ANL-6286

ANL-6286

MASTER

Argonne National Laboratory

CONCEPTUAL DESIGN OF A
COUPLED BREEDING SUPERHEATING REACTOR,
CBSR

by

R. Avery, W. V. Dewey,
R. Rohde and B. J. Toppel

DISCLAIMER

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

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

- A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or*
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.*

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

*Price \$1.75 . Available from the Office of Technical Services,
Department of Commerce, Washington 25, D.C.*

ANL-6286
Reactor Technology
(TID-4500, 16th Ed.)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

CONCEPTUAL DESIGN OF A
COUPLED BREEDING SUPERHEATING REACTOR, CBSR

by

R. Avery, W. V. Dewey, R. Rohde,
and B. J. Toppel

Reactor Engineering Division

March 1961

Operated by The University of Chicago
under
Contract W-31-109-eng-38

TABLE OF CONTENTS

	<u>Page</u>
I. INTRODUCTION.	7
II. SUMMARY	10
A. Reactor System	10
B. Conclusions.	15
C. Research and Development.	16
III. DESCRIPTION.	17
A. General	17
B. Reactor Vessel	18
C. Hexagonal Cans	19
D. Separator	21
E. Pressure Tubes	21
F. Fuel.	21
1. Fast Core (Zone 1)	21
2. Radial Buffer (Zone 2).	23
3. Upper and Lower Fast Axial Blankets (Zones 3A and 3B)	23
4. Thermal Core (Zone 5)	23
5. Radial Thermal Blanket (Zone 6).	23
6. High-density Radial Thermal Blanket (Zone 7).	24
7. Upper and Lower Axial Thermal Blanket (Zones 8A and 8B)	24
G. Control Rods	24
H. Fuel Handling	26
IV. PHYSICS ANALYSIS	27
A. Introduction.	27
B. Reference System.	27
C. Calculational Procedures and Cross Sections	29
D. Results.	34
E. Conclusions.	40

TABLE OF CONTENTS

	<u>Page</u>
V. ENGINEERING ANALYSIS	41
A. Introduction.	41
B. Heat Transfer	41
C. Steam Cycle	44
D. Startup and Shutdown.	46
1. Startup	46
2. Shutdown.	47
VI. RESEARCH AND DEVELOPMENT.	48
Appendix: PHYSICS	49
REFERENCES	67

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1	Summary Data	11-15
2	Engineering-Physics Regional Correspondence	28
3	Regional Compositions for Physics Calculations	28
4	Thermal Region Fuel Data for Physics Calculation.	29
5	13-group Macroscopic Cross Sections	30
6	Four-group Macroscopic Cross Sections	32
7	Summary of Physics Reference Problem	33
8	Neutron Inventory for Physics Reference System	35
9	Fuel Design Criteria.	41
A-1	Fuel Cluster Used in the Thermal-Fast Steam Superheater.	52
A-2	Composition of Various Regions of Thermal-Fast D ₂ O Coupled Steam Superheater	52
A-3	Composition of Various Regions of Fast-Thermal D ₂ O Coupled Steam Superheater	53
A-4	Composition of Various Regions of Fast-Thermal H ₂ O Coupled Steam Superheater	53
A-5	Composition of Regions of All-Fast Reference System	53
A-6	Four-group Microscopic Cross Sections.	54
A-7	Composition of Fuel and Moderator Regions for Cell Calculations in Clustered Fuel Zones.	55
A-8	Reactivity Characteristics of Reference Systems	64
A-9	Neutron Inventory for Thermal-Fast D ₂ O Coupled Steam Superheater	65
A-10	Neutron Inventory for Fast-Thermal D ₂ O Coupled Steam Superheater	65
A-11	Neutron Inventory for Fast-Thermal H ₂ O Reference System	66
A-12	Neutron Inventory for All-Fast Reference System	66
A-13	Comparison of Superheaters and the Fast Reference System	67

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1	Reactor Elevation.	17
2	Flow Diagram	18
3	Hexagonal Can and Fuel.	19
4	Core Plan View	20
5	Pressure Tube and Fuel	22
6	Control Rods	25
7	Schematic Drawing of Physics System	27
8	Four-group, 2D, and Contracted Thirteen-group, 1D Radial Flux Distributions at Midplane	36
9	Comparison of Four-group and Thirteen-group Power Dis- tributions at Midplane	37
10	Axial Flux Distribution at Central Axis	38
11	Axial Power Distribution at Central Axis	38
12	Axial Flux Distribution at 45.0-cm Radial Position.	39
13	Axial Power Distribution at 45.0-cm Radial Position	39
14	Heat Balance	45
A-1	Characteristics of a Boiling-Superheating System Assuming 100 Mw in the Superheater and 593°C Superheated Steam. . . .	50
A-2	Schematic Drawing of Thermal-Fast Coupled Steam Superheater.	51
A-3	Schematic Drawing of Fast-Thermal Coupled Steam Superheater.	51
A-4	Schematic Drawing of All-Fast Reference System	52
A-5	Fuel Cluster Used in Thermal-Fast Steam Superheater.	52
A-6	Radial Flux Distribution for Thermal-Fast D ₂ O-Coupled Steam Superheater at Midplane	60
A-7	Radial Flux Distribution for Fast-Thermal D ₂ O-Coupled Steam Superheater at Midplane	60
A-8	Radial Flux Distribution for Fast-Thermal H ₂ O-Coupled Steam Superheater at Midplane	61
A-9	Radial Flux Distribution for All-Fast Sodium-Cooled Reference System at Midplane	61

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
A-10	Radial Power Distribution for Thermal-Fast D ₂ O-Coupled Steam Superheater at Midplane	62
A-11	Radial Power Distribution for Fast-Thermal D ₂ O-Coupled Steam Superheater at Midplane	62
A-12	Radial Power Distribution for Fast-Thermal H ₂ O-Coupled Steam Superheater at Midplane	63
A-13	Radial Power Distribution for All-Fast Sodium-Cooled Reference System at Midplane	63

CONCEPTUAL DESIGN OF A COUPLED BREEDING SUPERHEATING REACTOR, CBSR

by

R. Avery, W. V. Dewey,
R. Rohde and B. J. Toppel

I. INTRODUCTION

The concept of a coupled fast-thermal reactor envisages coupling neutronically a fast assembly and a thermal assembly to each other. Various such systems have been studied in the past. The primary purpose of these studies has been to determine the extent to which the coupled system may possess favorable characteristics for various reactor applications. The major application given consideration has been with respect to a power breeder system. The primary potential advantages offered by a fast-thermal power breeder are the possibilities of attaining the high breeding gain of a fast assembly and the long neutron lifetime of a thermal assembly.

In a system where there is a small fraction of the neutrons that have very much longer prompt lifetimes than the others, the overall prompt neutron lifetime will be largely determined by these slowest neutrons as long as they are still needed to maintain the chain reaction. This is analogous to the delayed-neutron effect where the neutron generation time is primarily determined by the delayed neutrons as long as the system is below prompt critical. For this reason, the neutron lifetime in the coupled fast-thermal system is in the range characteristic of thermal reactors.

To obtain a breeding gain characteristic of fast assemblies, the power must be generated primarily in the fast part. In addition, it is necessary to shield the fast region as much as possible from the low-energy neutron spectrum in the thermal region. The reason for this is two-fold: (a) to maintain the energy of neutrons causing fissions in Pu^{239} high so as to attain a low value for the capture-to-fission ratio (α); and (b) to prevent a hot spot at the edge of the fast assembly.

The essential problem in the design of the system is to obtain sufficient coupling between the systems while at the same time introducing a barrier between the systems which sufficiently isolates the fast system from low-energy neutrons.

The coupled fast-thermal systems that have been studied have had cylindrical symmetry, with one of the assemblies forming an annulus surrounding the other. Because of the high critical mass usually associated with a fast assembly, it has usually been on the inside. The system reported on here retains the usual geometry. The fast core acts as a superheater and is steam cooled; the thermal part is cooled by water of approximately normal density.

The basic potential advantages of the system under study are: (a) a direct cycle using superheated steam; (b) a substantial breeding gain; (c) a long neutron lifetime; and (d) a light water system. This combination of characteristics in one system is unique. The most obvious potential disadvantage of the concept is the possibly low power density attainable in the fast core. This disadvantage is given importance by the large critical mass associated with the fast core.

Since the concept is being considered in terms of central station power, the cost of producing electric power is the criterion upon which the potential of the system should be determined. However, cost analyses have many uncertainties in them, particularly so for systems for which only conceptual design studies have been made, and for which many details relating to feasibility, design complexity, research and development programs are not fully determined. A further complicating feature arises whenever one of the main advantages of the system relates to neutron economy or breeding, since economic studies based on present supplies of fossil and nuclear fuel may no longer be applicable in the future.

For these reasons it is perhaps more reliable not to evaluate the system directly in terms of cost of producing electric power, but rather to make relative comparisons with other systems. For example, the present system may be compared with any of the current systems under study for producing superheated steam in thermal reactors. The basic difference is then that positive breeding gains are not possible in these other systems whereas the present system allows breeding ratios substantially over 1 and, in fact, as will be shown later, of the order of 1.4. Since the breeding ratio is so different in the thermal superheating systems from the present one, it is likely to be extremely difficult to reach any conclusions from the comparisons.

Other systems, either all fast or based on the coupled fast-thermal concept but primarily fast, may be more susceptible to direct comparison with the present system. For example, the present system may be compared, on the one hand, with the all-fast sodium system or the coupled fast-thermal sodium system, or, on the other hand, with the all-fast steam-cooled system. The relative advantages and disadvantages of the

all-fast system versus the coupled fast-thermal system are quite uncertain at this time, and it is not the purpose of the present study to shed much light on this question.

The system selected for this study has been the coupled fast-thermal steam superheater with a light water thermal region rather than the all-fast steam-cooled system in the belief that the possible hazard associated with full-density water entering the fast core is such as to put a premium on any variations of the system which might tend to increase safety.

In comparing the present system with the fast-thermal sodium-cooled system, there is little reason to believe a priori that the breeding gains of the two systems will differ appreciably, and calculations show that the breeding gains are quite similar. Since the limiting temperature conditions of heat transfer surface and fuel for both systems are about the same, and liquid metals have the advantage with respect to heat transfer, it is not expected that the steam-cooled system can compete on a specific power basis. In fact, the results of the study show that power densities of the order of one-third to one-half of that obtained in the sodium-cooled system are the most that can be expected in the steam-cooled case. In view of this, the main justification for studying the present concept results from the as yet unproven performance of the sodium-cooled systems and from the possibility that the considerably different steam cycle in the present system might, because of thermal efficiency, design simplicity, or in other ways, prove to be superior.

In order to evaluate the design features and performance of the system, a conceptual design study has been undertaken and is reported on here. No great effort was made to optimize the system in all detail. Rather, the motivation was to obtain a design which more or less characterizes the basic features of the concept. It is obvious that from a competitive viewpoint the coupled steam reactor must strive for a high power density. For this reason, it was decided to take a reasonably optimistic attitude with regards to the attainable power densities in the fast core.

Since the coupled steam-superheater concept is a departure from currently existing systems, it was recognized that research and development will be necessary, and consequently problem areas requiring work are outlined in the report.

II. SUMMARY

A. Reactor System

The conceptual design of the Coupled Breeding Superheating Reactor, CBSR, for achieving a positive breeding gain and for producing 65 Mw of electric power is presented. The design combines a steam-cooled fast region and a nonboiling pressurized light water-cooled thermal region. The advantage offered by this arrangement as compared with that using a solid moderator in the thermal zone is that, if a power excursion occurs, the water will increase the void content and tend to limit the excursion.

The total reactor power is 216 Mwt, of which 163 Mwt is used to superheat steam as it passes through the fast regions of the reactor and 53 Mwt is transferred to the pressurized water. For this power split the fast core is 4% subcritical without the reactivity contribution of the thermal region. A breeding ratio of 1.4 is calculated for an oxide-fueled fast and thermal core and a high-density, metal-fueled radial blanket.

The steam throttle conditions produced are 75 atm and 453°C for an average fast core power density of 500 Mw/l.

The original goal of 565°C throttle steam temperature and 1 Mw/l power density had to be compromised because of the surface temperature limitation of currently available cladding materials.

The system does not require a large external power source for producing the steam introduced into the fast core. This is possible through the use of a steam compressor that increases the pressure of a portion of the superheated steam and thus permits its use to generate the required saturated steam flow rate by vaporizing the feedwater from the steam cycle.

The design includes a pressure-balance system that equalizes the static pressure in both the pressurized water and steam systems. This pressure-balance system also provides a means of cooling the steam regions in an emergency by allowing the pressurized water to flash. These features are intended to permit easier startup, operation, and shutdown of the entire system.

A summary of the reactor design characteristics is given in Table 1.

Table 1
SUMMARY DATA

General

Total Power	216 Mwt
Power output	65 Mwe
Efficiency	30%
Breeding Ratio	1.4
Critical mass of fast core	242 kg Pu
Fast core cycle	100 days at 1% burnup
Steam-cooled zones	
Power	163 Mwt
Steam flow rate	295 kg/sec
Average inlet temperature	308°C
Average outlet temperature	453°C
Steam pressure	95 atm
Core pressure loss	20 atm
Pressurized water zones	
Power	53 Mwt
Water flow rate	1260 kg/sec
Average inlet temperature	294°C
Average outlet temperature	302°C
Core outlet pressure	95 atm
Core pressure loss	8.5 atm

Steam-cooled Zones (See Fig. 1)

	<u>Fast Core Zone 1</u>	<u>Radial Buffer Zone 2</u>	<u>Upper Axial Fast Blanket Zone 3A</u>	<u>Lower Axial Fast Blanket Zone 3B</u>
Power, Mwt	155	4.8	1.4	1.4
Power density, kw/l	506	58	4.8	4.8
Core characteristics, cm				
OD (equivalent)	69.9	78.7	78.7	78.7
ID (equivalent)	-	69.9	-	-

Table 1 (Cont'd.)

