a ΔT of 900°F , $\sigma_{\max}$ is 5200 psi. The compressive strength of AGOT graphite at room temperature is about 5300 psi.⁽³³⁾ Since the strength of graphite increases approximately linearly with increase in temperature to a value, at 4500°F , about twice that at room temperature, the compressive strength of graphite at 2000°F is of the order of 7500 to 8000 psi. The calculated maximum stress is reasonably below the compressive strength of graphite at these temperatures.

88

As mentioned previously, the 18 MW total heat production in the reflector is probably a liberal estimate. Hence, the calculated values of maximum temperature difference and thermal stress are rather pessimistic.

The qualitative analysis shows that no dangerously high temperatures should develop in the reflector during full power operation, and that the thermal stresses should not be excessive. (The thermal stresses can be further reduced by using graphite with lower values of α and E , and higher k .)

Of the 18 MW assumed to be generated in the reflector, approximately 60% will flow into the heat exchanger and about 40% into the core.

6.4 Temperature Distribution in the Graphite Between the Heat Exchanger and the Edge of the Reactor

The graphite in the region between the heat exchanger and the edge of the reactor will be borated. Most of the heat generated in this region will be due to the $B^{10}(n,\alpha)Li^7$ reaction, in which the combined energy of the α and Li^7 particles is 2.5 Mev. The total amount of heat generated is estimated as follows: if it is assumed that 30% of the neutrons in the core get into the reflector and 10% of these get out into the borated region, 3% of the core neutrons will be available for absorption by the boron. Since about one-half of the neutrons in the core produce fission, the amount of energy released per absorption in the core is $\frac{1}{2} \times 200 = 100$ Mev. The total amount of heat generated in this region of graphite is of the order of:

$$0.03 \times \frac{2.5 \text{ Mev/neutron absorbed in borated graphite}}{100 \text{ Mev/neutron absorbed in core}} \times 600 \text{ MW}$$

$$= 0.43 \text{ MW}$$

If the value of 1 MW is used,

$$S = \frac{1 \text{ MW}}{\frac{4\pi}{3}(10^3 - 9.2^3)} = 1.09 \times 10^3 \text{ watts/ft}^3 = 2.61 \times 10^2 \text{ cal/sec-ft}^3$$

$$= 9.2 \times 10^{-3} \text{ cal/cm}^3$$

The differential equation for the temperature distribution is:

$$\frac{d^2 T}{dr^2} + \frac{2}{r} = -\frac{S}{k} = -\frac{2.61 \times 10^2}{2.54} = -1.03 \times 10^2$$

$$k = 0.15 \text{ cal/cm-sec-}^\circ\text{C} = 2.54 \text{ cal/ft-sec-}^\circ\text{F}$$

The solution is

$$T = -17.2 r^2 + C_1 + C_2/r$$

with the boundary conditions:

- a. $T(9.2 \text{ ft}) = 1400^\circ\text{F}$ = temperature at heat exchanger core and graphite interface
- b. $T(10 \text{ ft}) = 1200^\circ\text{F}$ = temperature at graphite and inconel tank interface

Applying the boundary conditions, one obtains:

$$T = -17.2 r^2 + 3.65 \times 10^3 - \frac{7.30 \times 10^3}{r}$$

for r in feet and T in $^\circ\text{F}$, which is plotted in Fig. 6.1.

This amount of heat flowing from the outer surface of the graphite, using this model, is 3.6 MW which consists of the 1 MW heat generated and 2.6 MW from the heat exchanger.

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90

6.5 Boron Burn-up

The burn-up of boron is negligible. If n is the amount of boron present at any time, and N is the original amount, $n = N(1 - \sigma\phi t)$. For "complete" burn-up, $1 - \sigma\phi t = 0$ or $t = \frac{1}{\sigma\phi}$ sec. Assuming that the boron is exposed continuously to a flux of 10^{12} and using the value of 400 barns for the cross section, the burn-up time is about 80 years, much higher than the 10 year life expected for the reactor.

It should be emphasized again that the above calculations of heat generations, temperature distributions, heat flows, and thermal stresses are merely order of magnitude calculations, using pessimistic assumptions.

6.6 Salt-Sodium Heat Exchangers

Each of the primary heat exchangers is approximately 31 inches in diameter and 12 ft. 6 inches long and consists of a tube bundle inserted into a vertical cylindrical hole in the graphite, which acts as a shell. The sodium passes downward through a central pipe and into a ring header, or torus, from which it rises through 218 tubes one inch in diameter wound helically around the center pipe. The tubes terminate in another ring header at the top to which the outlet pipe is connected. All joints are welded, and where tubes are brought into a header on close spacing, new techniques may be required.

The fused salt is forced downward by an axial flow pump driven by a vertical electric motor mounted above the shield. Thus the salt passes from the upper part of the core to the pump inlet, downward in a helical path parallel to and outside of the tubes, and returns through an opening in the

91

graphite into the lower part of the core. The action is counter flow, single pass, and the design establishes conditions for good heat transfer in a limited space without excessive pressure drop. Design details are shown in the heat exchanger schedule, Table 5.3, and the general arrangement is indicated in the cross section of the plant in Fig. 1.1.

All parts of the heat exchangers are made of inconel, and the metal temperature on the salt side is not expected to exceed 1360°F. Since corrosion will occur mainly on the salt side, and since the corrosion products tend to act as an insulator, the metal temperature will probably decrease with time. In spite of this, however, long life cannot be expected⁽³⁴⁾ and it is planned to replace the tube bundles each year. An allowance of approximately 25% has been made in the heat transfer surface for scale formation and other uncertainties, but research and experience will be needed to determine the size which is practical.

One recent inconel loop test,⁽⁵⁾ hot leg at 1500°F and cold leg at 1300°F, with NaF-ZrF₄-UF₄ fuel, has shown that the corrosion depth in 500 hours is about 9 mils and increases to a depth of about 18 mils at 3000 hours with no signs of leveling off.

Removal of the tube bundle-salt pump unit will be accomplished by withdrawal upward along its vertical axis through an opening in the shield provided by a removable plug. The specific method of removal and replacement requires the use of a helium-tight coffin, remote disconnecting of the sodium pipes, and many other details difficult to arrange; but it is believed to be entirely feasible. The most serious possibility of interference in this

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92

process, and which cannot be evaluated at this time, lies in the possible distortion of the graphite sufficient to bind the tube bundle. The designed difference in the diameters of the tube bundle and the opening in the graphite is 0.5 inch.

6.7 Salt Dump Tank System

The inconel dump tank system having a volume slightly in excess of that required for a reactor charge is located at a lower elevation than the reactor structure. Raising the pressure of the blanketing helium above the salt in the dump tanks transfers the salt to the reactor.

Precise control of the salt level in the reactor is secured by control of helium pressure in a small salt supporting surge tank connected to the dump line on the reactor side of the closed salt dump valve. This valve need not be tightly sealed in its closed position provided suitable helium pressure is maintained in the dump tanks.

