ANL-7212
Reactor Technology
(TID-4500)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

GESTI PRICES

H.C. \$3.00; MN 65

AN EXPERIMENTAL BOILING-LIQUID-METAL FAST REACTOR

by

R. H. Armstrong

Reactor Engineering Division

and

J. C. Carter

Reactor Physics Division

RELEASED FOR ANNOUNCEMENT

IN NUCLEAR SCIENCE ABSTRACTS

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.

June 1966

Operated by The University of Chicago
under
Contract W-31-109-eng-38
with the
U. S. Atomic Energy Commission

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.

TABLE OF CONTENTS

	<u>Page</u>
NOMENCLATURE	8
ABSTRACT	9
I. INTRODUCTION	9
II. DESCRIPTION OF TEST UNIT	11
III. TECHNICAL ASPECTS	20
A. General	20
B. Nuclear	23
C. Heat Transfer	27
D. Dynamics	33
E. Choice of Liquid Metal	35
F. Mechanical Design	39
G. Materials of Construction	46
H. Shielding	49
I. Fuel Material	52
IV. SAFETY	55
V. PROGRAM OF ANALYTICAL WORK	56
A. Nuclear Analysis	56
B. Thermodynamic Analysis	59
C. Mechanical Analysis	64
D. Dynamic Analysis	66
VI. SUPPORTING RESEARCH AND DEVELOPMENT	68
A. General Discussion	68
B. Critical Experiments	69
C. Heat Transfer and Thermodynamics	69
D. Dynamics	70
VII. TEST PROGRAM	72
A. Mechanical Tests	72
B. Thermodynamic Tests	74

TABLE OF CONTENTS

	<u>Page</u>
C. Nuclear Tests	75
D. Metallurgical Tests	78
E. Safety Tests	79
VIII. COST ESTIMATE	82
A. Cost of Fabricated Components	82
B. Cost of Assembling Test Unit (Labor)	82
C. Cost of Research and Development	82
ACKNOWLEDGMENT	82
REFERENCES	83

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.	Longitudinal Cross Section of Test Unit	12
2.	Na and NaK Systems for EBR-I	13
3.	Core Cross Section	14
4.	Typical Fuel Element and Assembly	16
5.	Comparison of Core Cross Sections.	18
6.	Schematic of Flow Diagram Showing Instrumentation Sensing Points	22
7.	Enlarged View of Core Region	24
8.	Reactivity vs Reflector Effectiveness.	25
9.	Change in Reactivity as a Function of Change in Volume of Core.	26
10.	Change in Reactivity as a Function of Change in Density of Liquid Metal within Core.	26
11.	Radial Power Distribution as a Function of Cylinder Radius. . .	27
12.	Axial Power Distribution as a Function of % Axial Length	28
13.	Block Diagram of Control System	34
14.	Vapor Pressure vs Temperature	37
15.	Density vs Saturation Pressure.	37
16.	External Safety-rod Drive	39
17.	Safety-rod Acceleration and Velocity vs Travel	40
18.	Safety-rod Travel vs Time	40
19.	Safety Plug and Drive	41
20.	Safety-plug Acceleration vs Travel	42
21.	Safety-plug Velocity vs Travel	42
22.	Safety-plug Travel vs Time	42
23.	Elevator Mechanism	42
24.	Beryllium Oxide Reflector Drives	43
25.	Neutron- and Gamma-flux Distribution in Shielding	50
26.	Diagram Illustrating Subdivision of Core	61
27.	Diagram Illustrating Subdivision of a Fuel Rod	61

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
28.	Schematic Diagram of Flow of Coolant through Reactor	63
29.	Zone Arrangement of Core.	75

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
I.	Characteristics of the Core	21
II.	Dimensions and Composition of a Hypothetical Core	25
III.	Heat Transfer and Pressure Loss Associated with Liquid-metal Heater	31
IV.	Heat Transfer and Pressure Loss Associated with Liquid-metal Boiler	32
V.	Physical Properties of Liquid Metals.	38
VI.	Physical Properties of Structural Materials	47
VII.	Chemical Composition of Reactor-grade Columbium.	48
VIII.	Chemical Composition of Reactor-grade Tantalum	48
IX.	Gamma-ray Constants.	51
X.	Constants for Capture Gamma-ray Calculations.	52