Steam-cooled Zones (Cont'd.)

	Fast Core Zone 1	Radial Buffer Zone 2	Upper Axial Fast Blanket Zone 3A	Lower Axial Fast Blanket Zone 3B
Core characteristics				
Length, cm	80	80	60	60
Volume, ℓ	306	83	292	292
Volume fraction				
Steam	0.38	0.41	0.41	0.41
Structure	0.16	0.14	0.14	0.14
Fuel	0.46	0.45	0.45	0.45
Power, max/avg	1.78	1.66	6.6	7.2
Coolant
Maximum inlet velocity, cm/sec	4572	594	4240	10580
Flow area, cm^2	1411	413	1940	1940
Steam flow rate, kg/sec	282	13	295	295
Pressure, atm				
Inlet	-	-	95	-
Outlet	-	-	-	75
Temperatures, $^{\circ}\text{C}$				
Inlet, average	310	310	294	450
Outlet				
Average	456	386	309	453
Maximum	604	432	323	456
Surface, maximum	718	718	362	649
Internal, maximum	1450	1150	421	788
Heat Transfer				
Surface, cm^2	2.1×10^6	1.9×10^5	6.7×10^5	6.7×10^5
Heat flux, watts/cm^2				
Average	74	26	2.1	2.1
Maximum	178	49	20	20
Hot channel factors				
Fluid temp rise, $F_{\Delta t}$	2.0	1.6	1.9	1.9
Film temp difference, F_{θ}	2.4	1.9	9.8	9.8
Fuel temp rise, $F_{\Delta T}$	2.0	1.6	7.9	7.9

Table 1 (Cont'd.)

Steam-cooled Zones (Cont'd.)

	Fast Core Zone 1	Radial Buffer Zone 2	Upper Axial Fast Blanket Zone 3A	Lower Axial Fast Blanket Zone 3B
Fuel				
Rod diameter, cm	0.318	0.952	0.952	0.952
Cladding	← Stainless Steel →			
Clad thickness, cm	0.0127	0.043	0.043	0.043
Radial helium space, cm	0.0025	0.005	0.005	0.005
Pellet diameter, cm	0.287	0.856	0.856	0.856
Pellet material	PuO ₂ + Depleted UO ₂	← Depleted UO ₂ →		
Pellet density, g/cc	10.2	10.2	10.2	10.2
Equilateral rod pitch, cm	0.394	1.181	1.181	1.181
No. of rods/hexag- onal can	547	61	61	61
No. of hexagonal cans	48	13	61	61
Total number of rods	26,256	793	3,721	3,721

Pressurized Water Zones (See Fig. 1)

	Thermal Core Zone 5	Radial Blanket Zone 6	Thermal Blanket Zone 7	Upper Axial Thermal Blanket Zone 8A	Lower Axial Thermal Blanket Zone 8B
Power, Mwt	30	4	13.8	2.6	2.6
Power density, kw/ℓ	106	25	3.3	7.8	7.8
Core characteristics					
OD (equivalent) cm	104	115	198	115	115
ID (equivalent) cm	80	104	115	80	80
Length, cm	80	80	208	60	60
Volume, ℓ	282	161	4200	333	333
Volume fraction					
Water	← 0.33 →				
Structure	← 0.40 →				
Fuel	← 0.27 →				
Power, max/avg	1.3	1.6	17	6.9	6.6

Table 1 (Cont'd.)

Pressurized Water Zones (Cont'd.)

	Thermal Core Zone 5	Radial Blanket Zone 6	Thermal Blanket Zone 7	Upper Axial Thermal Blanket Zone 8A	Lower Axial Thermal Blanket Zone 8B
Coolant	←————— Water —————→				
Maximum inlet velocity, cm/sec	←————— 914 —————→				
Flow area, cm ²	1080	615	6300	1690	1690
Flow rate, kg/sec	357	204	707	561	561
Core outlet pressure, atm	←————— 95 —————→				
Core pressure loss, atm	←————— 8.5 —————→				
Temperature, °C					
Inlet, average	←————— 294 —————→				
Outlet					
Average	←————— 302 —————→				
Maximum	←————— 306 —————→				
Surface, maximum	←————— 316 —————→				
Internal, maximum	579	393	393	460	460
Heat Transfer					
Surface, cm ²	5.04x10 ⁵	2.88x10 ⁵	4.29x10 ⁶	5.95x10 ⁵	5.95x10 ⁵
Heat flux, watts/cm ²					
Average	59.6	13.7	3.2	4.4	4.4
Maximum	107	30	75	42	42
Hot channel factor					
Fluid temp rise, F _{Δt}	1.45	1.8	See text	1.7	1.7
Film temp difference, F _θ	1.8	2.2	23.3	9.5	9.5
Fuel temp rise, F _{ΔT}	1.8	2.2	23.3	9.5	9.5
Power ratio					
Burnout	2.0	8.3	5.2	27	27
Fuel melting	1.15	4.0	1.25	2.3	2.3

Table 1 (Cont'd.)

Pressurized Water Zones (Cont'd.)

	Thermal Core <u>Zone 5</u>	Radial Blanket <u>Zone 6</u>	Thermal <u>Zone 7</u>	Upper Axial Thermal Blanket <u>Zone 8A</u>	Lower Axial Thermal Blanket <u>Zone 8B</u>
Fuel					
Rod diameter, cm	0.785	0.785	1.407	0.785	0.785
Cladding	← Zircaloy-2 →				
Clad thickness, cm	0.051	0.051	0.089	0.051	0.051
Radial helium space, cm	0.005	0.005	0.008	0.005	0.005
Pellet diameter, cm	0.673	0.673	1.214	0.673	0.673
Pellet material	Natural UO ₂	Depleted UO ₂	Depleted uranium	Depleted UO ₂	Depleted UO ₂
Pellet density, g/cc	10.2	10.2	18.7	10.2	10.2
Rod pitch	← See Fig. 5 →				
No. of rods/pres- sure tube	61	61	19	61	61
No. of pressure tubes	42	24	246	66	66
Total number of rods	2562	1464	4674	4026	4026

B. Conclusions

The conclusions drawn from the study are:

(1) The coupled fast-thermal steam superheating concept is feasible to construct using present-day technology.

(2) A breeding ratio of about 1.4 can be obtained from a coupled fast-thermal reactor using as coolants light water in the thermal regions and light water steam in the fast regions.

(3) The large breeding ratio can be realized for this light water and steam-cooled system because of the small water density in the fast core and large fast power fraction.

(4) The breeding ratio in the light water system is not appreciably different from that of a corresponding heavy water system because the smaller neutron leakage in the light water system compensates for the greater parasitic loss. The parasitic loss in the external region of the light water system is minimized by the use of a high density of fuel.

(5) Because of the moderate power densities available in the fast core, electric power is probably uneconomical in this system when compared with a successful sodium-cooled fast system.

(6) To achieve a better economic position, higher power densities and higher steam temperatures are necessary. Consequently the development of a high-temperature fuel and cladding for a steam environment is essential.

C. Research and Development

In order to achieve an economical position with this concept, investigations in the following areas are necessary:

(1) determination of the limits for steam corrosion and erosion on high-temperature alloys;

(2) development of a low-cost, high-temperature alloy with good creep strength and corrosion resistance for fuel element cladding;

(3) research to produce a fuel having high conductivity, a high melting point, good fission gas retention, and compatibility with steam;

(4) development of less expensive methods of fuel fabrication;

(5) studies of heat transfer and fluid flow characteristics of steam in tightly packed fuel subassemblies;

(6) physics studies, perhaps including critical experiments, to determine the static and dynamic characteristics of the systems; and

(7) studies, perhaps including a reactor experiment, to verify feasibility of startup and shutdown performance, with particular emphasis on the hazards associated with flooding and voiding of various regions.

III. DESCRIPTION

A. General

The fast and thermal regions of the reactor core are contained in a pressure vessel as shown in Fig. 1. The various zones of the core are schematically illustrated on the left side and, for simplicity, may be regarded as cylindrical in nature and symmetrical about the vertical centerline. The right half shows the component parts and flow paths for the steam and pressurized water regions.

FIG. 1
REACTOR ELEVATION

Saturated steam at 95 atm is introduced into the vessel through the upper steam nozzles and flows downward through the upper fast axial blanket, the fast core and its radial buffer, and the lower fast axial blanket.

Water at 294°C and 104 atm is pumped into the upper water box and passes through the pressure tubes and returned to the lower water box prior to leaving the vessel. The steam is superheated to 453°C as it passes through the steam regions, with a pressure loss of 20 atm, and absorbs 163 Mwt before leaving the bottom steam nozzles. A small portion of the inlet steam bypasses the steam-cooled regions and condenses on the pressure tubes. The condensate is removed and enters the feedwater cycle.

The main steam flow scheme is shown in Fig. 2. Part of the steam from the fast regions goes to a steam compressor, part goes to the compressor drive unit, and the remainder is fed to the turbine-generator unit. The steam going to the compressor is cooled by mixing with feedwater before entering the compressor to lower the inlet temperature. The compressed steam is mixed with the feedwater from the heating stages and provides the saturated steam for the steam zones. The remaining steam is used for the steam compressor drive turbine and the turbine-generator unit. Five steam extractions are made from the turbine for heating the feedwater.

The pressurized water leaving the thermal regions at 302°C and 95 atm flows to a heat exchanger, where the water rejects 53 Mwt to the feedwater and is pumped back to the thermal regions. The water system is pressurized by virtue of a connection between the mixer-pressurizer and the pressurizer in the pressurized water system.

B. Reactor Vessel (See Fig. 1)

The carbon steel-stainless steel-clad pressure vessel has an inside diameter of 228 cm, a thickness of 15 cm, and an overall length of 648 cm. The upper head closure is ellipsoidal in shape and bolts in place.

The two 25-cm diameter inlet steam pipes are located below the head flange, 180 degrees apart.

A grid for positioning the hexagonal cans and the pressure tubes is located near the bottom and is fitted with seats which fit the hexagonal cans and prevent steam bypass. The sealing of the cans is accomplished by the steam pressure differential existing across the length of the cans. The grid also provides the locating holes for the pressure tubes which are sealed to the grid by bolts in the corners of the pressure tube flange.

Below the grid is a flow baffle that separates the inlet and outlet water streams.

Three inlet and three exit pipes, 25 cm in diameter, are provided for the water flow.

In the section below the grid there is an 86-cm diameter extension to which two 30-cm diameter outlet steam pipes are welded 180 degree apart. The extension is closed by a flat head penetrated by the shafts attached to the control rods.

C. Hexagonal Cans (Figs. 1, 3, and 4)

The fast core, the axial fast blankets, and the radial buffer are contained in 61 hexagonal stainless steel cans which are arranged, as shown in Fig. 4, in a hexagonal array. The length and equivalent diameters of these zones are listed in Table 1. The cans measure 9.32 cm across the inside flats of the hexagon. The wall thickness is 0.1 cm and the overall can length with fittings is 254 cm.

FIG. 3
HEXAGONAL CAN AND FUEL

NOTE
ALL DIMENSIONS IN cm

The hexagonal cans were chosen to compartment the fuel for refueling, for flow orificing, and to impede rapid flooding of the fast core in the event of a water leak within the vessel. The size of the cans is based on considerations of fuel handling and the number of connections involved.

In order to adhere to the core dimensions given in Table 1, the cans in the vicinity of the outer diameter of the fast core and in the radial buffer contain internal dividers, as shown in Fig. 4. These dividers separate the fast core fuel from the larger diameter radial buffer rods.

Each of the cans is fitted with a nozzle at the top end for positioning and orificing of the subassemblies. The orifice in each can is sized for an equal rise of steam temperature across the core. The lower end box is fitted with a ground seat for sealing.

D. Separator (Figs. 1 and 4)

A stainless steel separator is used to protect the Zircaloy pressure tubes from steam, to decrease the transfer of heat from the superheated steam to the thermal zones, and to prevent pressurized water from spraying into the fast zones in case of a leak in the pressure tubes. The separator is composed of six 0.64-cm stainless steel sheets, 229 cm long, that fit around the outside row of hexagonal cans.

E. Pressure Tubes (Figs. 1 and 5)

Each of the 312 Zircaloy pressure tubes has an inside diameter of 8.38 cm, an outside diameter of 9.65 cm, and an overall length of 254 cm. The tubes are arranged in a hexagonal array. The interstices formed by the tubes are filled with Zircaloy filler rods.

The pressure tubes are connected in pairs, and therefore a two-pass flow arrangement is necessary. This feature permits the upper part of the vessel to be cleared of large pipes and headers which otherwise would be required for once-through flow. The return bend connecting the two pressure tubes is capable of undergoing length changes without harm because of temperature differentials.

The sealing of the pressure tubes to the grid is accomplished by a special three-point washer and bolt every 60 degrees around the tube in the flange, as depicted in Fig. 5. The three-point washer assures contact in one plane regardless of slight thickness variations of the flanges. The neighboring pressure tubes help to complete the flange. The grid accepts the pressure tubes which have the necessary seal rings securely attached to them. The bolts used to tighten the seal are attached to the filler rods to prevent loss of bolts.

F. Fuel (Figs. 1, 3, 4, and 5)

1. Fast Core (Zone 1)

The fuel rods chosen for the fast core consist of 0.318-cm OD stainless steel tubes with a 0.01-cm wall thickness and a length of 85.1 cm. Pellets of PuO_2 mixed with depleted UO_2 fill the inside of the tubes. A fission gas space of 5.1 cm is left at the top of each tube. The ends of the tube are welded closed. The top end is welded to a cross bar and the other end is free to expand, but held to a cross bar to prevent radial movement by a mechanical connection. The cross bars form a grid at each end after welding. Radial movement of the fuel rods is prevented by a 0.076-cm wire which is affixed to the fuel rods. The wire curves around the rod in a helical pattern and touches the outside of the adjacent fuel rods.