When containing salt dumped from the reactor following long periods of operation, cooling facilities are required to absorb about 1.2% of the previous reactor heat load. For short storage periods the salt will be held at slightly above reactor mean operation temperature so as to be subcritical when transferred into the core.

For rapid dumping of the salt, the fission gas contaminated helium in the dump tanks, is quickly discharged to the helium receiver. During the dump, a pressure control valve admits helium from the dump tanks to the space above the salt in the reactor to maintain the pressure slightly below atmosphere.

6.8 Fission Gas Direct Collection

Above each of the salt circulating pump sections is a chamber where salt pressure is minimum. At these locations it is believed most of the fission gases will be liberated. Each collection chamber is connected to the off gas system, which aspirates gas consisting mainly of helium, but containing fission gases from the reactor. Helium is admitted to the reactor through the pump shaft seals and by leakage between the top of the reactor tank and the heat exchanger cover flanges. Aspiration is adjusted to hold the pressure slightly below atmosphere during normal operation. Fission gases consequently will not leak outward in the space between the reactor tank and shield.

6.9 Within-Shield Atmosphere

Except during replacement of primary heat exchangers, the helium atmosphere between the shield and the reactor tanks will be maintained at slightly above atmosphere, oxygen free. Thus leakage into the reactor tank will be helium only. If due to some unforeseen circumstance fission gases should escape into this space, the helium pressure will be held slightly below atmosphere by exhausting through induced draft fans to the stack (extent of contamination and outdoor atmospheric conditions permitting) while replacing gas in the within-shield space with fresh helium.

6.10 Points of Salt Withdrawal From and Return to Reactor in Salt Processing

Withdrawal points of salt for processing are at the discharges of two heat exchangers.

94

The return processed salt which contains UF_4 and ThF_4 feed materials will be fed to the suctions of half of the circulating pumps to assure uniform mixing and heating of the salt constituents in the core.

6.11 Reactor Instrumentation and Control Equipment

As indicated in sections 2.8 and 5.5, the negative temperature coefficient of reactivity and automatic adjustment of the flow of sodium which by-pass the sodium-water heat exchangers constitute the main control of reactor heat output. A study of the kinetics of the system may indicate the need of a control rod to provide for more prompt changes in reactivity than will result from the sodium by-pass flow--sodium temperature--salt temperature chain actuated by steam pressure.

At the moment of tripout of the fully loaded turbine, the rate of temperature rise of the salt is $5^{\circ}C$ per second. Thus there will be ample time for the shutdown heat exchangers to be brought into operation to limit the salt temperature and in turn limit the sodium temperature well below its boiling point (Fig. 6.2).

Inasmuch as a detailed study of control and instrumentation is not within the scope of this study in the time available, the estimate of cost is based on that of the MTR and the studies made by the industrial teams. The estimated cost of control instruments for this reactor should be less than any of the four studied. Costs of :\$200,000 for the small plant and \$250,000 for the large plant are chosen.

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FIG. 6.2

Shutdown Fission Product Heat
after Infinite Operation at 614mw

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6.12 Reactor Initial Preheating and Oxygen Removal

Before putting salt into the reactor structure, the structure will be preheated by circulating hot helium through it. This helium will be admitted to the dump line and exhausted from one or more heat exchanger cavities from which the exchangers have been removed. This preheating and oxygen scavenging may require several weeks.

6.13 Initial Salt Charge to the Reactor

Following preheating and oxygen scavenging of the reactor and dump tank system, a charge of lithium and beryllium fluorides will be put in dump tanks and transferred to the reactor. The salt temperature will then be raised to 1200°F by circulation of electrically heated sodium through the heat exchanger and operation of the salt circulation pumps. This temperature will be maintained for several days to permit penetration of fuel- and fertile-material free salt into the faces of the graphite blocks and between the blocks to the inconel tank surface. After this, UF_4 and ThF_4 can be added to attain criticality.

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97

Chapter 7

SHIELDING

7.1 General Description of the Shield

The biological shield surrounds the reactor and consists of sand and concrete. The sides of the reactor are shielded by 8 feet of sand followed by 2.5 feet of ordinary concrete (see Fig. 7.1). The shielding above and below the reactor is concrete. Removable sections are necessary above the reactor to permit the replacement of the reactor heat exchangers. The primary purpose of the graphite is to serve as a container for the fused salts. All the graphite within 50 cm. of the core is reactor grade graphite and this serves as a reflector for the core. Beyond the reflector high purity graphite is not required. By using borated graphite beyond the reflector, this part of the graphite serves as a thermal shield and reduces the activation of the sodium passing through the reactor heat exchangers. The sodium coming out of the reactor heat exchangers is radioactive and requires shielding. Three compartments outside the reactor shield contain sodium and require 2.5 feet of concrete shielding. These are the pump compartment, the heat exchanger compartment, and the compartment above the reactor. Holding the temperature of the inconel container at 1200°F to reduce corrosion requires the removal of 3.6 MW of heat. The leakage heat is removed by dropping the temperature to 200°F through 2 inches of fire brick and by carrying away the heat with a solid wall of water tubes.

7.2 Preliminary Calculations

An accurate knowledge of the flux and of the origin of the flux is

98

Fig. 7.1 Shield

ENCLOSURE 327

necessary before a shield can be designed. A model was set-up to determine the fraction of the total fissions that would occur in the heat exchanger. At first, the heat exchangers around the core were placed within 50 cm. of the core in the graphite. The core considered was 10 feet in diameter with a 50 cm. thick reflector. Assuming a bare reactor with the fuel in the graphite external to the core spread uniformly outside the 50 cm. reflector region, a three region, one velocity calculation was made. The ratio of fissions in the core to the fissions in the heat exchangers was only 2.96. Since the calculations showed such a large source near the outside of the graphite, with a resultant loss in breeding ratio, steps were taken to reduce the fissions in the heat exchanger region. The heat exchangers were moved to provide 106.75 cm. of graphite between the core and the exchangers. In addition, the graphite beyond the 50 cm. reflector region was to be impregnated with a high percentage of boron, thereby reducing the source from the heat exchangers and reducing the sodium activation.

7.3 Calculation of the Flux

To determine the shielding requirements for the reactor a knowledge of the flux distribution in the reflector is necessary. Since the necessary nuclear calculations could not be completed before the shield design was started a one-velocity flux at the core-reflector interface is roughly calculated by the following method. Assuming an average core flux as 10^{14} and the flux form as:

$$\phi = A \frac{\sin(Br)}{r}$$

327

102

100

the value of A is determined from

$$\bar{\phi} = 10^{14} = \frac{\int_{\text{core}} \phi \, dV}{\int_{\text{core}} dV} = A \frac{\int_0^{160} \frac{\sin(Br)}{r} \frac{4\pi r^2 \, dr}{4\pi r^3} \Big|_0^{160}}$$

$$A = 1.29 \times 10^{16}$$

$$B = 1.57 \times 10^{-2}$$

Using this value for A, the flux at the core-reflector interface is calculated to be:

$$\phi = 4.84 \times 10^{13} \, \text{n/cm}^2/\text{sec} \quad (r = 160 \, \text{cm})$$

Assuming a 50 cm. graphite reflector and using a diffusion length in graphite of 50.3 cm., the flux diffusing through the reflector is:

$$\begin{aligned} \phi_{210} &= \phi_{160} e^{-\left(\frac{R_2 - R_1}{L}\right) \frac{R_1}{R_2}} \\ &= 2.2 \times 10^{13} \, \text{n/cm}^2/\text{sec} \quad (r = 210 \, \text{cm}) \end{aligned}$$

Experiments by Clifford, et al, at the X-10 Graphite Reactor show a ratio of 10^{-6} r/h of gamma radiation per neutron/cm/sec at the outer edge of the graphite reflector. Using this ratio and the calculated flux, the gamma radiation dose rate at the outer edge of the reflector for the present reactor is calculated to be 2.2×10^7 r/hr (radius 210 cm.).