NOMENCLATURE

A	Area for liquid metal flow through core	ft ²
D	Hydraulic diameter	ft
C _p	Specific heat	Btu/lb-°F
G	Mass flow	lb/ft ² -sec
f	Friction factor	
g	Gravity constant	ft/sec ²
h	Heat-transfer rate	Btu/ft ² -°F-sec
h _{fg}	Latent heat of vaporization	Btu/lb
k	Reactivity	
K _{ex}	Excess reactivity	
K	Thermal conductivity	Btu/hr-ft-°F
l	Neutron lifetime	
L	Length	
n	Neutron	
Nu	Nusselt number	$\frac{hD}{k}$
P	Pressure	lb/in. ²
Pe	Peclet number	
Pr	Prandtl number	
Q	Total heat	Btu/sec
q	Heat flux	Btu/sec-ft ²
r	Radius of fuel rod	in.
R	Radius of reactor core	in.
R	<u>Outside diameter of rod</u> Pitch of rod	
Re	Reynolds number	
S	Surface	ft ²
T	Temperature	°C or °F
V	Volume	ft ³
v	Velocity	ft/sec
W	Flow rate	lb/sec
Z	Length of reactor core	in.
β	Delayed-neutron fraction	
λ	Decay constant	
μ	Viscosity	lb/ft-hr
ρ	Density	lb/ft ³
ψ	<u>Eddy diffusivity of heat</u> <u>Eddy diffusivity of momentum</u>	

Subscripts

f	Liquid
g	Vapor
i	Delayed group of neutrons
o	Origin
s	Surface

AN EXPERIMENTAL BOILING-LIQUID-METAL FAST REACTOR

by

R. H. Armstrong and J. C. Carter

ABSTRACT

This is a proposal to use the existing facilities of the EBR-I installation for the purpose of investigating the characteristics of fast solid-fuel reactors in which liquid metal is boiling under controlled and observable conditions.

An integral unit is designed to fit into the existing EBR-I core cavity and to connect to the existing auxiliary facilities; the technical aspects are discussed, and a test program is presented.

I. INTRODUCTION

It is proposed to use the existing EBR-I facilities to study the operating characteristics of boiling liquid metals in fast-reactor cores and to do this by integrating a sealed test unit containing a small core with these facilities.

One objective is to determine the effect of vapor formation on the nuclear characteristics. Vaporization of the coolant produces changes in reactivity. The reactivity feedbacks due to volumetric expansion as a function of fuel temperature, and also due to variation in liquid-metal density as a function of percent vapor, can be measured.

Another objective is to investigate the feasibility of using the vapor formed within the core as the working fluid in a power cycle. Liquid metal can be boiled in a reactor core, but conditions are unknown. The relationship between dynamic stability, vapor pressure, vapor quality, thermodynamic properties, and core geometry is not established. It does appear that an increase in pressure of vaporized liquid metal is compatible with an increase in vapor quality and the degree of dynamic stability.

The only necessary modifications to the existing EBR-I facilities will be those entailed in connecting the lines from the test unit to the existing facilities. The existing facilities are control system, instruments, shielding, primary and secondary NaK heat-transfer systems, reactor oscillator equipment, electrical equipment, heat, water, gas, hoist, and storage facilities of the EBR-I building.

In addition to the facilities presently existing for the reactor and building, the following equipment may be required:

1. Helium-turbine-driven centrifugal pump,
2. Liquid-metal cleanup trap,
3. NaK-cooled metal-vapor condenser,
4. Helium compressor,
5. Helium-to-air heat exchanger,
6. Vacuum-system instrumentation,
7. BeO reflector controls and operating mechanisms,
8. Oscillator drive, and
9. Source drive.

The magnitude and range of this additional equipment are those associated with a closed-loop thermodynamic cycle in which 1 MW of heat in the form of sodium vapor is condensed and recycled. The pressure may be any value up to 150 lb-in.² absolute.