NOTE:
ALL DIMENSIONS IN cm

The finished fuel subassembly contains 547 fuel rods which are inserted into a hexagonal can. Figure 4 shows that there are a total of 61 hexagonal cans with the radial buffer sharing the equivalent of 13 cans as indicated. With this arrangement 43 to 48 hexagonal cans filled with fuel comprise the fast core, depending on the position of the 12 control and shutdown rods which occupy the outer hexagonal row of the steam-cooled zones.

2. Radial Buffer (Zone 2)

As shown in Fig. 4, the radial buffer occupies a space that is the equivalent of 13 of the hexagonal cans. The rods in the radial buffer are constructed of 0.952-cm OD stainless steel tubes, with 0.043-cm thick walls, and 85.1 cm long. Depleted UO_2 pellets are inserted in the tubes. A fission gas space of 1.27 cm is left at the top end. A separator of 0.16-cm stainless steel within the hexagonal cans divides the fast and radial zones. This separator permits orificing at the upper grid to control the steam flow in the radial buffer.

3. Upper and Lower Fast Axial Blankets (Zones 3A and 3B)

The volumes immediately above and below the fast core and its radial buffer are occupied by the axial blankets. The fuel used in these zones is depleted uranium oxide in pellet form held in 0.952-cm OD stainless steel tubes. The tubes have a wall thickness of 0.043 cm and an overall length of 61.2 cm. Each rod has a fission gas space of 1.27-cm length.

Each of the 61 subassemblies which fit into the hexagonal cans contains 61 rods on an equilateral pitch. No longitudinal support wires are attached to these rods, but cross grids of wires are placed periodically to prevent vibration of the tubes under operating conditions. The rods are welded to one grid and mechanically held in the other grid to permit longitudinal expansions.

4. Thermal Core (Zone 5)

The fuel rods in the thermal core contain natural UO_2 pellets which are enclosed in 0.784-cm OD Zircaloy tubes, 0.051 cm thick and 85.1 cm long. The rods are attached at the top and bottom to grids which permit axial movement of the rods. Wires periodically support the rods. The grids with 61 fuel rods are positioned inside the equivalent of 42 pressure tubes thus forming the thermal core.

5. Radial Thermal Blanket (Zone 6)

As shown in Fig. 4, the thermal core does not occupy complete rows of pressure tubes. Since the first hexagonal row of pressure tubes

numbers 30, it was necessary to make up the remaining 12 pressure tubes by using one-third of the 36 pressure tubes in the next row. The remainder of the 36 tubes contains depleted UO_2 .

6. High-density Radial Thermal Blanket (Zone 7)

The high-density thermal blanket is composed of 1.407-cm OD, 0.089-cm thick wall Zircaloy tubing, 207.5 cm long, into which is placed depleted uranium metal. No bonding is required. Nineteen fuel rods form a cluster which fits into a pressure tube. Spacers made of wire are located periodically along the length to prevent vibration of the tubes. The pressure tubes are welded closed at the top after insertion of the rods.

7. Upper and Lower Axial Thermal Blanket (Zones 8A and 8B)

The blanket rods above and below the thermal core are geometrically the same as the thermal core, but depleted uranium oxide is used. The rods are 61.2 cm long and attached to grids at each end. Axial expansion and vibration is accommodated by the design.

The fuel is assembled with the grids and inserted into the pressure tubes and the pressure tubes are welded closed after insertion of all the rods.

G. Control Rods (Figs. 1, 4, and 6)

The location of the three control rods and nine shutdown rods is in the outer row of hexagonal cans, as illustrated in Fig. 4. The control and shutdown rods are contained in slightly smaller hexagonal cans which fit within the large hexagonal cans of the steam-cooled zones. Various parts of the control and shutdown rods have the composition of the region in which they reside when in a position of maximum reactivity. With this arrangement the reactor is controlled by the movement of fuel into or out of the core. The shutdown rods differ physically from the control rods since a section of B_4C (with 100% B^{10}), with an overall length of 87.6 cm, is added at the top. All the rods are available for rapid shutdown of the core.

One of the three control rods is shown on the right of Fig. 6. Sections 1 and 3 have the same composition as the fast axial blanket zones. The operating and shutdown positions of the control rods are also shown in Fig. 6. Part of Section 2 contains some fuel rods which have the same geometry as the fast core, and the remaining area is filled with rods corresponding to the radial buffer zone as shown in Fig. 4.

One of the nine shutdown rods is shown at the left of Fig. 6. The normal operating position of Section 2 is such that it is situated in the fast core and radial buffer zones and approaches the idealized geometry. Both the

lower and upper sections, designated as 1 and 3, are positioned in their normal place in the fast axial blanket to reduce neutron leakage. The B_4C section consists of 61 rods made of 0.952-cm OD stainless steel tubes with a 0.043-cm wall thickness. Pellets of B_4C are encased in the tube and the tube is attached to grids at each end. Expansion space is allowed for gas release and changes in length of the B_4C rods because of thermal expansions.

NOTE:
ALL DIMENSIONS IN cm

FIG. 6
CONTROL RODS

H. Fuel Handling

The method envisioned for fuel handling consists of adding borated water to the reactor vessel after the core and vessel have been cooled. The reactor vessel cover will be removed after the vessel pressure has been decreased to atmospheric and that part of the pressurized water system exterior to the vessel is isolated from the vessel by gate valves to prevent mixing the pressurized water with the borated water. The space above the vessel may also be flooded with borated water to a height sufficient for shielding purposes.

The cover is removed under water and placed to one side. Grappling tools can then remove the hexagonal can by grasping the nozzle. The hexagonal can complete with fuel is lifted out and transferred to a storage pit where the various sections are removed after cutting the can. To reinsert fuel in the reactor, the different sections are placed inside new hexagonal cans and the nozzle is attached by welding. The hexagonal cans are then grasped by the grappling tool and replaced in the core.

A pressure tube is removed by unbolting its flange from the grid. A special impact tool will remove all the bolts and prevent their loss. The fuel is removed by cutting the pressure tube; consequently, in refueling, a new pair of pressure tubes is filled with fuel and the pressure tubes are then welded closed.

After the necessary changes have been made in the fuel, the head is bolted to the vessel and the borated water is removed by opening drains in the vessel and flushing with steam. The flushing operation must take place at a temperature at which condensation does not take place.

IV. PHYSICS ANALYSIS

A. Introduction

The primary consideration of the requirement of a large breeding ratio originally led to consideration of a system having D_2O for the coolant because of the lower parasitic capture as compared with H_2O . The results (presented in the Appendix) indicate, however, that an H_2O system can be devised which has a substantial breeding ratio. Owing to this fact, and to the greater technological problems which would be encountered with the use of D_2O , the concept has turned from the original D_2O system to the present H_2O system.

This section describes the physics calculations for a system modeled closely after the engineering design. The calculations have so far been concentrated on proper choice of cross sections and on calculation of the critical configuration. It is believed that the results of the survey calculations given in the Appendix can be used to deduce to a large extent the results of calculations which have not yet been made for the conceptual design.

B. Reference System

FIG. 7
SCHEMATIC DRAWING OF PHYSICS SYSTEM

Figure 7 shows a schematic drawing of the model used for the physics calculations. Slight differences exist between the physics model and the engineering design data owing to recent slight changes in the engineering concept. The regions are numbered differently in Fig. 7 than in the engineering drawings. Table 2 gives the engineering-physics regional designation correspondence. As indicated in the table and in Fig. 7, the stainless steel separator is not included in the physics calculation. Instead, the radial buffer and fast axial blankets have been extended radially to include the region normally occupied by the separator. In this section of the report, all regional numbers will refer to the physics designation.

The regional compositions are given in Table 3 in terms of the volume fractions of the various constituents. The details of the thermal region fuel in the zirconium pressure tubes are given in Table 4.

Table 2

ENGINEERING-PHYSICS REGIONAL CORRESPONDENCE

Region	Engineering Designation	Physics Designation
Fast Core	1	1
Radial Buffer	2	2
Upper Fast Axial Blanket	3A	7
Lower Fast Axial Blanket	3B	12
Separator	4	- ^a
Thermal Core	5	3
Radial Thermal Blanket	6	4
High-density Radial Thermal Blanket	7	5
Upper Thermal Axial Blanket	8A	6
Lower Thermal Axial Blanket	8B	6
Grid Region	-	8
Gas Space	-	9
Grid Region	-	10
Gas Space	-	11

^aThe separator is not included in the Physics Calculation. Region 2 has been extended radially to include the region normally occupied by the separator.

Table 3

REGIONAL COMPOSITIONS FOR PHYSICS CALCULATIONS

Physics Region	Volume Fraction						
	PuO ₂	U ²³⁸ O ₂	U ²³⁵ O ₂	Depleted U ^a	Fe	Zr	H ₂ O ^b
1	0.08556	0.37369	0.00075	-	0.156	-	0.014131
2	-	0.42016	0.00084	-	0.157	-	0.01553
3	-	0.29456	0.00214	-	-	0.4151 ^c	0.2087
4	-	0.29611	0.00059	-	-	0.4151 ^c	0.2087
5	-	-	-	0.3715	-	0.4372 ^d	0.1350
6	-	-	-	0.2967	-	0.4151 ^c	0.2087
7	-	0.42016	0.00084	-	0.157	-	0.02245
8	-	-	-	-	0.616	-	0.014131
9	-	-	-	-	0.156	-	0.014131
10	-	-	-	-	-	0.7118	0.2087
11	-	-	-	-	-	0.4151	0.2087
12	-	0.42016	0.00084	-	0.157	-	0.01084

^a99.8% U²³⁸

^bEquivalent water of normal density

^cVolume fraction of Zr in clad is 0.0878

^dVolume fraction of Zr in clad is 0.1098

Table 4

THERMAL REGION FUEL DATA FOR PHYSICS CALCULATION

	Physics Region			
	3	4	5	6
Fuel	Natural UO_2	Depleted UO_2	Depleted U Metal	Depleted UO_2
Rod Outer Diameter, cm	0.544	0.544	2.710	0.544
Clad	Zirconium	Zirconium	Zirconium	Zirconium
Clad Thickness, cm	0.033	0.033	0.165	0.033
Rod Pitch (Hexagonal)	0.645	0.645	2.774	0.645
Number of Rods per Pressure Tube	139	139	7	139
Water Volume Fraction Inside Pressure Tubes ^a	0.2079	0.2079	0.1342	0.2079

^aEquivalent normal density water.

C. Calculational Procedures and Cross Sections

The physics system was examined in cylindrical geometry using 13-group, one-dimensional diffusion theory on UNIVAC⁽³⁾ (13-group, 1D) and 4-group, two-dimensional diffusion theory⁽⁴⁾ (4-group, 2D) on the IBM-704. Previous calculations⁽²⁾ have indicated that, at the axial mid-plane, 4-group calculations of flux and power distributions in one and two dimensions were in excellent agreement for liquid metal-cooled coupled systems. If we assume such agreement to carry over to the present concept, the 13-group, 1D and 4-group, 2D results may be compared to indicate the extent to which the few-group calculation can reproduce the many-group results. Agreement between the two methods of calculation is important, since not all of the problems to be examined can be solved using only one or the other approach.

Table 5 gives the 13-group macroscopic cross sections. The Pu, Fe, H_2O and O cross sections were obtained using the spectrum appropriate to the steam-cooled oxide fast core. The U^{238} , U^{235} , and zirconium cross sections were obtained using essentially a $1/E$ weighting.