7.4 Reactor Shield Calculations

The use of several feet of borated graphite outside the reflector will greatly reduce the neutron flux from the reactor. If we make the pessimistic assumption that the gamma flux at the outer edge of the reflector is due to hard gammas (8 Mev) and calculate the shielding required on that basis the neutron level will be far below tolerance outside the shield. To

327

103

101

attenuate the 8 Mev gammas a mass absorption coefficient of 40 gms/cm² will be used through the shield.⁽³⁹⁾ This figure includes build-up through the shield. Using a density of 1.6 gms/cc for the borated graphite the linear absorption coefficient becomes, .04 cm⁻¹. Assuming spherical geometry and uniform borated graphite from the reflector to the inconel, the dose rate at the inconel is:

$$D = 2.27 \times 10^7 e^{-.04(141)} \left(\frac{210}{351} \right)^2 \text{ r/hr}$$

$$= 2.9 \times 10^4 \text{ r/hr} \quad (r = 351 \text{ cm.})$$

Neglecting the effect of the inconel container, the fire brick layer, and the water pipes, the gamma flux is attenuated through 8 feet of sand. Taking 1.8 gms/cc as the density of sand, the absorption coefficient is .045 cm⁻¹. Thus the dose rate is reduced to:

$$D = 2.9 \times 10^4 e^{-.045(244)} \left(\frac{11.5}{19.5} \right)^2$$

$$= .175 \text{ r/hr} \quad (\text{radius} = 19.5 \text{ ft})$$

The final attenuation is obtained through the concrete necessary for structural purposes and for containing the sand. Using 2.5 gms/cc as the density of ordinary concrete the absorption coefficient is .063 cm⁻¹. The dose rate outside the shield becomes:

$$D = 175 e^{-.063(763)} \left(\frac{19.5}{23.5} \right)^2$$

$$= .12 \text{ mr/hr} \quad (r = 23.5 \text{ ft})$$

This value is much below the 7.5 mr/hr permissible dose rate for an 8 hour work day.

327 104

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102

The graphite below the core is only 2.5 feet thick. Therefore, the attenuation through the borated graphite below the reflector gives:

$$D = 2.2 \times 10^7 e^{-\left(\frac{26.3}{50.3}\right) \left(\frac{210}{236}\right)^2}$$

$$= 1.05 \times 10^7 \text{ r/hr} \quad (r = 236 \text{ cm})$$

The inconel container rests on a concrete cylinder that is cooled with water tubes near the top of the concrete. The fire brick is placed between the inconel and the concrete. Including the 3 foot floor pad, the concrete below the reactor is 9 feet thick. The gamma flux below the reactor gives a dose rate of 750 mr/hr. This is low enough to insure against changes in the foundation below the reactor due to gamma heating.

The graphite above the core is about 9 feet thick. Thus the radiation reaching the inconel will be lower than that at the sides of the reactor.

$$D = 2.27 \times 10^7 e^{-.04(243) \left(\frac{210}{453}\right)^2}$$

$$= 3.1 \times 10^2 \text{ r/hr} \quad (\text{radius} = 453 \text{ cm.})$$

Above the reactor is a large section for handling the radioactive heat exchangers. Sodium pipes go through this region and the gamma radiation is high. The radiation from the reactor need not be attenuated below the radiation in the compartment from the sodium. Using 6 feet of concrete the gamma flux gives:

$$D = 3.1 \times 10^2 e^{-.063(153) \left(\frac{453}{605}\right)^2} \text{ r/hr}$$

$$= 12.4 \text{ mr/hr} \quad (\text{radius} = 605 \text{ cm})$$

Sections of the concrete above the reactor must be removable to change the heat exchangers (see Chap. 6). In addition to this, the concrete must serve to apply pressure to the inconel. The proposal is to use a thin slab of iron above the fire bricks to distribute the load evenly. Placed on the iron plate will be cooling tubes. Between the tubes will be placed iron spacers to transmit the load of the concrete to the iron plate. About 800 lbs/sq ft are required to overcome buoyancy effects from the core. Four feet of the concrete directly above the reactor will be made up of concrete blocks. The remaining two feet will be composed of large removable sections that rest on projections from the wall structure (see Fig. 7.1).

7.5 Thermal Insulation and Heat Removal

Calculations concerning gamma heating and thermal stresses are found in Chap. 6. From these calculations the heat leakage from the graphite is 3.6 megawatts. This heat is to be removed by water flow in iron pipes outside the inconel container. The inconel is to be maintained at 1200°F in order that the corrosion not be excessive. A thermal insulator is required between the inconel and the water tubes to hold the temperature of the inconel at 1200°F while keeping the maximum temperature at the water tubes below 200°F. Fire clay in brick form is proposed for the insulator. The thickness required is found from;

104

$$Q = K A \frac{\Delta t}{\Delta x}$$

$$Q = 3.6 \text{ megawatts } (1.23 \times 10^7 \text{ Btu/hr})$$

$$A = 2,420 \text{ sq ft}$$

$$\Delta t = 1000^\circ\text{F}$$

$$K = .75 \text{ Btu/hr/sq ft}/^\circ\text{F/ft}$$

$$\Delta x = \frac{.75 \times 2420 \times 1000 \times 12''}{1.23 \times 10^7}$$

$$= 1.95 \text{ inches}$$

The water pipes are arranged around the outside of the fire bricks in a solid bank and are held in place by iron bands around the outside. Pipes are welded to the iron bands at a nominal spacing. Assuming a 50°F rise in the water the amount of water needed is:

$$W = \frac{Q}{C_p \Delta t} = \frac{1.23 \times 10^7 \text{ Btu/hr}}{1 \text{ Btu/lb}^\circ\text{F} \times 50^\circ\text{F} \times 8.34 \text{ lb/gal}} = 2.95 \times 10^4 \text{ gal/hr}$$

The flux in the region of the cooling tubes is low enough that the pipes coming out of the shield need not be shielded.

7.6 Sodium Activation

The sodium passing through the heat exchangers in the graphite becomes radioactive due to the neutron flux in that region. The shielding required around the sodium lines which emerge from the reactor shield depends on the flux the sodium sees and the time the sodium remains in the flux. The time the sodium spends in the reactor is fixed by heat transfer considerations. Shielding for the sodium lines can be reduced by lowering the flux at the heat exchangers.