The phenomena expected to exist in the test unit must be analyzed thoroughly before fabrication. A preliminary analysis is part of this proposal. This preliminary analysis indicates what should be measured and how much flexibility must be designed into the constraints of the physical system in order to include the range of variables in the areas of uncertainty. Frequently, valuable information is never obtained because the full information objective was not known when the test was conceived and the equipment fabricated. In regard to flexibility, provision should be made for handling higher temperatures and pressures than anticipated; the liquid-metal pump should have variable capacity, and the condenser should be oversize.

Good analytical work can be done by writing the equations that define and relate the space and time-dependent variables, and then simulating these equations on a hybrid computer.

This is an easy way to see how each variable affects the system response to a perturbation. An analog computer can show the range of a variable at any place in the system at any time and is of value in determining the amount of flexibility and instrumentation to design into the test unit.

During the tests the results will be continuously analyzed and used to improve the equations of the mathematical model and the test technique and equipment.

Fig. 1. Longitudinal Cross Section of Test Unit

II. DESCRIPTION OF TEST UNIT

The estimated thermal rating of the proposed test core is 1.0 MW with sodium as the working fluid. It is a simple version of a conventional solid-fuel fast-reactor core with reflector control. The intent is to design the apparatus so that (1) reactivity fluctuations due to the boiling phenomenon cannot exceed β , and (2) the reactivity feedback due to this phenomenon is relatively small and dominated by the strong negative feedback due to the volumetric expansion of the core.

The test unit (Fig. 1) is a double-wall shell containing a fast-neutron core, liquid metal, and a condenser. The liquid metal is boiled and condensed within this shell. All the existing EBR-I facilities required for operating a fast-neutron core can either be directly connected or connected through heat exchangers to this test unit (see Fig. 2).

The test unit can operate with any liquid metal compatible with columbium, over a range of pressure and temperature with top limits of 10 atm and 1200°C. No significant modifications to the existing EBR-I facilities are necessary to accomplish this.

The physical features of the test unit are illustrated in two drawings. Figure 1 shows longitudinal cross sections of the test unit. Figure 3 shows a horizontal cross section through the reactor core.

The intent is to replace the present EBR-I core with the test unit. The double-wall shell is similar to a large vacuum bottle with an inner shell of columbium and an outer shell of stainless steel. The space between the two shells provides a means of detecting leaks in the inner shell, a means of protecting the columbium from oxidation at high temperatures, and a means of creating a large temperature gradient between the columbium and the stainless-steel shells. The reason for this temperature gradient is that heat can be transferred between the shells across a vacuum only by radiation, which varies as the fourth power of the absolute temperature. There are two closure plates with space between them which, in addition to the above reasons, eliminate temperature gradients in the closure joints.

The annular space between the stainless-steel shell of the test unit and the existing EBR-I stainless-steel core cavity liner is full of NaK and is in series with the existing primary heat-transfer system. Calculations indicate that the amount of heat to be carried away at shutdown is 150 kW maximum. This heat is radiated from the inner shell to the outer shell and removed by conduction and convection to the NaK. Thus, the existing primary heat-transfer system provides an ever-present and adequate way to remove shutdown heat, to cool the reflector control which is in the annular space, and to keep the liquid metal in the test unit above the melting point during startup.

Fig. 2. Na and NaK Systems for EBR-I

Fig. 3. Core Cross Section

The core is controlled by the existing EBR-I controls, augmented by the movable cylindrical reflectors surrounding the core. There are now no calculations to indicate the amount of interaction between the existing controls and the movable reflector.

The beryllium oxide movable cylindrical reflectors are assembled to a framework affixed to the containment shell and are actuated by means of lead screws, which can be actuated independently or can be integrated with the existing EBR-I control system. Either the reflector controls or the existing control system alone is capable of controlling the core. The reflector control is included because its integration with the characteristics of the core is indispensable to the preservation of dynamic similitude between test unit and an actual power plant.

The core is a bundle of niobium-clad uranium carbide rods in the form of a right cylinder. Specifically, the core shown in Fig. 3 consists of six segmental assemblies, each containing 146 fuel elements, or a total of 876 elements. The liquid metal is forced to flow longitudinally by a multiphase pump, through the lateral spacing of the rods. The two cylindrical shells of the concentric beryllium oxide reflectors move longitudinally relative to the transverse centerline of the core. The reactivity worth of the reflector shells is such that their removal from the core region will scram the reactor.