Table 5

13 GROUP MACROSCOPIC CROSS SECTIONS (cm⁻¹)

J	Lower Energy E _L	Fission Spectrum	Pu ²³⁹ O ₂								U ²³⁸ O ₂							
			Σ _f	ν	Σ _c	3Σ _{tr}	Σ _{J→J+1}	Σ _{J→J+2}	Σ _{J→J+3}	Σ _{J→J+4}	Σ _f	ν	Σ _c	3Σ _{tr}	Σ _{J→J+1}	Σ _{J→J+2}	Σ _{J→J+3}	Σ _{J→J+4}
1	1 353 Mev	0 575	0 0466	3 2	0 0028	0 5365	0 0206	0 0045	0 0012	0 0004	0 0121	2 8	0 0023	0 5113	0 0395	0 0102	0 0028	0 0009
2	0 4979 Mev	0 301	0 0405	2 99	0 0023	0 8896	0 0348	0 0038	0 0009	0 0003	0	0	0 0027	0 8294	0 0408	0 0056	0 0014	0 0004
3	0 1832 Mev	0 091	0 0384	2 93	0 0047	1 2737	0 0383	0 0026	0 0008	0	0	0	0 0036	1 1301	0 0376	0 0025	0 0008	0
4	0 0674 Mev	0 025	0 0401	2 91	0 0105	1 2308	0 0216	0 0011	0	0	0	0	0 0057	1 0744	0 0214	0 0010	0	0
5	0 0248 Mev	0 006	0 0480	2 9	0 0179	1 4125	0 0164	0	0	0	0	0	0 0080	1 2796	0 0167	0	0	0
6	0 00912 Mev	0 002	0 0559	2 9	0 0261	1 5523	0 0168	0	0	0	0	0	0 0114	1 3480	0 0171	0	0	0
7	0 00335 Mev	0	0 0596	2 9	0 0326	1 7114	0 0174	0	0	0	0	0	0 0125	1 3874	0 0176	0	0	0
8	1234 ev	0	0 0785	2 9	0 0480	1 7603	0 0193	0	0	0	0	0	0 0137	1 4558	0 0195	0	0	0
9	454 ev	0	0 1153	2 9	0 0774	1 2431	0 0141	0	0	0	0	0	0 0148	1 4558	0 0161	0	0	0
10	200 ev	0	0 1890	2 9	0 1417	2 3824	0 0155	0	0	0	0	0	0 0160	1 5242	0 0170	0	0	0
11	6 ev	0	0 4380	2 9	0 3504	3 9901	0 0014	0	0	0	0	0	0 0342	1 2848	0 0017	0	0	0
12	0 4 ev	0	0 5275	2 9	0 3430	3 7874	0 0009	0	0	0	0	0	0 0087	1 1138	0 0014	0	0	0
13	thermal (273 89°C)	0	15 6652	2 9	7 9428	72 054	0	0	0	0	0	0	0 0401	1 2310	0	0	0	0
			n = 0 0233 x 10 ²⁴ molecules/cm ³								n = 0 0228 x 10 ²⁴ molecules/cm ³							

J	Fe					H ₂ O ^a							Zr						
	Σ _c	3Σ _{tr}	Σ _{J→J+1}	Σ _{J→J+2}	Σ _{J→J+3}	Σ _c	3Σ _{tr}	Σ _{J→J+1}	Σ _{J→J+2}	Σ _{J→J+3}	Σ _{J→J+4}	Σ _{J→J+5}	Σ _{J→J+6}	Σ _c	3Σ _{tr}	Σ _{J→J+1}	Σ _{J→J+2}	Σ _{J→J+3}	Σ _{J→J+4}
1	0 0002	0 5436	0 0594	0 0023	0 0008	0 0010	0 3085	0 0665	0 0230	0 0084	0 0031	0 0011	0 0008	0 0008	0 3825	0 0241	0 0072	0 0020	0 0006
2	0 0003	0 5269	0 0158	0 0027	0 0016	0	0 6887	0 1586	0 0528	0 0194	0 0071	0 0026	0 0017	0 0008	0 6375	0 0137	0 0026	0 0006	0 0002
3	0 0004	0 8011	0 0082	0	0	0	1 0271	0 2343	0 0793	0 0292	0 0108	0 0039	0 0027	0 0008	1 0200	0 0106	0 0008	0 0002	0
4	0 0007	0 8755	0 0096	0	0	0	1 1389	0 3375	0 1198	0 0441	0 0162	0 0059	0 0035	0 0008	1 0840	0 0080	0	0	0
5	0 0009	1 1194	0 0131	0	0	0	1 4150	0 4279	0 1536	0 0563	0 0208	0 0067	0 0053	0 0008	1 0200	0 0075	0	0	0
6	0 0011	0 8802	0 0103	0	0	0	1 5686	0 4886	0 1752	0 0646	0 0210	0 0160	0 0005	0 0008	0 8925	0 0065	0	0	0
7	0 0005	1 8869	0 0165	0	0	0 0001	1 6742	0 5110	0 1843	0 0600	0 0458	0 0014	0 0001	0 0008	0 8925	0 0065	0	0	0
8	0 0008	1 6366	0 0160	0	0	0 0001	1 7009	0 5405	0 1719	0 1313	0 0038	0 0004	0	0 0008	0 8288	0 0061	0	0	0
9	0 0014	2 3171	0 0182	0	0	0 0001	1 7009	0 4524	0 3379	0 0098	0 0008	0	0	0 0008	0 7650	0 0056	0	0	0
10	0 0022	2 6536	0 0209	0	0	0 0002	1 7009	0 8396	0 0239	0 0018	0	0	0	0 0008	0 8925	0 0080	0	0	0
11	0 0049	2 6618	0 0018	0	0	0 0005	1 7343	0 1727	0 0122	0	0	0	0	0 0008	0 7778	0 0017	0	0	0
12	0 0243	2 7200	0 0013	0	0	0 0023	1 7677	0 2624	0	0	0	0	0	0 0010	0 7778	0 0013	0	0	0
13	0 1556	3 1138	0	0	0	0 0140	4 8148	0	0	0	0	0	0	0 0050	0 7828	0	0	0	0
n = 0 0848 × 10 ²⁴ atoms/cm ³ n = 0 0334 × 10 ²⁴ molecules/cm ³ n = 0 0425 × 10 ²⁴ atoms/cm ³																			

J	U ²³⁵ O ₂								Depleted U								10 ⁷ ν̄
	Σ _f	ν	Σ _c	3Σ _{tr}	Σ _{j→j+1}	Σ _{j→j+2}	Σ _{j→j+3}	Σ _{j→j+4}	Σ _f	ν	Σ _c	3Σ _{tr}	Σ _{j→j+1}	Σ _{j→j+2}	Σ _{j→j+3}	Σ _{j→j+4}	
1	0 0299	2 8	0 0038	0 5296	0 0317	0 0078	0 0022	0 0007	0 0255	2 8	0 0019	0 6625	0 0715	0 0214	0 0060	0 0019	0 0045
2	0 0276	2 54	0 0033	0 8712	0 0411	0 0056	0 0014	0 0004	0 0001	2 54	0 0058	0 7201	0 0437	0 0117	0 0028	0 0008	0 0081
3	0 0322	2 46	0 0124	1 1745	0 0377	0 0025	0 0008	0	0 0001	2 46	0 0077	0 9361	0 0242	0 0053	0 0016	0	0 0133
4	0 0391	2 44	0 0094	1 1835	0 0214	0 0010	0	0	0 0002	2 44	0 0120	1 2961	0 0108	0 0022	0	0	0 0219
5	0 0506	2 43	0 0152	1 2908	0 0164	0	0	0	0 0002	2 43	0 0168	1 7280	0 0047	0	0	0	0 0361
6	0 0690	2 42	0 0242	1 4288	0 0168	0	0	0	0 0003	2 42	0 0240	1 8723	0 0050	0	0	0	0 0596
7	0 1035	2 42	0 0414	1 6066	0 0172	0	0	0	0 0004	2 42	0 0265	1 8729	0 0050	0	0	0	0 0982
8	0 1495	2 42	0 0673	1 8136	0 0190	0	0	0	0 0006	2 42	0 0290	2 0174	0 0054	0	0	0	0 1620
9	0 1955	2 42	0 0880	2 4346	0 0167	0	0	0	0 0008	2 42	0 0315	2 0200	0 0054	0	0	0	0 2670
10	0 3220	2 42	0 1610	3 1246	0 0178	0	0	0	0 0013	2 42	0 0342	2 1667	0 0070	0	0	0	0 4200
11	0 9200	2 42	0 4600	5 3326	0 0018	0	0	0	0 0038	2 42	0 0738	1 6728	0 0012	0	0	0	1 3900
12	0 9200	2 42	0 4600	5 3326	0 0015	0	0	0	0 0058	2 42	0 0211	1 3222	0 0013	0	0	0	4 8320
13	8 1073	2 42	1 5017	30 0703	0	0	0	0	0 0338	2 42	0 0906	1 5533	0	0	0	0	29 4920
n = 0 0230 x 10 ²⁴ molecules/cm ³ n = 0 048 x 10 ²⁴ atoms/cm ³																	

^a The calculated age to 0 4 ev is 34 1 cm²

The same H₂O cross sections were used for the steam-cooled regions and for the water zones, since the "age" to 0.4 ev calculated using these cross sections (34.1 cm²) is in close enough agreement with experiment for the present conceptual study. The UO₂ density was taken as 10.2 gm/cc and that of PuO₂ as 10.5 gm/cc.

The 4-group set given in Table 6 was obtained by collapsing the 13-group set using the spectra given by the 13-group, 1D calculation. The spectrum in the fast core (region 1, Fig. 7) was used to collapse the PuO₂ and H₂O cross sections. The 4-group depleted U values were obtained using the high-density radial thermal blanket (region 5) spectrum. In the 4-group case, two water sets were required, since the "age" to 0.4 ev calculated for the H₂O cross sections to be used in the steam zones (110.4 cm²) is unrealistically large for H₂O to be used in the thermal zones. The thermal core (region 3) spectrum was used to obtain the H₂O cross sections for use in the thermal zones. These cross sections yielded a calculated "age" to 0.4 ev of 30.7 cm².

The effective resonance integrals for U²³⁸ in the water regions were obtained using Eq. A-5 of the Appendix in the case of the high-density radial thermal blanket (region 5) and using the equation

$$I_{\text{eff}}^{28} = 11.6 + 22.8 (S_{\text{eff}}/M) \quad (1)$$

for the water-cooled oxide regions 3, 4, and 6. A Dancoff-Ginsburg calculation⁽¹⁴⁾ was used in determining the values of S_{eff} . In the steam zones, the homogeneous formula

$$I_{\text{eff}}^{28} = 3.8 \left(\frac{10^{24} \Sigma_s}{N_U} \right)^{0.42} \quad (2)$$

was used, where Σ_s is the homogenized resonance group scattering cross section and N_U is the number of U²³⁸ nuclei per cc.

In the 13-group set, group 11 was taken to be the resonance group, so that

$$\Sigma_{11}^{\text{removal}} = p_{11} \Sigma_{a11} \quad , \quad (3)$$

in analogy with the four-group equivalent $p \Sigma_{a_3} = \Sigma_{3 \rightarrow 4}$ given in the Appendix. The resonance group capture cross sections (σ_{c11} or σ_{c3} in the 13 and 4-group cases) were adjusted for each region so that, when combined with the other group capture cross sections as

$$\sum_{j=1}^{k-1} \Delta u_j \sigma_{c_j} \quad ,$$

the total effective resonance integral was obtained.

Table 6

FOUR-GROUP MACROSCOPIC CROSS SECTIONS (cm⁻¹)

J	Lower Energy, E _L	Fission Spectrum	Pu ²³⁹ O ₂					U ²³⁸ O ₂				
			Σ _f	ν	Σ _c	3Σ _{tr}	Σ _{J→J+1}	Σ _f	ν	Σ _c	3Σ _{tr}	Σ _{J→J+1}
1	1.353 Mev	0.575	0.0466	3.2	0.0028	0.5365	0.0267	0.0121	2.8	0.0023	0.5113	0.0535
2	9.12 kev	0.425	0.0425	2.93	0.0095	1.2088	0.0016	0	0	0.0056	1.0938	0.0023
3	0.4 ev	0	0.1154	2.9	0.0772	1.8645	0.00002	0	0	0.0182	1.3194	0.0004
4	thermal (273.89°C)	0	15.6652	2.9	7.9428	72.054	0	0	0	0.0401	1.2310	0
n = 0.0233 x 10 ²⁴ molecules/cm ³								n = 0.0228 x 10 ²⁴ molecules/cm ³				

J	U ²³⁵ O ₂					Fe			H ₂ O ^a			H ₂ O ^b		
	Σ _f	ν	Σ _c	3Σ _{tr}	Σ _{J→J+1}	Σ _c	3Σ _{tr}	Σ _{J→J+1}	Σ _c	3Σ _{tr}	Σ _{J→J+1}	Σ _c	3Σ _{tr}	Σ _{J→J+1}
1	0.0299	2.8	0.0038	0.5296	0.0424	0.0002	0.5436	0.0625	0.0010	0.3085	0.1021	0.0010	0.3085	0.1021
2	0.0404	2.45	0.0114	1.1503	0.0023	0.0006	0.8019	0.0010	0	1.0740	0.1317	0	1.0902	0.1572
3	0.5710	2.42	0.2820	3.8393	0.0004	0.0016	2.0202	0.00003	0.0002	1.6932	0.0067	0.0009	1.7276	0.0839
4	8.1073	2.42	1.5017	30.0703	0	0.1556	3.1138	0	0.0140	4.8148	0	0.0140	4.8148	0
n = 0.0230 x 10 ²⁴ molecules/cm ³						n = 0.0848 x 10 ²⁴ atoms/cm ³			n = 0.0334 x 10 ²⁴ molecules/cm ³			n = 0.0334 x 10 ²⁴ molecules/cm ³		

J	Zr			Depleted U				
	Σ _c	3Σ _{tr}	Σ _{J→J+1}	Σ _f	ν	Σ _c	3Σ _{tr}	Σ _{J→J+1}
1	0.0008	0.3825	0.0340	0.0255	2.8	0.0019	0.6625	0.1008
2	0.0008	0.9152	0.0009	0.0002	2.46	0.0116	1.1954	0.0007
3	0.0009	0.8088	0.0003	0.0032	2.42	0.0396	1.7049	0.0004
4	0.0050	0.7828	0	0.0338	2.42	0.0906	1.5533	0
n = 0.0425 x 10 ²⁴ atoms/cm ³				n = 0.048 x 10 ²⁴ atoms/cm ³				

^aFor use in steam regions. The calculated age to 0.4 ev is 110.4 cm².

^bFor use in water regions. The calculated age to 0.4 ev is 30.7 cm².

(Here k is the number of groups and Δ_{u_j} the group lethargy widths.) For materials other than U^{238} , infinite dilution resonance integrals were assumed.⁽⁵⁾ The resonance escape probability was calculated using Eq. A-3 of the Appendix and the appropriate part of the total effective resonance integral corresponding to the resonance group ($\Delta_{u_{11}} \sigma_{c_{11}}$ or $\Delta_{u_3} \sigma_{c_3}$ in the 13 and 4-group cases). The total p was taken as in the Appendix to be the product of the escape probabilities for the various materials in the region.

Diffusion theory cell calculations in the water regions indicated a thermal disadvantage factor within 2% of unity. In addition, it was found that the disadvantage factor was insensitive to the value of the cell "fuel zone" absorption cross section. Variations of the absorption cross section by a factor of two resulted in less than 0.1% change in the thermal disadvantage factor. The thermal cross sections were therefore obtained by simple homogenizing assuming a unit disadvantage factor.