The flux at the heat exchangers comes from radiation from the core, delayed neutrons in the heat exchanger region, and $\text{Be}(\gamma, n)$ reactions in the

fused salt. The ratio of the neutrons born in the heat exchanger to the total neutrons in the system is found to be .0004. This figure is found by taking the ratio of the fused salts in the heat exchanger (for the 616 MW reactor) to the fused salt in the whole system, and multiplying by the fraction of neutrons from $\text{Be}(\gamma, n)$ reactions (.0003 calculated in section 3.8), plus all the delayed neutrons from the 22 second and 55 second half-life neutrons, (.0008).⁽¹⁹⁾ Many of the neutrons born in the exchanger region will escape into the graphite or be absorbed by the inconel rather than by the sodium. Since the fraction of neutrons that are born in the heat exchangers is so small, calculations are made assuming the activating flux comes from the core. Since the graphite further than 50 cm. from the core is essentially a thermal shield region, it can contain a large percentage of boron. A measure of the activation of sodium from a fission source behind a boron containing shield can be made from comparison with ORNL Lid Tank data.⁽³⁸⁾

The Lid Tank data gives activation of sodium versus thickness of B_4C (Fig. 7.2). Correcting the relaxation length for the change in density for graphite and using a ratio of the leakage flux the following formula is used:

$$A = A_L \frac{P}{P_L} \frac{L}{L_L} \left[e^{-\mu (R_2 - R_1)} \right] \left(\frac{R_1}{R_2} \right) \left(\frac{R_2}{R_3} \right)^2 \frac{t}{T}$$

A_L = saturated activity of sodium behind B_4C

P = power of fused salt reactor (800 MW)

P_L = power of Lid Tank source (6 watts)⁽³⁶⁾

L_L = Lid tank leakage (.25)

107

$$e^{-\mu \left(\frac{R_2 - R_1}{R_2} \right) \left(\frac{R_1}{R_2} \right)} = \text{attenuation of flux from core to borated graphite region}$$

$$\left(\frac{R_2}{R_3} \right)^2 = \text{Allard's correction for saturated activity when sodium is passed through a flux where } t' \text{ and } T \text{ are both short times. (41)}$$

t' = time in the reactor in seconds (2 sec)

T = time for complete cycle in seconds (6 sec)

L = leakage from fused salt reactor ($1 - P_{NL}$)

$$P_{NL} = \frac{e^{-B^2 \tau}}{1 + L^2 B^2} = \frac{e^{-\left(\frac{\pi}{160} \right)^2 (457)}}{1 + 66 \left(\frac{\pi}{160} \right)^2} = .816$$

$$L = .184$$

Assuming the flux activating the sodium is the flux at the circle 20 cm. inside the circle of the centers of the heat exchangers, the flux is attenuated through more than 60 cm. of borated graphite. The activity found is:

$$A = .1 \frac{8 \times 10^8}{6} \cdot \frac{.184}{.25} \left[e^{-\left(\frac{50}{50.3} \right)} \right] \left(\frac{160}{210} \right) \left(\frac{210}{270} \right)^2 \cdot \frac{2}{6}$$

$$= 5.6 \times 10^5 \text{ disintegrations/gm/sec}$$

Converting the activity for use with nomograms gives $A = .046$ curies/gallon

7.7 Shielding of Sodium Piping

The nomograms in the Shielding Symposium Notes are used to calculate the required shielding. (37)

The first region to consider is the heat exchanger handling compartment. There are 16 pipes with an average length of 16 feet in the

108

compartment. Each pipe is 1 foot in diameter. Assuming a sidewise cylindrical source with all pipes at the center of the compartment, a 2.5 foot thickness is required to give a dose rate outside the shield of .5 mr/hr.

The second region to consider is the heat exchanger compartment. To use the nomograms a sidewise cylindrical source is used for each complete circuit and all eight circuits are assumed to be in one place 4 feet from the wall. Each source is assumed to be 40 feet high and contain 224 cubic feet of sodium. Again using the 2.5 feet thickness for the shield the radiation outside the shield is .6 mr/hr. The floor of the compartment will be 3 feet thick so the radiation below the floor is much less.

The third region to consider is the pump compartment. The total dose inside one cell, if the sodium is drained from that cell, comes from the heat exchanger room, the reactor, and the adjacent compartments. The dose-rate from the heat exchanger is less than .6 mr/hr and from the reactor the dose is .12 mr/hr. Allowing .1 mr/hr from the adjacent compartments the total dose received is less than .8 mr/hr. This excludes residual activity in the pumps during a time when a pump is being replaced. The source as seen in adjacent compartments consists of lengths of pipe each 1 foot in diameter. A worker in a compartment is exposed to four such sources. To give a dose of .04 mr/hr from each source, a thickness of 3 feet of concrete are required. On the outside walls 2.5 feet are sufficient.

7.8 Breakdown of Shield Cost

The following costs were estimated from the amounts of materials necessary for the complete shielding of the 616 megawatt reactor and sodium

109

compartments. The figures include labor but not overhead.

<u>Material</u>	<u>Amount</u>	<u>Cost Installed</u>
Sand	893 cu yds	\$ 13,395.00
Concrete - Pads	377 cu yds	11,310.00
Concrete - Raised	2,812 cu yds	146,600.00
Cooling tubes and headers		10,000.00
Pumps	4	4,000.00
Heat exchangers	2	5,000.00
Contingency	40%	<u>73,600.00</u>
Total		\$ 257,905.00

Part of this total will be charged off on a 10 year basis. The amount of material that would have to be replaced if only enough shielding is removed to rebuild the reactor is \$45,960.00

The reduction in shielding cost for the 310 megawatt reactor is only \$49,000.00. The only saving realized is a smaller amount of concrete required for the heat exchanger compartment and the pump compartment.

327
112

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Chapter 8ECONOMIC ANALYSIS8.1 Cost Summary

Capital costs and operating costs for the 115 MW and 235 MW reactor power plants are shown in Tables 8.1 and 8.2, together with comparative costs for coal-fired plants of the same capacities. In order to show the effect of capital costs a load factor of 80% has been used, the resulting generation being 7000 kwh per year per kw of capability. The usual experience for many power plants, however, has been to operate at a load factor well above 80% for several years, after which the load factor gradually decreases and approaches zero in 25 to 30 years. Since the turbine cycle in this case is of high efficiency and since the reactor is expected to be replaced with improved equipment at the end of 10 years, it is quite possible that operation at high load factors may continue indefinitely, that is, if the system consists principally of coal-fired plants. If, however, all newer capacity added to the system utilizes reactors, then the reactor discussed herein will operate on a load factor which will decrease as the new (and better) reactor plants are put into operation.