Two longitudinal regions may exist in the core: the liquid heating and the boiling. The lengths in which each of the two phases of fluid may exist depend on the desired operating conditions and on which liquid metal is in the core. The control system varies the pump speed and the reflector movement until the desired condition of the liquid metal leaving the core is obtained. This may be as a hot liquid, or as a mixture of saturated liquid and vapor.

The flow of boiling liquid metal is upward through the core. The metal vapor leaves the free surface above the core (see Fig. 1) and rises into the bundle of condenser tubes. The condensate falls by gravity back into the pool of liquid metal.

Orificing is not currently provided in the core, but some type of orificing may need to be added to get correct flow distribution. This will be determined by experimental work before the core is built.

The test unit is arranged so that the core loading can be adjusted after the core is in place.

The fuel element tentatively selected is shown in Fig. 4. The fuel is uranium carbide. There is metal-to-metal contact between the uranium carbide and the columbium cladding. The three radial spacers are continuous over the length of the fuel and reflector sections of the fuel elements.

Fig. 4. Typical Fuel Element and Assembly

The fuel element is a simple solid rod, which is satisfactory in its nuclear, thermal, and fabrication aspects. Its physical appearance is similar to that of the EBR-I, Mark III, fuel rods, which are considered to be satisfactory. It may be that increased power density can be attained with more intricate designs. With this hope in mind, some ideas for fuel elements are presented in Fig. 5. Fuel element type number 9 is the one shown in detail in Fig. 4. All the cladding and structural material of the core in contact with the liquid metal is columbium.

A helium-gas turbine-driven pump is shown in Fig. 1. A similar pump has demonstrated its capability for pumping boiling water when the available head is zero.¹ Its performance with liquid metal near the saturation point has not been investigated. This pump appears to be a potentially good one for this concept. Circulation of helium is from the compressor and gas heat exchanger to the inlet of the turbine that drives the pump impeller.

The condenser consists of a series of helical tubes through which NaK flows and on which the metal vapor is condensed. The outer tube of the condenser is constructed of columbium, 1% zirconium; the inner tube is stainless steel; the annulus between them is filled with helium.

The compressor in the helium circuit is an axial-flow type with a pressure ratio of 1.07. The pressure and temperature of the helium entering the compressor are 4.7 atm and 400°C.

The helium-to-air heat exchanger and the compressor are constructed of stainless steel.

The small holdup volume of liquid metal in this proposed test unit may result in more activation of the secondary system NaK than is the case in the present EBR-I operations. The use of NaK as the coolant in the condenser will provide a means of handling the large temperature difference between the boiling liquid metal in the test unit and the temperature permissible in the NaK of the existing EBR-I heat-rejection system.

The entire reactor structure is expected to be at a uniform temperature equal to that of the boiling liquid metal (2000°F).

The distinctive features of this concept are:

1. The flow through the core is assured by a liquid-metal pump. The flow is independent of the strength of the gravity field or the orientation of the core.

2. The diameter of the core is such that the reactivity worth of the reflector is large and sufficient to scram the reactor.