The calculated ratio of xenon to U^{238} captures in region 3 was less than 0.03. The initial breeding ratio quoted in Table 7 includes the estimated effect of the parasitic xenon captures.

Table 7

SUMMARY OF PHYSICS REFERENCE PROBLEM

Core Volume	Critical Mass PuO_2	Fraction of Power in Thermal Fissions	Neutron Leakage ^a	Initial Breeding Ratio ^b
306.5ℓ	275.0 kg	0.213	0.006	1.44

^aNormalized to one fission neutron in the fast core.

^bNot counting U^{235} burned in depleted uranium and including the estimated effect of the stainless steel separator and the Xe poisoning in the thermal core.

The fast effect due to the geometric clustering of the pressure tubes is expected to be small, since the region between the tubes is almost all zirconium. Equation A-10 was used to calculate ϵ for a mixture having the composition of the contents of a pressure tube and for a mixture corresponding to homogenizing the entire pressure tube region. The calculated values of ϵ for the thermal core were, respectively, 1.091 and 1.070. The correct value would be expected to lie between these two limiting values. The value

deduced from the 2D calculation for the thermal core (region 3) was 1.083. The neutron inventory and quoted initial breeding ratio given in Tables 7 and 8 were based on the 2D calculation. The rather tight limits on ϵ indicate that the quoted values can not be seriously in error owing to use of the 2D results.

The influence of the stainless steel separator on the neutron inventory was estimated by means of the calculated fluxes at radial position 40.0 cm (the radial outerbound of the separator). The absorptions in the steel were calculated assuming there existed a constant radial flux through the separator equal in magnitude to the 40.0-cm radial value. The axial average was obtained using the 4-group, 2D calculated fluxes. The neutron inventory in Table 8 includes the estimated effect of the steel separator captures.

D. Results

The 13-group fluxes have been collapsed to the equivalent 4-group structure. Thus,

$$\sum_{j=2}^6 \phi_j = \Phi_2$$

and

$$\sum_{j=7}^{12} \phi_j = \Phi_3 \quad .$$

The highest energy and thermal groups are the same in the 13 and 4-group structure. Figure 8 gives the calculated radial 13-group, 1D flux distributions at the axial midplane and permits a comparison with the 4-group, 2D results. The agreement is seen to be very good except for the group 3. The 4-group thermal flux at the fast core-buffer interface is also seen to be higher than the 13-group thermal value. This is reflected in the much higher power spike at the core outer edge seen in Fig. 9, which compares the power distributions for the two calculations. The thermal (and total) power in the thermal core is also higher in the 4-group calculation. These discrepancies may indicate that the resonance-capture considerations in the 4-group calculation need to be refined. In particular, the $\Sigma_{3 \rightarrow 4}$ cross section in the fast core appears to be too large, causing a depression of Φ_3 and an increase in Φ_4 .

Table 8

NEUTRON INVENTORY FOR PHYSICS REFERENCE SYSTEM

Event*	Region							
	Fast Core	Radial Buffer	Fast Axial Blankets	Stainless Separator	Thermal Core	Radial Thermal Blankets	High Density Radial Thermal Blankets	Thermal Axial Blankets
Pu ²³⁹ Fissions	0.292	--	--	--	--	--	--	--
Pu ²³⁹ Captures	0.086	--	--	--	--	--	--	--
U ²³⁵ Fissions	0.004	0.003	0.002	--	0.067	0.007	0.024	0.010
U ²³⁵ Captures	0.002	0.001	0.0008	--	0.015	0.002	0.006	0.002
U ²³⁸ Fissions	0.045	0.006	0.003	--	0.008	0.002	0.008	0.002
U ²³⁸ Captures	0.200	0.055	0.056	--	0.128	0.042	0.131	0.054
Fe Captures	0.007	0.008	0.003	~0.019	--	--	--	--
Zr Captures	--	--	--	--	0.013	0.004	0.007	0.003
H ₂ O Captures	0.0002	0.0001	0	--	0.011	0.005	0.003	0.003

Event*	Region			
	Fast Grid Regions	Fast Gas Spaces	Thermal Grid Regions	Thermal Gas Spaces
Pu ²³⁹ Fissions	--	--	--	--
Pu ²³⁹ Captures	--	--	--	--
U ²³⁵ Fissions	--	--	--	--
U ²³⁵ Captures	--	--	--	--
U ²³⁸ Fissions	--	--	--	--
U ²³⁸ Captures	--	--	--	--
Fe Captures	0.003	0.001	--	--
Zr Captures	--	--	0.001	0.0009
H ₂ O Captures	0	0	0.0007	0.0009

Neutrons Leaking = 0.006

* All numbers normalized to one fission neutron in the fast core.

FIG. 8
FOUR-GROUP, 2D AND CONTRACTED THIRTEEN-GROUP,
1D RADIAL FLUX DISTRIBUTIONS AT MIDPLANE

The critical plutonium masses calculated for the 13-group, 1D and 4-group, 2D were, respectively, 222 kg and 242 kg. Not all of this 9% discrepancy need be due to inconsistencies in the cross sections, since the assumed reflector savings of 15 cm at each end for the 1D calculation may be in error.

Figures 10 and 11 show, respectively, the 4-group, 2D axial flux and power distributions at the central axis. Figures 12 and 13 show the axial flux and power distributions at a radial position of 45.0 cm, which is

near the center (radially) of the thermal core. The asymmetry seen in Figs. 10 and 11 is due to the difference in water content in the upper and lower fast axial blankets (regions 7 and 12), which is due to the different steam conditions.

Table 7 gives a summary of the 4-group, 2D results and Table 8 gives the neutron inventory as calculated using the 4-group, 2D problem. As seen in Table 8, the water and zirconium contribute approximately equally to the parasitic losses in the water-cooled zones.

FIG. 9
COMPARISON OF FOUR GROUP AND THIRTEEN GROUP
POWER DISTRIBUTIONS AT MIDPLANE

FIG. 10
AXIAL FLUX DISTRIBUTION AT CENTRAL AXIS

FIG. 11
AXIAL POWER DISTRIBUTION AT CENTRAL AXIS

FIG. 12
AXIAL FLUX DISTRIBUTION AT 45.0 cm RADIAL POSITION

FIG. 13
AXIAL POWER DISTRIBUTION AT 45.0 cm RADIAL POSITION

E. Conclusions

The results given above indicate that a reactor embodying the present concept should exhibit a breeding ratio of the order of 1.4. Some of the factors contributing to this result are the large power fraction in the fast part of the coupled system, the small water density in the steam zones, and the relatively small water volume ratio in the water-moderated thermal parts of the system.

The small water density of the steam zones results in a sufficiently hard spectrum so that the plutonium capture-to-fission ratio (α) is still quite low. In particular, the spatially integrated fluxes for the fast core from the 13-group calculation yield an average plutonium α for the fast core of 0.227. Fundamental mode calculations have shown that steam-cooled systems having a water density comparable to the present system yield spectra not very different from equivalent metal-cooled systems.

The water and zirconium captures in the water regions are rather small owing to the small water volume fraction as compared, say, with a conventional light water-moderated thermal reactor. This relatively low parasitic loss, of course, contributes to the high breeding ratio.

The disagreement between the 4 and 13-group result has yet to be resolved. One practical difficulty in this regard is that the engineering heat transfer calculations rely on the two-dimensional, four-group calculations for information about power peaking. The 13-group, 1D result for the radial maximum to average power in the fast core is 1.32. The engineering design, on the other hand, was done using a 4-group 2D problem, which indicated a radial maximum-to-average power in the fast core of 1.42. This discrepancy directly influences the fast core power and hence the overall economic aspect of the system. The engineering design has in fact assumed the more conservative maximum-to-average core power ratio.

V. ENGINEERING ANALYSIS

A. Introduction

The objectives for attaining a high heat transfer are to obtain high power density, high steam throttle temperature, and large thermal efficiency. The performance of the system is considered in the light of presently available technology.

The steam pressure chosen is within the pressure range of commercially available 65-Mwe turbine-generator units.

The initial heat transfer calculations indicated that with an oxide fuel and a cladding surface temperature limit of 720°C it was impractical to obtain 565°C superheated steam with 1000 kw/ℓ and 453°C steam, respectively. The conditions imposed on the fuel and cladding are summarized in Table 9.

Table 9

FUEL DESIGN CRITERIA

Flow pattern	Axial
Fuel rod spacing	Equilateral pitch
Minimum rod diameter	0.318 cm
Oxide density, g/cc	
PuO ₂	10.5
UO ₂	10.2
Steam-cooled zones	
Clad material	Stainless steel, Type 347
Maximum temperature, °C	
Surface	720
Oxide	2500
Pressurized water zones	
Clad material	Zircaloy-2
Maximum temperature, °C	
Surface	316
Oxide	2500
Uranium	650

B. Heat Transfer

The power distribution are based on a four-group, 2D, IBM-704 calculation. Typical examples are shown in Figs. 9, 11, and 13. The maximum-to-average ratios from the power distributions were combined with hot spot factors to yield the hot channel factors listed in Table 1.

The flow path of the steam was selected to be downward through the hexagonal cans to provide a means of holding down the cans and sealing them to the grid. The flow is orificed to provide equal steam outlet temperatures. The steam also surrounds the pressure tubes, which are cooler than the steam, so that condensation takes place. Because the pressure tubes do not have a geometrical arrangement conducive to good heat exchange, only 0.5 Mwt is transferred to the pressurized water. The choice of accepting the condensation was made in the interest of using a pressure-balance system.

The pressurized water flow scheme is essentially a two-pass flow arrangement through the pressure tubes from the inlet and outlet headers, as shown in Fig. 2. The 312 pressure tubes which make up the thermal zones are arranged in pairs. Local boiling is allowed in the channels of the thermal zones to permit using a high inlet water temperature. The average voids resulting from local boiling are estimated, based on the work of Griffith,⁽⁶⁾ to be no more than about 3%. Net steam generation is not permitted in the thermal zones during normal operation.

The heat transfer correlation⁽⁷⁾ used for the steam film coefficient in the steam-cooled areas is

$$hD/K = 0.0214 \left[1 + \frac{2.3}{L/D} \right] \left[\frac{GD}{\mu_f} \right]^{0.8} \left[\frac{\mu_{Cp}}{K} \right]_f^{1/3}, \quad (5)$$

where the symbols are defined in the reference.

The physical properties of steam used in this correlation were obtained from Ref. (8). The correlation has an accuracy of $\pm 17\%$. The lower limit was used.

The maximum internal and surface temperatures used in the design of the fuel are listed in Table 9. The temperatures were calculated on the basis of the four-group, 2D, IBM-704 results and on the hot channel factors listed in Table 1. An estimation of the variation in the surface temperature around the periphery of a rod at a horizontal cross section was made based on the work of Deissler.⁽⁹⁾

The oxide fuel radial temperature difference was calculated on the basis of the parameter

$$\int K(\theta) d\theta,$$

which yields a higher apparent conductivity⁽¹⁰⁾ than the value of 0.017 watt/(cm)(°C), which is an average conductivity sometimes used. For the results presented here the conductivity of UO_2 was assumed to apply also to PuO_2 and varied according to the following equation:

$$K = 57/T \quad , \quad (6)$$

where

K = thermal conductivity, watts/(cm)(°C)

T = temperature, °K.

By using the parameter

$$\int K(\theta) d\theta$$

between the expected operating temperature limits and combining this with the required volume fractions, a relation between the number of rods and the diameter of the rods was obtained.

In all cases the equivalent film coefficient between the pellet and the inside of the rod was assumed to be 0.57 watt/(cm²)(°C).

In the thermal zones the maximum surface temperature was based on fully developed local boiling. The surface temperature is equal to the saturation temperature plus the quantity ΔT_{sat} .⁽¹¹⁾

For determining the burnout limits the Jens and Lottes burnout correlation⁽¹¹⁾ was used. The constants were interpolated for the operating pressure of 95 atm and, since the equation involves many unknowns, the allowable burnout heat flux was reduced by two-thirds. The maximum outlet temperature was fixed at 308°C, as based on a 15% power increase. The 15% power increase coincides with the power increase necessary to cause a fuel element center temperature of 2800°C in the thermal core. A minimum safety factor of 2 on burnout heat flux was specified. The final result from these heat transfer considerations fixed the maximum coolant velocity at 914 cm/sec. All the pressure tube flows were adjusted for approximately equal temperature rises of the water by orificing.

Since the high-density radial blanket, Zone 7, consists of pressure tubes with widely varying power densities, a number of hot channel factors were used. The factors are consistent with the values used in the other water zones.

The heat transfer analysis of the core resulted in the coolant velocities, rod sizes, and temperatures for the various regions listed in Table 1.

The heat transfer analysis leading to the prediction of core performance is considered to be optimistic in the steam-cooled regions. This optimism is justified on the basis that future research and development work in the field of steam heat transfer for analytical and experimental

points of view will be performed to establish the true limits of cooling the core. In the pressurized water zones the design is consistent with established technology.

The results of the heat transfer calculations for the concept showed that the power capability of the fast steam-cooled core is limited by the allowable surface temperature. This design used small rods, 0.32 cm in diameter, in order to provide increased surface for increased power density. The use of small rods resulted in increased estimated costs of fuel fabrication. In order to improve the situation, an economical high-temperature, high-creep strength, steam corrosion-resistant cladding material will be required. Coincident with this development item is the necessity of providing a high-conductivity, high-melting point fuel with good burnup and fission gas retention properties. Until these items are developed, the fast steam-cooled core will not be capable of producing the high power density and high steam temperature necessary for an economical system.

C. Steam Cycle (Figs. 2 and 14)

The method of producing electric power from the available steam and pressurized water was studied with the objectives of producing a system that had good startup, operating, transient, and shutdown characteristics as well as a high cycle efficiency.