8.2 Fixed Charges

The fixed charges are derived as shown in Table 8.3. Income taxes are assumed to be 50%, and depreciation is taken on a straight line basis with different estimated lives assigned to the various components of the plant as seems proper. There is much experience on the part of the public utilities to indicate expected lives of 30 years and longer for plants of the

Table 8.1

111

Summary of Estimates of Investment Costs

	<u>115 MW</u>	<u>235 MW</u>
Net plant capability		
Turbine and general plant	\$18,800,000	\$35,000,000
Steam generator equipment	1,810,000	3,220,000
Reactor and appurtenances	2,660,000	2,850,000
Chemical process equipment	800,000	1,000,000
Land and improvements (excluding Safety Area)	100,000	100,000
Fused salt inventory	<u>11,680,000</u>	<u>14,070,000</u>
Total	\$35,850,000	\$56,240,000
Investment per kw	\$312	\$239

BREAKDOWN OF ESTIMATES OF INVESTMENT COSTS

	<u>115 MW</u>	<u>235 MW</u>
Turbine and General Plant		
Turbo-generator, condenser, intake and discharge tunnels, auxiliaries, instruments, switchyard, trans- formers, office section, contingencies, engineering and overheads	\$18,800,000	\$35,000,000
Steam Generator Equipment		
Drum and internals	600,000	1,000,000
Evaporating sections	80,000	160,000
Superheaters	24,000	48,000
Reheaters	40,000	80,000
Circulating tubes, saturated steam piping, etc.	25,000	50,000
Auxiliary steam source	10,000	10,000
Water demineralization equipment	50,000	75,000
Inconel piping	85,000	169,000

327
114

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112

Table 8.1 (cont.)

	115 MW	235 MW
Welding, supports and erection	\$ 13,000	\$ 26,000
Surface heaters and control	36,000	72,000
Insulation	6,000	12,000
Sodium valves	33,000	66,000
Na pumps, motors and controls	100,000	200,000
Na dump and surge tanks, heaters, insulation, and controls	66,000	132,000
Shutdown heat exchanger equipment	20,000	35,000
Sodium	13,000	26,000
Secondary shield	80,000	129,000
Reactor shield cooling equipment	9,000	9,000
	<hr/> \$ 1,290,000	<hr/> \$ 2,299,000
Contingencies, engineering and overheads (40%)	517,000	923,000
Total	<hr/> \$ 1,807,000 (1,810,000)	<hr/> \$ 3,222,000 (3,220,000)
Reactor and Appurtenances		
Graphite	\$ 543,000	\$ 519,000
Inconel tank	71,000	71,000
Primary heat exchangers	80,000	160,000
Salt pumps	10,000	20,000
Remote primary heat exchanger replacement equipment	250,000	250,000
Reactor room crane	25,000	25,000
Ventilating fans and ducts	50,000	50,000
500 ft stack	500,000	500,000
Reactor instruments and control equipment	200,000	250,000
Helium supply equipment	15,000	20,000
Reactor building	50,000	50,000
Dump tanks with cooling coils	26,000	31,000
Tank heaters and control	9,000	11,000
Insulation	2,000	2,000
Salt-Na-Air cooling system	7,000	12,000
Salt dump valve, piping and surge tank	5,000	5,000
Helium receiver	4,000	4,000
Helium reclamation compressor	5,000	5,000
Reactor - concrete pad	5,000	5,000
- formed concrete	18,000	18,000
- sand	14,000	14,000

113

Table 8.1 (cont.)

	115 MW	235 MW
Reactor - cooling tubes	\$ 10,000	\$ 10,000
	\$ 1,899,000	\$ 2,032,000
Contingencies, engineering and overheads (40%)	761,000	814,000
Total	\$ 2,660,000	\$ 2,846,000 (2,850,000)
Chemical process equipment including contingencies, engineering, and overheads (for details, see section 4.5)	\$ 800,000	\$ 1,000,000
Land and improvements including contingencies, engineering and overheads, but excluding Safety Area	\$ 100,000	\$ 100,000
Fused Salt Inventory	115 MW	235 MW
U-233 at \$25 per gm as UF ₄ *	\$ 9,500,000	\$ 11,500,000
Li ⁷ at \$200 per lb as LiF ₄ *	1,870,000	2,206,000
Be at \$15 per lb as BeF ₂	182,000	214,000
Th at \$7.50 per lb as ThF ₄	128,000	151,000
	\$ 11,680,000	\$ 14,071,000
Contingencies, engineering and overheads	0	0
Total	\$ 11,680,000	\$ 14,071,000 (14,070,000)

* See Table 8.2 for effect upon cost per kwh of different than assumed prices of U-233 and Li⁷. See page 63 for basis for U-233 cost.

327 116

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114

Table 8.2

Summary of Estimates of Fixed and Operating Costs

	<u>115 MW Plant</u>		<u>235 MW Plant</u>	
	<u>Reactor</u>	<u>Coal</u>	<u>Reactor</u>	<u>Coal</u>
Estimated Investment				
Total	\$ 35,850,000	\$ 21,800,000	\$ 56,240,000	\$ 42,300,000
Per kw	\$ 312	\$ 190	\$ 239	\$ 180
Fixed charges, total				
interest at 9.25%	\$ 3,320,000	2,020,000	5,210,000	3,910,000
depreciation				
at 3.06-3.15%	1,100,000		1,770,000	
at 3.33%		730,000		1,410,000
Taxes and insurance				
at 2.00%	720,000	430,000	1,120,000	850,000
	\$ 5,140,000	\$ 3,180,000	\$ 8,100,000	\$ 6,170,000
Fixed charges, per kw-yr	\$ 44.70	\$ 27.70	\$ 34.47	\$ 26.30
Fixed charges, mills per kwh				
(80% load factor = 7000 kw/yr)	6.4	4.0	4.9	3.8
Production costs, mills/kwh				
Fuel	-	*2.8	-	*2.8
Labor and supervision	0.8	0.6	0.8	0.6
Maintenance	0.6	0.4	0.6	0.4
Chemical process	0.4	-	0.3	-
Total	1.8	3.8	1.7	3.8
Total Cost, production -				
Fixed charges, mills/kwh	8.2	7.8	6.6	7.6
Increase in total cost per				
\$100/ kg of Li ⁷ mills/kwh	1.2	-	0.6	-
Increase in total cost per				
\$10/gm of U-233 mills/kwh	4.7	-	2.8	-

*Coal at 25¢ per million Btu.