The basic problem encountered with the reactor system is that the power produced in the thermal regions is only 25% of the total reactor power. This fraction of the power could only provide about 10% of the required saturated steam flow rate. The superheated steam leaving the core cannot be used directly to vaporize the other 90% of the required steam flow rate because the pressure loss through the core causes a temperature crossover point to occur that limits the amount of heat that can be transferred. An external boiler could be used, but the power of the boiler would have to have a capacity of about 400 Mwt. This exceeds the entire power output of the reactor and tends to negate the whole engineering concept. Other ideas along this line were investigated and rejected on the same basis.

The cycle recommended is shown in Figs. 2 and 14. It utilizes a steam compressor to compress about two-thirds of the reactor steam from 75 atm, the reactor steam outlet pressure, to 95 atm, the reactor inlet pressure. The steam compressor supplies the required pressure buildup to permit using reactor steam for generating the inlet steam conditions. The compressed steam flows through a mixer-pressurizer and generates the inlet steam conditions.

The steam compressor and its drive unit can be built with present technology. The efficiency of the compressor is expected to be in the range

FIG. 14
HEAT BALANCE

between 70% and 80%. In the cycle analysis, an efficiency of 75% was used for the compressor and 85% was assumed for the turbine drive unit.

The type of turbine-generator unit for electric power generation would be a tandem compound double flow 66,000-kw AIEE-ASME preferred standard. The units have provisions for five feedwater extraction stages.

The heat balance for the steam cycle is shown in Fig. 14 with the pertinent data listed. The overall steam cycle efficiency is 30.4%, which takes into account the power expended in driving the steam compressor. The electrical output of the system is 65,500 kw.

D. Startup and Shutdown (Fig. 2)

1. Startup

One of the basic requirements of the steam and water systems is that the entire system must provide a method of bringing the reactor to power with a minimum of auxiliary steam and power supplies. The system shown in Fig. 2 fulfills the requirement because it allows the reactor to increase its steam flow and power by reactor heat with a small startup auxiliary steam supply. The auxiliary steam supply would be capable of supplying saturated steam at 95 atm and also the necessary steam shutdown flow. The auxiliary steam supply can be made available within a short time interval, but normally delivers steam only at startup or shutdown.

Thermal shocking of the system and steam condensation within it are avoided by using the auxiliary steam supply to heat a separate hot gas preheat system. The hot gas system would be used until the core and vessel temperature were raised to 150°C. Steam would then be fed to the system to bring it to operating temperature.

The steam and water systems are connected in a manner which automatically permits the pressures in both systems to equalize. This connection, as shown in Fig. 2, goes from the mixer-pressurizer to the steam dome of the pressurizer in the water system. With this arrangement, the pressure in both systems tends to equalize. Slight differentials exist because of dynamic pressure losses in each system. The pressurized water system can be heated by absorption of the pump power provided the coolant is circulated without normal heat removal.

Once the entire system is at pressure and temperature, the reactor can be made critical and slowly brought to power. In the course of going to power, steam is fed to the compressor and its turbine drive unit. The compressed steam will then vaporize additional steam for cooling the fast core and its blankets. The remaining steam not used by the compressor and its drive can be fed to the condenser and feedwater stages. When sufficient power is developed the turbine can be put into service.

2. Shutdown

The shutdown of the entire system requires removal of the fueled portions of the control and shutdown rods. The steam flow to the core is decreased to the maximum flow of the auxiliary steam supply. As the decay heat in the steam-cooled zones decreases, the auxiliary steam flow rate, temperature, and pressure are decreased and the system is gradually cooled down. The pressure in the thermal regions will decrease as the pressure is lowered in the steam system. When the pressurized water system power is reduced sufficiently, a small pump will circulate the coolant to remove decay heat.

In an emergency shutdown situation, such as steam compressor failure, the pressurized water system and the feedwater side of the pressurized water heat exchanger contain a large volume of water that could flash to steam and pass through the steam-pressurizing line and thus cool the steam zones. An orificed coolant supply to an emergency condenser would condense the steam and return the condensate to the feedwater side of the pressurized water heat exchanger, which also supplies steam by flashing. An auxiliary powered pump would take suction from this reservoir of water and make up the loss in the pressurized water system.

VI. RESEARCH AND DEVELOPMENT

The technical feasibility of the concept is believed to be a real possibility, but ascertaining the final design and performance of the system requires a research and development program. The program should serve to establish the design and potential of the concept to a better degree. The areas which require a research and development program are those involved with heat transfer, materials, physics, and hazards analyses.

Development of a fuel with a high thermal conductivity, high melting point, good fission gas retention properties, and compatibility with steam is essential to the future success of a direct steam-cooled reactor that produces high-temperature steam. Investigations are being conducted now to provide good fission gas retention, and possibly these methods can be used with a high-conductivity, high-melting point fuel that will tolerate the unintentional contact of steam. From the compatibility, gas retention, burnup and melting standpoints, UO_2 is satisfactory but fails with regards to conductivity. Metal fuels are limited to about 700°C and therefore cannot be used even if a high-temperature cladding is developed.

The fuel elements specified in the conceptual design are made of Type 347 stainless steel. A practical present limit is about 720°C . Methods of increasing this limit by reducing corrosion and erosion and also by increasing creep strengths should be explored. The usual autoclave or non-heat transfer type of test could be used for screening purposes. Once a promising technique or material is found, corrosion, erosion, and creep strength investigations should be conducted with heat transfer and fluid flow patterned after the conditions specified in this report.

Extensive study is required to substantiate the physics analysis. As indicated in Section IV, the physics calculations require refinements with regard to obtaining a valid few-group representation of the problem. This is of practical importance, since the engineering results rely heavily on the few-group physics calculations. The kinetic behavior of the system must be exhaustively studied in order to understand the safety and operating peculiarities of the system.

Because of the unusual nature of the proposed concept, experimental verification of the physics calculations is essential. Whether such verification requires a power demonstration experiment or can be achieved more simply by means of zero power experimentation can be determined only after more thorough examination of the system.

Appendix

PHYSICS

A. Introduction

The present section describes the exploratory physics calculations which have preceded and led to the concept as presented in the main body of the report. In all of the early systems metal rather than oxide fuels were used. The original concept assumed that all of the steam would be produced in the thermal region of the reactor, which would, therefore, be a more or less conventional boiling system. Because of the thermodynamics of steam, if the thermal region were the sole source of saturated steam, it would be difficult to achieve a system having more than about 25% of the power in the fast superheater.

This is illustrated by a specific numerical example in Fig. A-1 where the characteristics of a boiling-superheating system are given. The breeding ratio attainable with such a system would not be expected to approach the values common for all-fast reactor systems. On the other hand, depending upon the extent to which auxiliary superheated steam-powered boilers were employed, the thermal regions could be used to produce only part of the saturated steam which is to be superheated. In particular, if the thermal region were used only as a nonboiling pressurized water preheater for the auxiliary boilers, so that the fast part of the system contained most of the power, breeding ratios comparable with those of the fast liquid metal systems might be attainable.

B. Reference Systems

The systems examined in this Appendix are examples of the coupled superheater concept which represent the extremes discussed in the Introduction. The first is one in which all of the steam which is to be superheated is produced in the boiling, thermal portion of the reactor. Since for this system only about 25% of the power will be produced in the fast parts of the reactor, this system will be designated the thermal-fast system. The second system is one in which none of the steam is generated in the reactor. In this case, the thermal portion of the coupled reactor will act as a pressurized water-preheater for external boilers, and about 80% of the power will be generated in the fast portion of the reactor. This system will be designated the fast-thermal system.

Figures A-2 and A-3 show schematic drawings indicating the relative dimensions of the thermal-fast and fast-thermal coupled steam superheaters, respectively. Only heavy water for the coolant has been examined for the case of the thermal-fast system. The fast-thermal concept, on the

FIG. A1

CHARACTERISTICS OF A BOILING-SUPERHEATING SYSTEM ASSUMING 100 MW
IN THE SUPERHEATER AND 593°C SUPERHEATED STEAM

other hand, has been investigated both for heavy water and light water coolants. In the light water case, the radial thickness of region 4 in Fig. A-3 is only about half as great as in the heavy water case. The relative dimensions shown in Fig. A-3 are as used in the heavy water studies. For purposes of comparison, a prototype, all-fast, sodium-cooled reference system has also been examined. This system is shown schematically in Fig. A-4.

FIG. A2
 SCHEMATIC DRAWING OF
 THERMAL-FAST COUPLED
 STEAM SUPERHEATER

FIG. A3
 SCHEMATIC DRAWING OF
 FAST-THERMAL COUPLED
 STEAM SUPERHEATER

In the thermal-fast concept, since only about 25% of the power will be generated in the fast part of the system, the thermal portion must be highly reactive; a large k_{∞} is required for the thermal system. For this reason regions 4 and 5 of Fig. A-2 contain clusters of fuel. This clustering increases the resonance escape probability as compared with a non-clustered fuel arrangement. In the case of the fast-thermal system, on the other hand, since the thermal part of the system need not be highly reactive, nonclustered fuel arrangements may be used. The fuel cluster arrangement assumed for the thermal-fast concept is shown in Fig. A-5. The cluster consists of 55 zirconium-clad uranium pins located within a double zirconium shroud. The dimensions of the cluster are given in Table A-1. The clusters contain natural uranium in region 4 and depleted uranium in region 5 (Fig. A-2). The uranium, zirconium clad, and boiling D_2O are at a temperature of $300^{\circ}C$. The nonboiling D_2O and zirconium shrouds are at a temperature of $100^{\circ}C$. (The region between the shrouds contains nonboiling D_2O .) In the case of the boiling D_2O , steam voids were taken into account by assuming the density to be 0.65 times that of non-boiling D_2O at $300^{\circ}C$.

1 - SODIUM COOLED PLUTONIUM FAST CORE
2 - SODIUM COOLED DEPLETED URANIUM BLANKET

FIG. A4
SCHEMATIC DRAWING OF ALL-FAST REFERENCE SYSTEM

FIG. A5
FUEL CLUSTER USED IN THERMAL-FAST STEAM SUPERHEATER

Table A-1

FUEL CLUSTER USED IN THE THERMAL-FAST STEAM SUPERHEATER

	cm
Radius of Natural Uranium Pin	0.1588
Thickness of Zirconium Clad	0.0254
Triangular Pitch of Pins	0.6223
Inner Shroud, ID	4.953
Inner Shroud, OD	5.055
Outer Shroud, ID	5.258
Outer Shroud, OD	5.461

The composition of the various regions for the reference systems are given in Tables A-2 through A-5. Tables A-3 and A-4 represent, respectively, heavy and light water versions for the fast-thermal concept. Table A-2 refers to Fig. A-2, Tables A-3 and A-4 to Fig. A-3, and Table A-5 to Fig. A-4.

Table A-2

COMPOSITION OF VARIOUS REGIONS OF THERMAL-FAST D₂O COUPLED STEAM SUPERHEATER

Region	Volume Fractions					
	Pu ²³⁹	U ²³⁸	U ²³⁵	Fe	Zr	D ₂ O ^a
Core	0.0447	0.4553	0	0.15	0	0.0084
Inner Blanket	0	0.4966	0.0036	0	0.1729	0.1559
Water Annulus	0	0.0461	0.0003	0	0.0427	0.8362
Water Blanket	0	0.0463	0.0001	0	0.0427	0.8362
Inner Blanket						
Reflector	0	0.4990	0.0010	0	0.1729	0.1559
Core Blanket	0	0.4990	0.0010	0.15	0	0.0084

^aEquivalent amount of D₂O at 100°C.

Table A-3

COMPOSITION OF VARIOUS REGIONS OF FAST-THERMAL
D₂O COUPLED STEAM SUPERHEATER

Region	Volume Fractions					
	Pu ²³⁹	U ²³⁸	U ²³⁵	Fe	Zr	D ₂ O ^a
Core	0.0531	0.4469	0	0.15	0	0.0114
Core Blanket	0	0.4990	0.0010	0.15	0	0.0114
Water Annulus	0	0.0993	0.0007	0	0.0102	0.8898
Water Blanket	0	0.0998	0.0002	0	0.0102	0.8898

^a Equivalent amount of D₂O at 300°C.

Table A-4

COMPOSITION OF VARIOUS REGIONS OF FAST-THERMAL
H₂O COUPLED STEAM SUPERHEATER

Region	Volume Fractions					
	Pu ²³⁹	U ²³⁸	U ²³⁵	Fe	Zr	H ₂ O ^a
Core	0.0487	0.4513	0	0.15	0	0.0114
Core Blanket	0	0.4990	0.0010	0.15	0	0.0114
Water Annulus	0	0.19856	0.00144	0	0.02048	0.77952
Water Blanket	0	0.7984	0.0016	0	0.08192	0.11808

^a Equivalent amount of H₂O at 300°C.

Table A-5

COMPOSITION OF REGIONS OF ALL-FAST
REFERENCE SYSTEM

Region	Volume Fractions				
	Pu ²³⁹	U ²³⁸	U ²³⁵	Fe	Na ^a
Core	0.0460	0.4540	0	0.15	0.35
Blanket	0	0.4990	0.0010	0.15	0.35

^a At 480°C.

C. Calculational Procedures

1. General

Calculations were made using four-group diffusion theory, both in one and two-dimensional (r-z) geometry. The one-dimensional (1D) calculations were done with UNIVAC⁽³⁾ and assumed a bare axial height, including an assumed reflector savings of 15 cm on each end. The two-dimensional (2D) problems were studied with the IBM-704 using the PDQ code.⁽⁴⁾

In the fast parts of the system (in general the steam-cooled regions) the regional cross sections were obtained by a simple homogenization of the basic cross section set (Table A-6). This procedure is considered to be valid because of the hardness of the spectrum in the steam-cooled zones. In the water zones special attention was given to calculation of the resonance escape probability and, in the case of clustered elements, to calculation of fast effect and thermal disadvantage factors. In the case of the nonclustered lattices in the water regions of the fast-thermal systems, thermal cross sections were obtained by simple homogenization.