327

117

115

Table 8.3

Estimate of Fixed Charge RatesInterest Computation

<u>Capitalization</u>	<u>Net Rate of Return</u>	<u>Amount of Return</u>	<u>Inc. Tax at 50%</u>	<u>Total Required</u>	<u>Gross Rate of Return</u>
Bonds : : \$.50	3.5%	1.75¢	-	1.75¢	
Preferred Stock .15	4.5	0.68	0.67¢	1.35	
Common Stock .35	8.8	3.07	3.08	6.15	
Total \$ 1.00	5.5%	5.50¢	3.75¢	9.25¢	= 9.25%

Average Depreciation

	<u>Est. Life</u> <u>yr</u>	<u>115 MW Net Capability</u>		<u>235 MW Net Capability</u>	
		<u>Invest.</u> <u>per kw</u>	<u>Deprec.</u> <u>per kw</u> <u>per yr</u>	<u>Invest.</u> <u>per kw</u>	<u>Deprec.</u> <u>per kw</u> <u>per yr</u>
Turbine and general plant	30	\$ 163.50	\$ 5.45	\$ 149.00	\$ 5.00
Steam Generator Eq.	30	10.10	0.34	6.00	0.20
Reactor and Appurtenances (12 avg)	10-30	32.40	2.90	23.40	1.90
Chem. process Eq. (8 avg)	5-30	8.00	1.00	4.30	0.55
Land and Improvements		0.90	-	0.50	-
Fused salt inventory		101.50	-	59.80	-
Total		\$ 316.40	\$ 9.69 3.06%	\$ 243.00	\$ 7.65 3.15%

Summary of Fixed Charges

	<u>Reactor Plant</u>		<u>Coal Fired Plant</u>
	<u>115 MW</u>	<u>235 MW</u>	
Interest	9.25%	9.25%	9.25%
Depreciation	3.06	3.15	3.33
*Property taxes and insurance	2.00	2.00	2.00
Total	14.3%	14.4%	14.58%

* In certain states such as Pennsylvania, there is at present no local property tax on utility plants, leaving only a negligible insurance cost of 0.1 to 0.5%

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type built in the past. On the other hand the life of the reactor must be assumed to be much less, with obsolescence (especially in the chemical process) perhaps being as important as physical deterioration, and a life of 10 years has been assumed. For this reason land space is provided for the construction of a second reactor to be ready for service at the end of the tenth year. The situation at the end of 20 years scarcely can be anticipated, but for the sake of continuity it is presumed that a third reactor (if needed) can then be built on the site of the first. Only in this way can the large investment in the longer-lived equipment (\$157 per kw for the 235 MW plant) be utilized economically.

8.3 General Discussion

If the total of the production cost plus fixed charges for the reactor power plant is compared to that for an equivalent coal-fired plant as in Table 8.2 the reactor plant does not appear unfavorable, especially in the large size (235 MW net capability). However, it must be realized that uncertainties which appear in this study may, when resolved, change either the investment or the operating costs; and such changes may well be upward in spite of efforts to be conservative at this time.

Reliability is very difficult to judge when new problems of such magnitude are involved. The design, nevertheless, is basically simple, and many of the components appear to offer probability of excellent service. The reactor heat exchangers and salt circulating pumps are not in this class, however, and much prior development will be required before the risk for

327
119

117

important use of that equipment can be undertaken.

8.4 Inventory Material Prices

Attention is called to the fact that the prices of U^{233} and Li^7 are in effect postulated, leaving the possibility of noticeable error in the inventory and in the kwh costs. At the bottom of Table 8.2 rates are given for computing the change in kwh cost for deviations in prices of these non-depreciating inventory elements.

327 120

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118
Chapter 9

RECOMMENDED RESEARCH

9.1 Summary

In the design of the reactor, a number of factors that directly affect the feasibility of the design have not been measured, and are estimated in the calculations. Many more factors have had only rough measurement, and since these are the only values available, these values have been used in the design. If a fused salt breeder reactor is to be constructed for use in a power plant, a large research program, designed to measure these unknowns more accurately, must precede design.

Since there are so many unknown, or only roughly known, factors in the design, it has been a necessary part of this study to determine the research and development that needs to be done so that a final decision on the feasibility of the design can be made. This chapter lists the recommendations for further research that proceed from this study.

9.2 Nuclear Problems

a) α , the capture to fission ratio, for U^{233} has been measured only for thermal neutrons. The value probably is different in the resonance region, and since the reactor has a large number of resonance absorptions, this will affect the breeding ratio very strongly. The measurement of α for U^{233} is one of the more vital research problems needed to determine the feasibility of not only fused fluoride reactors, but of all reactors which use the U^{233} -Th breeding system.

b) The resonance integral of U^{233} has not been measured, and calculated values from cross section measurements differ considerably. Measurements of both the fission and absorption integrals as functions of energy and concentration would be very useful. Again, this is a problem faced not only by this reactor design, but by all U^{233} reactors.

c) The resonance integral of thorium has been measured as a function of concentration, but not as a function of energy. There is a large discrepancy between the measured values of resonance integral and the measured cross section values. This should be resolved by measuring the resonance integral as a function of energy.

d) The fission yield curve of U^{233} is only sketchily known, so that the equilibrium fission products in the reactor have had to be estimated. Better measurements of the fission yield are very important for designing the chemical process and for determining the possibility of unity breeding ratio.

e) The neutron cross sections of U^{233} and Th^{232} have been measured in the resonance region, but better measurements would be useful. The cross section of Pa^{233} has not been measured in the resonance region. Since the chemical process and breeding ratio depend on this, it should be measured.

f) Fission product cross sections are known at thermal energies, although the resonance cross sections are not known. A measurement of the resonance integral of the bulk of the fission products would be very useful in calculating a resonance reactor, and in determining the amount of chemical processing required.

120

g) A measurement of the variation of the thorium resonance integral with temperature, for a concentrated amount of thorium (see section 3.4 and Appendix B) would be a useful check of the calculation of the temperature coefficient of reactivity. This should also be done for U^{233} .

h) High energy neutron cross sections in general are poorly known.

In particular, fission source measurements should be made for the following reactions: $Be(n,2n)$, $Li^7(n,2n)$, $Be(n,\alpha)$, $F(n,\alpha)$. These reactions have an appreciable affect on this reactor.

i) Rough measurements have been made for the $Be(\gamma,n)$ reaction in a reactor. Particular measurements for a $LiF - BeF_2$ fused salt should be made in the central flux of a reactor.

9.3 Engineering Problems

a) No measurements of the corrosion of graphite in fused salts have been made, although fused salts are contained by graphite in phase diagram studies. Specific corrosion tests should be made, both in reactor flux and out, for temperatures between 1000°F and 2000°F.

b) Corrosion of inconel in fused salts has received much attention, but specific tests near 1500°F have not been made for the $LiF - BeF_2 - UF_4 - ThF_4$ system.

c) The viscosity, density, specific heat, and thermal conductivity of the fused salt used in this reactor should be measured as a function of temperature and composition.

d) The scale coefficients of fused salts on inconel and of sodium on inconel have not been measured. These are very important in calculating heat exchanger sizes, not only for this reactor, but also for other fused salt reactors.

e) Methods for the detection of leaks of water into the sodium system, and of sodium into the fused salt system, must be developed. Tests to determine the effects of small leaks are of great importance.

f) The design of axial flow pumps, bearings, shaft seals, and impellers for fused salts needs considerable study and experimental testing.

g) The distortion of graphite structures with the variation of load, temperature, temperature gradient, and radiation should be tested experimentally. The close fit required between the heat exchangers and the graphite, and the criticality of the reactor, require that any distortion of the graphite be small.

h) The penetration of the fused salt into graphite, and the diffusion of the fuel into graphite, should be small if the reactor is to operate as designed. Measurement of these factors is therefore important.

i) Graphite may not be the best container for fused salts, so that a research program should be aimed at development of better high temperature metals and ceramics for use with fused salts.