Table A 6
4-GROUP MICROSCOPIC CROSS SECTIONS (barns)

J	Lower Energy E _L	Fission Spectrum	Pu ²³⁹					U ²³⁸ ^a					U ²³⁵					Fe		
			σ_{tr}	σ_f	ν	σ_c	$\sigma_{j \rightarrow j+1}$	σ_{tr}	σ_f	ν	σ_c	$\sigma_{j \rightarrow j+1}$	σ_{tr}	σ_f	ν	σ_c	$\sigma_{j \rightarrow j+1}$	σ_{tr}	σ_c	$\sigma_{j \rightarrow j+1}$
1	1.353 Mev	0.575	4.8	2.00	3.07	0.06	0.9	4.70	0.53	2.65	0.04	2.10	4.70	1.30	2.65	0.100	1.40	2.14	0.00249	0.7372
2	9.12 kev	0.425	7.0	1.80	2.96	0.45	0	7.00	0	0	0.18	0	7.00	1.40	2.55	0.300	0	2.93	0.00570	0.0025
3	0.4 ev	0	74.0	41.00	2.90	23.0	0.01	11.01	0	0	1.037	0.0085	56.21	25.94	2.55	20.3	0.0085	9.62	0.127	0.0331
4	thermal	0	1055.3	687.40	2.90	358.3	0	10.18	0	0	1.738	0	444.17	365.83	2.50	68.40	0	11.22	1.800	0

J	H ₂ O			D ₂ O			Zr			B ¹⁰			Na ^b			10 ⁷ /v
	σ_{tr}	σ_c	$\sigma_{j \rightarrow j+1}$	σ_{tr}	σ_c	$\sigma_{j \rightarrow j+1}$	σ_{tr}	σ_c	$\sigma_{j \rightarrow j+1}$	σ_{tr}	σ_c	$\sigma_{j \rightarrow j+1}$	σ_{tr}	σ_c	$\sigma_{j \rightarrow j+1}$	
1	3.08	0	2.81	4.413	0	5.393	3.00	0.020	0.800	1.600	0.282	0.188	2.00	0.0005	0.300	0.00412
2	10.52	0	4.04	7.864	0	1.140	7.395	0.020	0.0738	4.839	1.714	0.131	3.885	0.00045	0.0676	0.0204
3	16.55	0.035	4.14	8.129	0	0.537	7.395	0.020	0.0369	194.25	190.52	0.0747	8.988	0.0550	0.0773	2.267
4	48.95	0.403	0	11.329	0.00132	0	6.139	0.1146	0	1971.20	1967.51	0	3.532	0.279	0	28.97

^a U²³⁸ as given was used for the all fast reference problems. For the various superheater calculations the group three values were adjusted for each problem for the various water zones. In the steam-cooled regions the values given in the table were used in the case of the D₂O studies. For the H₂O cases group three for U²³⁸ had $\sigma_{tr} = 11.23$ and $\sigma_c = 1.257$ barns.

^b The sodium is at 480°C. All other materials are at 300°C.

Material	Atomic Density (atoms/cc)
Pu ²³⁹	0.040 x 10 ⁺²⁴
U ²³⁸	0.048
U ²³⁵	0.048
Fe	0.0848
H ₂ O	0.02375
D ₂ O	0.0234
Zr	0.0425
B ¹⁰	0.1499
Na	0.0220

The four-group equations which were solved were written as

$$\begin{aligned}
 D_1 \nabla^2 \phi_1 - \Sigma_{a1} \phi_1 + \beta_1 S &= 0 \\
 D_2 \nabla^2 \phi_2 - \Sigma_{a2} \phi_2 + \Sigma_{1 \rightarrow 2} \phi_1 + \beta_2 S &= 0 \\
 D_3 \nabla^2 \phi_3 - \Sigma_{a3} \phi_3 + \Sigma_{2 \rightarrow 3} \phi_2 &= 0 \\
 D_4 \nabla^2 \phi_4 - \Sigma_{a4} \phi_4 + \Sigma_{3 \rightarrow 4} \phi_3 &= 0 \quad ,
 \end{aligned}
 \tag{A-1}$$

where

$$S = \sum_{j=1}^4 \nu_j \Sigma_{fj} \phi_j \quad . \tag{A-2}$$

The D_i are the group diffusion coefficients and are given by $(3\Sigma_{tr,i})^{-1}$, where the $\Sigma_{tr,i}$ are the group transport cross sections. The $\Sigma_{a,i}$ are the total group removal cross sections and the $\Sigma_{i \rightarrow i+1}$ are the group transfer cross sections. The β_i give the fraction of the fission neutrons born into group i . We note that neutrons are assumed to be born only in groups 1 and 2 (with $\beta_1 + \beta_2 = 1$).

In the water regions, the resonance escape probability p is taken into account by assuming $p\Sigma_{a3} = \Sigma_{3 \rightarrow 4}$, so that in the fourth equation $\Sigma_{3 \rightarrow 4}$ is replaced by the value of $p\Sigma_{a3}$.

In the case of the clustered elements, all of the thermal cross sections were obtained by a spatial flux weighting. A diffusion theory cell calculation was used for evaluating the thermal disadvantage factors. As usual, the gradient of the flux was set equal to zero at the outer boundary of the cell moderator region. The sources in fuel and moderator regions of the cell were assumed to be proportional to the amount of D_2O present in each region. The composition of the fuel and moderator zones for the cell calculations used in the clustered geometries are given in Table A-7.

Table A-7

COMPOSITION OF FUEL AND MODERATOR REGIONS FOR
CELL CALCULATIONS IN CLUSTERED FUEL ZONES^a

<u>Fuel Region</u>		<u>Moderator Region</u>	
<u>Material</u>	<u>Volume Fraction</u>	<u>Material</u>	<u>Volume Fraction</u>
Uranium ^b	0.2360	Boiling D_2O	0.0372
Zirconium	0.0816	Nonboiling D_2O	0.9305
Boiling D_2O	0.6824 ^c	Zirconium	0.0323

^aCluster pitch = 10.414 cm.

^bNatural uranium in region 4; depleted uranium in region 5 (Fig. A-2).

^c0.65 times this value used for homogenizing cross sections.

2. Resonance Escape Probability

The resonance escape probability was calculated using the expression

$$p = \exp \left\{ - \frac{N}{\bar{\xi} \Sigma_s} \right\} I_{\text{eff}} \quad , \quad (\text{A-3})$$

where

$$\bar{\xi} = \frac{\sum_i \xi_i \Sigma_{s_i}}{\sum_i \Sigma_{s_i}} = \frac{\sum_i \xi_i \Sigma_{s_i}}{\Sigma_s} \quad (\text{A-4})$$

and the summation extends over all materials of the mixture. Here N is the number of absorber atoms per cc, ξ_i is the average logarithmic energy loss suffered by a neutron per collision with material i , and Σ_{s_i} is the resonance group scattering cross section of the material i . The same expression was used both for the clustered and nonclustered geometries, so that resonance disadvantage factors were ignored in the clustered case.

For the U^{238} , the effective resonance integral I_{eff} was evaluated by means of the formula given by Hellstrand:⁽¹²⁾

$$I_{\text{eff}} = \frac{7.8}{F} + 30.2 \frac{S_{\text{eff}}}{M} \quad , \quad (\text{A-5})$$

where F ⁽¹³⁾ is the ratio of resonance energy flux at the rod surface to average resonance energy flux in the rod interior. Further, F was evaluated using constants appropriate to the resonance energy group. A 1.4-barn $1/v$ correction was added to the values obtained using Eq. A-5 before inserting in Eq. A-3.

For the nonclustered lattices, S_{eff} has been evaluated following the method of Dancoff and Ginsburg.^(14,15) The actual values for the Dancoff correction factor, $1 - C$, for two rods of radius ρ and separation d immersed in a medium of mean free path λ , were obtained using an approximate formula similar to that given by Thie,⁽¹⁶⁾ namely,

$$1 - C = 1 - \frac{4}{\pi^2} \sin^{-1} \frac{\rho}{d} \text{Ki}_3 \left[\left(\frac{d}{\rho} - 1.78 \right) \frac{\rho}{\lambda} \right] \quad . \quad (\text{A-6})$$

The Ki_3 function was evaluated using tabulated values obtained from a recent IBM-704 program.⁽¹⁷⁾

For the clustered geometries, the empirical method of Carlvik and Pershagen^(12,18) was used to obtain the value of S_{eff} for use in Eq. A-5. In this case, the interaction between the various clusters was ignored.*

In both the nonclustered and clustered geometries, the resonance escape probabilities for the U^{235} and Zr were evaluated using the infinite dilution values of the resonance integrals.⁽⁵⁾ For U^{235} , the total I_{eff} was obtained from the fission I_{eff} by multiplying by $1 + \alpha$, where $\alpha = 0.783$ is the assumed capture-to-fission ratio for the U^{235} in the resonance group 3.

The total resonance escape probability was taken as

$$p = (p^{28})(p^{25})(p^{\text{Zr}}) \quad . \quad (\text{A-7})$$

The individual materials were assumed to have group three capture plus fission cross sections given by $N_i I_{\text{eff}_i} / \Delta u$, where Δu is the lethargy width of group 3 (namely, 10.03), N_i the number of atoms per cubic centimeter for the isotope under consideration, and I_{eff_i} the corresponding effective resonance integral. Then, the group 3 to 4 transfer cross section is taken to be

$$\Sigma_{3 \rightarrow 4} = \frac{p}{1-p} \sum_i \frac{N_i I_{\text{eff}_i}}{\Delta u} \quad . \quad (\text{A-8})$$

Sigma transport was taken to be

$$\Sigma_{\text{tr}_i} = \frac{N_i I_{\text{eff}_i}}{\Delta u} + (1 - \bar{\mu}_i) \Sigma_{s_i} \quad . \quad (\text{A-9})$$

For the light water lattices, an effective resonance integral for H_2O of 0.35 barn was used. As in the clustered arrangement, the total resonance escape probability to be used in Eq. A-8 was taken as the product of the individual escape probabilities.

3. Fast Effect

In a homogeneous system, the fast fission bonus will be automatically accounted for in a multigroup calculation if values for the U^{238} fission cross section are included in the multigroup cross sections. In the case of a one-group calculation, on the other hand, the fast fission factor ϵ may be accounted for by using a homogeneous formula such as⁽¹⁹⁾

*For example, in the case of a 10.414-cm cluster pitch, if one performs a Dancoff-Ginsburg calculation considering the cluster as a large rod, a value $1 - C = 0.96$ is obtained.

$$\epsilon = 1 + \frac{\frac{\bar{\Sigma}_f^{28}}{\Sigma_r} (\nu^{28} - 1 - \alpha^{28})}{1 - \nu^{28} \frac{\bar{\Sigma}_f^{28}}{\Sigma_r}} \quad (\text{A-10})$$

and including the value of ϵ in the four factor formula ($k_\infty = \epsilon p f \eta$). In Eq. A-10, the factor $\bar{\Sigma}_f^{28}$ is the U^{238} fission cross section averaged over the fission spectrum, Σ_r is the total cross section for removing neutrons from above the U^{238} fission threshold, ν^{28} is the number of fission neutrons released per U^{238} fission, and α^{28} is the ratio of U^{238} captures to U^{238} fissions in the region above the U^{238} threshold.

In the case of a heterogeneous system, the geometric enhancement of ϵ must be considered separately, since, even though the calculation may be of a multigroup type, it is still a homogeneous calculation.

The fast fission factor is commonly calculated for a heterogeneous system by means of the formula⁽²⁰⁾

$$\epsilon = 1 + \frac{(\nu^{28} - 1 - \alpha^{28}) \left(\frac{\bar{\Sigma}_f^{28}}{\Sigma_{tr}} \right) P_1}{1 - (\nu^{28} \bar{\Sigma}_f^{28} + \Sigma_e) \frac{P_2}{\Sigma_{tr}}} \quad (\text{A-11})$$

In Eq. A-11, the term Σ_{tr} is the transport cross section for the fuel element, and Σ_e is the elastic scattering cross section, which is taken to be the transport cross section minus the capture plus fission plus inelastic cross sections. The P_1 and P_2 are first and second generation collision probabilities.⁽²¹⁾ One way to account for ϵ in a heterogeneous multigroup calculation would be to calculate ϵ using Eq. A-11 and then to perform the multigroup calculation setting ν^{28} equal to zero and replacing the thermal value of ν , ν_{th} by $\epsilon \nu_{th}$.

Equation A-11 was used to evaluate ϵ for the various clustered zones of the thermal-fast D_2O system. The procedure used was to homogenize the material inside the shrouds (Fig. A-5) and consider the resulting mixture as constituting a large "fuel rod." Also, P_1 and P_2 were taken to be equal and to have the value 0.28.⁽²²⁾ The cross sections varied slightly according to the region considered, but the resulting values of ϵ were sufficiently constant so that for all clustered zones ϵ may be taken to be 1.013. The value of ϵ deduced from a 2D calculation (which was just a homogeneous calculation) for the region 4 (Fig. A-2) was 1.007. The neutron inventory and breeding ratio given in Tables A-9 and A-13, pages 65 and 67, are based on the homogeneous value and hence the quoted breeding ratio slightly underestimates the correct value.