327 124

9.4 Chemical Processing Problems

- a) The vapor pressure of PaF_5 in fused salt systems as a function of temperature is not known. This is critically important to the design of the chemical processing plant, and an important factor in obtaining a breeding ratio of unity.
- b) Removal of xenon and other fission product gases from a fused salt may be difficult. The removal rate as a function of temperature, pressure, and composition of the salt should be investigated.
- c) New methods of processing fused salts to remove fission products need considerable investigation. A chemical process which removes fission products directly from the melt would reduce chemical processing costs considerably. Some methods that should be investigated experimentally are: electrolysis of the fused salt melt, extraction of fission products from fused salts by contacting the melt with liquid metal, direct precipitation from the melt by the addition of a reagent, and high temperature selective distillation.
- d) A pilot plant check on the chemical process used in this report would give useful information for processing fused salts in general.

Appendix ATHORIUM RESONANCE CROSS SECTIONS

In the process of setting up the cross sections of thorium as a function of energy for use in the multi-group calculations, it was found that there is no agreement between measured values of the resonance cross section^(9,16) and measurements of the resonance integral.⁽¹⁷⁾ In the analysis of the data, it was felt that the resonance integral measurements are more reliable. However, the integral has not been measured as a function of energy, so the following calculation is used to obtain the energy dependence.

The measured value of the thorium resonance integral is:

$$\int_{.4 \text{ ev}}^{E_{\text{fission}}} \sigma_{a0} \frac{dE}{E} = 81.3 \text{ barns}$$

The lowest energy resonance peak has been observed at 23 ev, and higher energy peaks seem to be spaced about 45 ev apart.^(9,16) Although these measured peaks are apparently double, the Breit-Wigner one level formula is used in the analysis.⁽¹²⁾

$$\sigma \approx \frac{\lambda^2}{4\pi} \frac{\Gamma_n \Gamma_\gamma}{(E - E_r)^2 + \frac{\Gamma^2}{4}} \quad \text{and} \quad \sigma_{\text{max}} = \frac{\lambda^2}{\pi} \frac{\Gamma_n \Gamma_\gamma}{2}$$

$$\Gamma = \Gamma_n + \Gamma_\gamma \approx \Gamma_\gamma + \Gamma_{\text{Doppler}}$$

$$\Gamma = (.1 + .0216 E^{(1/2)}) \text{ ev at room temperature}$$

$$\text{since } \lambda^2 \sim \frac{1}{E}$$

$$\sigma_{\text{max}} \sim \frac{1}{E} \times \frac{E^{(1/2)}}{(.1 + .0216 E^{(1/2)})^2} \sim \frac{1}{E^{3/2} (E^{1/2} + 4.63)^2}$$

124

Assuming a square shape for the resonance peak, the resonance integral over the peak can be written

$$\int_{\text{Peak}} \sigma_{a_0} \frac{dE}{E} \approx \frac{\sigma_{\text{max}} \Gamma}{E} \sim \frac{1}{E^{3/2} (E^{1/2} + 4.63)}$$

This is now fitted to the measured resonance integral. The results are:

Integrated over cross section measurements	<u>E (ev)</u>	<u>$\int \sigma_{a_0} \frac{dE}{E}$ (barns)</u>
	0.4 - 20.	5.4
Peak 1	23	56.1
Peak 2	58	8.1
Peak 3	113	3.2
All higher Peaks	150 - ∞	8.5
<hr/> Total		81.3

As a check on these calculations, the variation of the effective resonance integral as a function of concentration was computed, using these results. The calculated effective integral agreed fairly well with the measured values,⁽¹⁷⁾ but differed sufficiently that the fitting process was used for each concentration.

Appendix BCHANGE OF RESONANCE INTEGRAL WITH TEMPERATURE

At high temperatures, a peak in the neutron absorption curve for an isotope is broadened because of the so-called Doppler Effect. For small concentrations of absorber, or for small peaks, this effect does not change the resonance integral, ⁽⁸⁾ i.e.,

$$\int_{\text{Peak}} \sigma_{a0} \frac{dE}{E} \quad \text{does not change with temperature.} \quad \text{However, for larger}$$

concentrations of absorber, the effective resonance integral is used for calculating absorptions, and this does change with temperature. The effective resonance integral is defined by

$$\text{ERI} = \int_{\text{Peak}} \sigma_a \text{ eff } \frac{dE}{E} = \int_{\text{Peak}} \frac{\sigma_{a0}}{1 + N \frac{\sigma_{a0}}{\Sigma_s}}$$

Consider the integral over a single resonance peak. If the approximation is made that the peak has a square shape, the calculations are simplified, and for a sharp peak this is not too bad an assumption. Then

$$\text{ERI} \approx \frac{\sigma_{a \text{ max}}}{1 + \frac{N \sigma_{a \text{ max}}}{\Sigma_s}} \times \frac{\Gamma}{E}$$

Where Γ is the width of the peak at half the maximum cross section. For a change in temperature the ratio $\frac{N}{\Sigma_s}$ is constant, but both $\sigma_{a \text{ max}}$ and Γ vary.

327 128

126

$$d [ERI] = \frac{d \left[\frac{\sigma_{a \max} \Gamma}{E} \right]}{1 + \frac{N \sigma_{a \max}}{\Sigma_s}} - \frac{\frac{\sigma_{a \max} \Gamma}{E} \times \frac{N}{\Sigma_s} d [\sigma_{a \max}]}{\left[1 + \frac{N \sigma_{a \max}}{\Sigma_s} \right]^2}$$

Since $\int \sigma_{a0} \frac{dE}{E}$ is constant in temperature,

$$d \left[\frac{\sigma_{a \max} \Gamma}{E} \right] = 0$$

$$\text{and } d [\sigma_{a \max}] = - \sigma_{a \max} \frac{d(\Gamma)}{\Gamma}$$

so that

$$d [ERI] = + \frac{\frac{\sigma_{a \max} \Gamma}{E} \times \frac{N}{\Sigma_s} \times \sigma_{a \max} \frac{d(\Gamma)}{\Gamma}}{\left[1 + \frac{N \sigma_{a \max}}{\Sigma_s} \right]^2}$$

For heavy elements and for fairly low neutron energies, the total peak width is the sum of the gamma ray width and the Doppler width. (12)

$$\begin{aligned} \Gamma &= \Gamma_{\gamma} + \Gamma_{\text{Doppler}} \\ &= \text{Constant} + 2 \left(\frac{m E k T}{M} \right)^{1/2} \end{aligned}$$

$$\text{Then } d(\Gamma) = 2 \left(\frac{m E k}{M} \right)^{1/2} \cdot \frac{dT}{2T^{1/2}} = \Gamma_{\text{Doppler}} \frac{dT}{2T}$$

so that the change in the integral becomes.