4. Xenon Poison Effects

In the portions of the system which have an appreciable thermal flux, it is necessary to examine the effects due to the xenon poisoning. The ratio of the average equilibrium xenon absorption cross section to fission cross section was calculated using the expression⁽²³⁾

$$\frac{\Sigma_{Xe}}{\Sigma_f} = \frac{y \tau_{Xe} \sigma_{Xe} \bar{\phi}}{1 + \tau_{Xe} \sigma_{Xe} \bar{\phi}} \quad , \quad (A-12)$$

where y is the fission yield of Xe^{135} , τ_{Xe} is the mean lifetime of Xe^{135} , σ_{Xe} the microscopic thermal absorption cross section of the Xe^{135} , and $\bar{\phi}$ is the volume average value of the thermal flux for the region under consideration. For the thermal-fast superheater concept, $\bar{\phi}$ was calculated assuming a power of 314 Mw in the boiling zone (Region 4 of Fig. A-2). For the fast-thermal concept, a power of 40 Mw was assumed for region 4 of Fig. A-3. In the first case, $\bar{\phi}$ was calculated to be 8.5×10^{13} neutrons/cm²/sec; for the fast-thermal system, $\bar{\phi}$ was found to be 2.0×10^{14} neutrons/cm²/sec. Fluxes in the other thermal regions were calculated as above using a suitable two-dimensional problem to determine the relative thermal fluxes in the various regions.

The influence of the xenon poisoning on the breeding ratio was found to be small. In particular, the xenon captures were generally less than 4% of the U^{238} captures in the same region. The change in breeding ratio expected would be only of the order of 0.02 for the cases examined.

D. Results

1. Flux and Power Distributions

The calculated radial flux distributions are shown in Figs. A-6 through A-9 for the thermal-fast and fast-thermal D₂O systems, the fast-thermal H₂O system, and the all-fast reference system. The corresponding power distributions are given in Figs. A-10 through A-13. The power flattening in the fast core typical of coupled as opposed to all-fast systems can be seen from the power distributions. The size of the power spike at the outer edge of the fast core for the coupled systems depends strongly upon the value of the thermal flux at that point. In practice, the thickness of the buffer zone would be varied to limit the size of the spike. (The discussion at the end of Section IV-E is pertinent in this regard.)

FIG. A6
RADIAL FLUX DISTRIBUTION FOR THERMAL-FAST
 D_2O COUPLED STEAM SUPERHEATER AT MIDPLANE

FIG. A7
RADIAL FLUX DISTRIBUTION FOR FAST-THERMAL
 D_2O COUPLED STEAM SUPERHEATER AT MIDPLANE

FIG. A8
RADIAL FLUX DISTRIBUTION FOR FAST-THERMAL
H₂O COUPLED STEAM SUPERHEATER AT MIDPLANE

FIG. A9
RADIAL FLUX DISTRIBUTION FOR
ALL-FAST SODIUM COOLED
REFERENCE SYSTEM AT MIDPLANE

FIG. A10
RADIAL POWER DISTRIBUTION FOR THERMAL-FAST
 D_2O COUPLED STEAM SUPERHEATER AT MIDPLANE

FIG. A11
RADIAL POWER DISTRIBUTION FOR FAST-THERMAL
 D_2O COUPLED STEAM SUPERHEATER AT MIDPLANE

FIG. A12
RADIAL POWER DISTRIBUTION FOR FAST-THERMAL
 H_2O COUPLED STEAM SUPERHEATER AT MIDPLANE

FIG. A13
RADIAL POWER DISTRIBUTION FOR
ALL-FAST SODIUM COOLED
REFERENCE SYSTEM AT MIDPLANE

2. Reactivity Effects

The magnitude of the prompt neutron lifetime and of the coefficients of reactivity for water addition in the fast core and thermal regions are of direct interest with regard to the kinetic behavior and safety of the coupled steam-superheater concept. In addition, knowledge of the fraction of reactivity contributed by the fast part of the system is important, since it determines how far the system is from being critical with respect to the fast neutrons alone. An indication of the reactivity due to only the fast part of the system can be obtained by calculating the k_{eff} of the system when no thermal fissions are permitted. (Actually, the calculated k_{eff} will be greater than the true value owing to distortion in the flux distribution due to the method of calculation.⁽²⁾ The correct value may, of course, be obtained from the coupling parameters⁽¹⁾ of the system.)

Table A-8 lists the prompt neutron lifetime, water reactivity coefficient for the fast core and thermal annulus, and k_{eff} for the case of no thermal fissions for the three superheater reference systems. (The prompt lifetime was calculated by uniform insertion of $1/v$ poison.) The very long lifetime for the thermal-fast system is due to the large power fraction in the thermal regions and to the fact that the moderator is D_2O . The water reactivity coefficients can be used to deduce temperature coefficients of reactivity for the various systems. For example, if one assumes a constant pressure of 95 atmospheres for a core temperature of 448.9°C , one obtains a fast core $(\delta k/k)/^\circ\text{C}$ due to water density changes of -1.8×10^{-5} , -2.2×10^{-5} , and -7.2×10^{-5} for the thermal-fast D_2O , fast-thermal D_2O , and fast-thermal H_2O systems, respectively.

Table A-8

REACTIVITY CHARACTERISTICS OF REFERENCE SYSTEMS

System	Prompt Neutron Lifetime, μsec	$\frac{\delta k}{k} / \left(\frac{\delta M}{M} \right)$ fast core water	$\frac{\delta k}{k} / \left(\frac{\delta M}{M} \right)$ thermal annulus water	$(k_{\text{eff}})_{\nu_{\text{thermal}}=0}$
Thermal-Fast D_2O	400 ^a	0.009	-	-
Fast-Thermal D_2O	4	0.011	0.032	0.978
Fast-Thermal H_2O	4	0.036 ^b	0.022	0.961

^aObtained from system similar to thermal-fast D_2O reference system but utilizing U^{235} rather than Pu^{239} in the fast core

^bObtained from system similar to fast thermal H_2O reference system but having the following composition in region 4 (Fig. A-3): natural uranium 45% zirconium 4.608% water 50.392%

3. Neutron Inventories and Breeding Ratios

Tables A-9 through A-12 list the neutron inventories for the three coupled systems and for the all-fast reference system. As indicated in the tables, all numbers are normalized with respect to one fission neutron in the fast core. It can be seen that the zirconium captures for the case of the thermal-fast D_2O system represent a serious parasitic loss to

the neutron inventory. For the fast-thermal systems, the zirconium is not an important parasitic agent, although for these cases the stainless steel separator is seen to have a more serious effect on the inventory than for the boiling reference system. In all cases, the neutrons leaking from the system could be reduced by increasing blanket thicknesses.

Table A-9

NEUTRON INVENTORY FOR THERMAL-FAST D₂O
COUPLED STEAM SUPERHEATER

Event ^a	Region					
	Core	Stainless Steel Tank	Inner Blanket	Water Annulus	Water Blanket	Remaining Blanket
Pu ²³⁹ Fissions	0.271	-	-	-	-	-
Pu ²³⁹ Captures	0.075	-	-	-	-	-
U ²³⁵ Fissions	-	-	0.100	0.965	0.028	0.011
U ²³⁵ Captures	-	-	0.036	0.222	0.006	0.003
U ²³⁸ Fissions	0.076	-	0.022	0.011	0.0004	0.005
U ²³⁸ Captures	0.269	-	0.240	1.090	0.095	0.095
D ₂ O Captures	0	-	0	0.008	0.0008	0
Zr Captures	-	-	0.017	0.134	0.009	0.004
Fe Captures	0.006	0.011	-	-	-	0.001

Neutrons Leaking = 0.060

^aAll numbers normalized to one fission neutron in the core.

Table A-10

NEUTRON INVENTORY FOR FAST-THERMAL D₂O
COUPLED STEAM SUPERHEATER

Event ^a	Region				
	Core	Core Blanket	Stainless Steel Tank	Water Annulus	Annulus Blanket
Pu ²³⁹ Fissions	0.277	-	-	-	-
Pu ²³⁹ Captures	0.073	-	-	-	-
U ²³⁵ Fissions	-	0.005	-	0.066	0.036
U ²³⁵ Captures	-	0.002	-	0.019	0.009
U ²³⁸ Fissions	0.074	0.013	-	0.002	0.001
U ²³⁸ Captures	0.238	0.132	-	0.126	0.169
D ₂ O Captures	0	0	-	0.0001	0.0003
Zr Captures	-	-	-	0.0004	0.0007
Fe Captures	0.005	0.006	0.031	-	-

Neutrons Leaking = 0.044

^aAll numbers normalized to one fission neutron in the core.

Table A-11

NEUTRON INVENTORY FOR FAST-THERMAL
H₂O REFERENCE SYSTEM

Event ^a	Region				
	Core	Core Blanket	Stainless Steel Tank	Water Annulus	Annulus Blanket
Pu ²³⁹ Fissions	0.269	-	-	-	-
Pu ²³⁹ Captures	0.077	-	-	-	-
U ²³⁵ Fissions	-	0.006	-	0.107	0.022
U ²³⁵ Captures	-	0.002	-	0.025	0.008
U ²³⁸ Fissions	0.076	0.016	-	0.008	0.019
U ²³⁸ Captures	0.234	0.123	-	0.137	0.231
H ₂ O Captures	0	0	-	0.032	0.0009
Zr Captures	-	-	-	0.0006	0.001
Fe Captures	0.005	0.007	0.038	-	-

Neutrons Leaking = 0.004

^aAll numbers normalized to one fission neutron in the core.

Table A-12

NEUTRON INVENTORY FOR ALL-FAST REFERENCE SYSTEM

Event ^a	Region		Event ^a	Region	
	Core	Blanket		Core	Blanket
Pu ²³⁹ Fissions	0.273	-	U ²³⁸ Fissions	0.077	0.018
Pu ²³⁹ Captures	0.070	-	U ²³⁸ Captures	0.281	0.305
U ²³⁵ Fissions	-	0.006	Na Captures	0.0004	0.0007
U ²³⁵ Captures	-	0.002	Fe Captures	0.006	0.007

Neutrons Leaking = 0.030

^aAll numbers normalized to one fission neutron in the core.

Table A-13 compares the various reference systems. The fast-thermal D₂O and H₂O systems are seen to have similar breeding ratios; the greater parasitic loss of the H₂O as compared with the D₂O is compensated for by the smaller leakage in the H₂O system.

Table A-13

COMPARISON OF SUPERHEATERS AND THE FAST REFERENCE SYSTEM

System	Core Volume, liters	Critical Mass, kg	Fraction of Power in Thermal Fissions	Neutrons Leaking ^a	Initial Breeding Ratio ^b
Thermal-Fast D ₂ O Superheater	402	285	0.669	0.060	1.07
Fast-Thermal D ₂ O Superheater	326	275	0.200	0.044	1.53
Fast-Thermal H ₂ O Superheater	326	251	0.230	0.004	1.52
All-Fast Reference	402	293	0	0.030	1.71

^aNormalized to one fission neutron in the core.

^bNot counting U²³⁵ burned in depleted uranium.

References

1. R. Avery, Coupled Fast-Thermal Breeder, Nucl. Sci. Eng. 3, 129 (1958).
2. R. Avery et al., Coupled Fast-Thermal Power Reactor Critical Experiment, Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland (1958), Vol. 12, p. 151.
3. M. K. Butler and J. M. Cook, UNIVAC Programs for the Solution of One-dimensional Multigroup Reactor Equations, ANL-5437.
4. G. G. Bilodeau et al., PDQ - An IBM-704 Code to Solve the Two-dimensional Few-group Neutron-diffusion Equations, WAPD-TM-70 (1957).
5. Reactor Physics Constants, ANL-5800, (1958) p. 83.
6. P. Griffith et al., Void Volumes in Subcooled Boiling Systems, ASME Paper No. 58-HT-19 (1958).
7. McAdams et al., Heat Transfer to Superheated Steam at High Pressures, ASME Paper No. 49-A-32 (1949).

8. L. Fano et al., Compressibility Factor, Density, Specific Heat, Enthalpy, Entropy, Free Energy Function, Viscosity, and Thermal Conductivity of Steam, NBS, NACA Technical Note 3273 (1956).
9. R. G. Deissler and M. F. Taylor, Analysis of Axial Turbulent Flow and Heat Transfer Through Banks of Rods or Tubes, NACA, Reactor Heat Transfer Conference, TID-7529, Pt. 1, Book 2.
10. K. Berg et al., Thermal Conductivity of UO_2 , Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland (1958), Vol. 7, p. 691.
11. The Reactor Handbook, Vol. 2, Section 1: "Light and Heavy Water-cooled System," AECD-3646 [Also McGraw-Hill Book Company, Inc. New York (1955)].
12. E. Hellstrand, Measurements of the Effective Resonance Integral in Uranium Metal and Oxide in Different Geometries, J. Appl. Phys. 28, 1493 (1957).
13. S. Glasstone and M. C. Edlund, The Elements of Nuclear Reactor Theory, D. Van Nostrand, Inc., Princeton, New Jersey (1952).
14. M. Dancoff and M. Ginsburg, Surface Resonance Absorption in a Close-packed Lattice, CP-2157 (1944).
15. I. Carlvik and B. Pershagen, The Dancoff Correction in Various Geometries, AE-16 (1959).
16. J. Thie, A Simple Analytical Formulation of the Dancoff Correction, Nucl. Sci. Eng., 5, 75 (1959).
17. J. Wheeler, Unpublished code.
18. I. Carlvik and B. Pershagen, AEFI-58, A.B. Atomenergi, Sweden (1957) (in Swedish).
19. Reactor Physics Constants, ANL-5800, (1958) p. 132.
20. A. M. Weinberg and E. P. Wigner, The Physical Theory of Neutron Chain Reactors, The University of Chicago Press, Chicago, Illinois (1958), p. 706.
21. Ibid, p. 705.
22. Ibid, p. 712.
23. Ibid, p. 600.