$$d [ERI] = + \frac{\sigma_{a \max} \frac{\Gamma}{E}}{\left[1 + \frac{N \sigma_{a \max}}{\Sigma_s} \right]^2} \times \frac{N \sigma_{a \max}}{\Sigma_s} \times \frac{\Gamma_{\text{Doppler}}}{\Gamma} \times \frac{dT}{2T}$$

$$\text{or } \frac{d \left[\int_{\text{Peak}} \sigma_{a \text{ eff}} \frac{dE}{E} \right]}{\int_{\text{Peak}} \sigma_{a \text{ eff}} \frac{dE}{E}} = + \frac{\frac{N \sigma_{a \max}}{\Sigma_s}}{1 + \frac{N \sigma_{a \max}}{\Sigma_s}} \times \frac{\Gamma_{\text{Doppler}}}{\Gamma} \times \frac{dT}{2T}$$

327 129

Appendix CTHERMAL NEUTRON CROSS SECTION

The energy of a thermal neutron is customarily taken to be 0.025 ev which corresponds to a temperature of 20°C. Since this reactor is to operate at 1400°F (760°C) a correction must be applied to the table values. For a good many nuclei, the thermal cross section is proportional to $\frac{1}{v}$, and a correction factor can be applied uniformly to these. For other nuclei, corrections must be made by reference to cross section charts.⁽⁹⁾

At 760°C the energy of a neutron is 0.089 ev, so that a $\frac{1}{v}$ cross section can be found by

$$\sigma_{.089} = \sigma_{.025} \sqrt{\frac{.025}{.089}} = .53 \sigma_{.025}.$$

Using this factor for the $\frac{1}{v}$ cross sections, and using the charts for the others, a table is made of the cross sections used in this report.

Isotope	$\sigma_{.025}$ <u>barns</u>	$\sigma_{.089}$ <u>barns</u>	Reference
<u>Li⁶</u>	910	483	AECU-2040 ⁽⁹⁾
Li ⁷	0.033	0.0175	"
Be ⁹	0.009	0.0048	"
C ¹²	0.0045	0.0024	"
F ¹⁹	0.010	0.0053	"
Inconel X	4.11	3.4	"
Xe ¹³⁵	2.8×10^6	2.8×10^6	BNL-170 ⁽⁹⁾

128

<u>Isotope</u>	<u>σ.025 barns</u>	<u>σ.089 barns</u>	<u>Reference</u>
Sm ¹⁴⁹	50,000	70,000	AECU-2040
Sm ¹⁵¹	7,000		"
*Rare Earths "r"	85,000	85,000	Estimated
*Rare Earths "a"	25	25	"
Th ²³²	7.0	3.7	H. Pomerance, Phys. Rev. <u>88</u> , 412 (1952)
Th ²³³	1,400	740	BNL-170
Pa ²³³	125	66	H. Pomerance, private communication
U ²³³ (σ_a)	564	298	BNL-170
U ²³³ (σ_f)	514	272	"

327 131

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129

Appendix D

SAMPLE MULTIGROUP CALCULATIONS

Two tables of data follow. For a discussion of the calculations,
see Chapter 3.

327 132

130
MASTER ~ MULTIGROUP CALCULATION DATA

GROUP NO	ENERGY	LETHARGY	CHANGE IN LETHARGY	INTEGRATED FISSION SPECTRUM	CROSS SECTIONS ← AVERAGED OVER LETHARGY GROUP IN barns →					INFINITE DILUTE	$\frac{N_2}{N_H} = 330$ CONC. 2.73%			
					$\int f w \Delta u$	$D(u)$	$\sigma_a(Li^6)$	$\sigma_a(Be)$	$\sigma_a(Zr)$					
0	10 Mev	0	.											
1	2 Mev	1.62	1.62	0.39	3.1	—	.0343	2.2	.157	.157				
2	.01 Mev	6.91	5.29	0.61	1.02	0.512	.0025	2.633	.006	.006				
3	500 ev	9.90	2.99	0	1.20	3	0	8	0	0				
4	150	11.11	1.21	↑	↑	8.64	↑	23.6	3.3	3.3				
5	100	11.52	0.41			13.0		32	12.2	5.3				
6	30	12.74	1.22			19.5		56.7	9.1	3.84				
7	20	13.15	0.41			29.0		105	136.6	43.5				
8	3	15.06	1.91	↓	↓	53.5	↓	115	1.35	1.35				
9	0.4	17.08	2.02	0	1.20	142.5	0	219	1.35	1.35				
THERMAL	.089	18.52			1.00	480	.0048	298	3.70	3.70				

327

133

FIG. D-1
-127-

327

133

FIG. D-1
-127-

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Thermal
Li⁷
F

$$\frac{\Sigma_a(Th)}{\xi \Sigma_T} \Delta u \bar{q}_n \text{ for thermal} = \frac{\Sigma_a^{th}(Th)}{\Sigma_a^{th} + D^{th} B^2} q_n$$
$$\frac{\Sigma_f}{\xi \Sigma_T} \Delta u \bar{q}_n \text{ for thermal} = \frac{\Sigma_f^{th}}{\Sigma_a^{th} + D^{th} B^2} q_n$$
$$K_{eff} = \nu \sum_n \frac{\Sigma_f}{\xi \Sigma_T} \Delta u \bar{q}_n$$
$$B.R. = \sum_n \left\{ \frac{[\Sigma_a(Th)/\xi \Sigma_T] \Delta u \bar{q}_n}{[\Sigma_f/\xi \Sigma_T] \Delta u \bar{q}_n} \right\} \frac{\Sigma_f^{23}}{\Sigma_a}$$

$$\frac{\Sigma_a}{\xi \Sigma_T} \text{ for Group 7} = \frac{\Sigma_a^{th}}{\xi \Sigma_s} + \frac{\Sigma_a^{total} - \Sigma_a^{th}}{\xi \Sigma_T}$$
$$\frac{\Sigma_a^{23}}{\Sigma_f^{23}} = 1.098 = \frac{1}{0.91075}$$
$$\Sigma_f^{th} \text{ for Groups 1 and 2} = \Sigma_a^{th}$$
$$q_n = \frac{2q_{n-1} + f(u) \Delta u}{\frac{DB^2 + \Sigma_a}{\xi \Sigma_T} + 2}$$
$$q_n = 2 \bar{q}_n - q_{n-1}$$

FAST FISSIONS	.25593
Thermal FISSIONS	.13922
TOTAL	.39515
FAST TH. ABS.	.40979
THERMAL TH. ABS.	.04710
TOTAL	.45689
FAST LEAKAGE	.06993
THERMAL "	.00346
TOTAL	.07339

FIG. D-2
-128-

131

Appendix E

FUSED SALT EXPERIMENTAL DATA

The phase diagrams of $\text{LiF-BeF}_2\text{-UF}_4$ (Fig. E-1) and of $\text{LiF-BeF}_2\text{-ThF}_4$ (Fig. E-2) are copies from reference (1). The viscosity and density data (Figs. E-3 and E-4) are from J. K. Davidson, KAPL (private communication).

132

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FIG. E-1

-130-

327 136

136

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FIG. E-2

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137

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138

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DENSITY DATA FOR
LiF-BE₂ FUSED SALT

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327 143