UC = 12.6 gm/cc
BeO = 3 gm/cc
Cb = 8.6 gm/cc
W = 19 gm/cc
Na = .71 gm/cc

ALL DIMENSIONS IN METRIC UNITS

FUEL ELEMENT TYPE	PITCH cm	CORE DIA cm	DATA FOR ONE FUEL ELEMENT							Cb AREA cm ²		DATA FOR CORE										MWI. H.F. 31.54 cm ²	REMARKS	
			AREA - cm ²					EQUIV. DIA. cm	UC VOL. cm ²	FUEL CANS	CENT. TUBE	INNER AREA cm ²	VOL. cm ²	% RATIO			NO FUEL ELE.	HT. TR. AREA cm ²	FLOW AREA cm ²	VOL. UC cm ³	WEIGHT UC KG.			
			Cb	UC	METAL	FLOW TUBE	HEAT TRANS.							Cb	UC	Na								
#1 LUNAR . 208 H x .127 W . 833 OD x .773 ID	1.041	24 40	15620 40	.46828 40	.62448 40	.31454 40	.90903 40	62.825 4052	11.2401 18.7334	23.95 42.59	.5399	427.50 1299.9	10263 52013	19.32 17.95	40.75 43.88	39.93 38.17	372 1218	23371.6 127537	170.66Δ 383.11X	4182.10 22821.7	52.695 287.55	0.737 4.022	M/V RATIO SAME AS LUNAR-.62	
#2 COMPOUND "A" . 117 H x .089 W 1.026 OD x .965 ID . 812 OD x .752 ID . 050T . 442 OD x .396 ID	1.143	24 40	25276 40	.33612 40	.58889 40	.54250 40	1.1314 40	167.38 278.96	IN 2444 OUT 3222	8.0681 13.4476	23.08 43.69	.5399	425.80 1317.6	10215 52720	24.44 23.39	25.12 26.63	50.44 49.98	318 1044	53226.9 291250	214.60Δ 172.59X	2566.12 14041	32.332 176.92	1.679 9.185	M/V RATIO = .52
#3 ROD "A" . 358 H x .096 W . 785 OD x .653 ID	1.143	24 40	25283 40	.33467 40	.58751 40	.54388 40	1.1314 40	59.186 98.638	8.0332 13.3886	23.56 42.60	.5399	436.5 1306.8	10480 52289	23.25 19.50	23.46 25.82	53.29 54.68	306 1008	18111.4 99424.7	166.42X 548.23X	2458.62 13498.3	30.978 170.08	0.5712 3.157		
#4 TUBULAR "A" . 127 H x .096 W 1.016 OD x .930 ID	1.143	24 40	25247 40	.33623 40	.58870 40	.54270 40	1.1314 40	119.90 199.83	IN .574 OUT .310	8.0703 13.4509	23.03 43.52	.5399	423.3 1310.0	10162 52416	24.54 20.12	25.26 26.80	50.20 53.08	314 1044	38129.2 208632	172.57X 566.57X	2566.9 14045.3	32.343 176.97	1.203 6.580	
#5 COMPOUND "B" . 046 H x .147 W . 779 OD x .703 ID 1.097 OD x 1.021 ID . 066T . 444 OD x .394 ID	1.143	24 40	27909 40	.55180 40	.83089 40	.30049 40	1.1314 40	163.74 236	IN 155 OUT 236	13.2449 23.14	.5399	410.1 1306.8	9846 52289	27.97 19.50	39.56 25.82	32.47 54.68	294 1008	48090.3 99424.7	133.15Δ 88.35X	3894.7 13498.3	49.073 170.08	1.517 3.157	M/V RATIO FOR EBR I MOD. = .74	
#6 ROD "B" . 056 H x .157 W . 986 OD x .828 ID	1.041	24 40	24905 40	.54016 40	.78922 40	.14999 40	.93926 40	74.315 241	12.9654 23.95	23.95 42.59	.5399	427.5 1306.8	10263 52289	27.40 19.50	47.00 25.82	25.60 54.68	372 1008	27645.2 99424.7	109.40Δ 548.23X	4824.0 13498.3	60.782 170.08	.872 3.157		
#7 TUBULAR "B" . 038 H x .157 W 1.003 OD x .901 ID	1.041	24 40	19445 40	.53496 40	.72942 40	.20979 40	.93926 40	99.560 317	IN 317 OUT 345	12.8406 13.4509	23.95 43.52	.5399	427.5 1310.0	10263 52416	22.65 20.12	46.55 26.80	30.80 53.08	372 1044	37050.6 208632	131.65Δ 566.57X	4777.6 14045.3	70.197 176.97		1.169 6.580
#8 EBR II PIN . 441 OD x .365 ID . 124 DIA.	.5664	24 40	.06050 40	.10507 40	.16557 40	.10981 40	.27767 40	33.161 0.0405	2.5219 24.04	24.04 42.59	.5399	431.4 1306.8	10358 52289	25.01 19.50	33.89 25.82	41.10 54.68	1392 1008	46163.5 99424.7	177.28Δ 548.23X	3511.2 13498.3	44.241 170.08	1.435 3.157	EBR II PIN M/V = .59	
#9 EBR I (M-5) . 119 H x .081 W . 713 OD x .653 ID	.833	28 40	.09535 40	.33467 40	.43002 40	.17188 40	.6029 40	62.72 256	9.352 23.95	23.95 42.59	.5399	526.56 1474.4	1474.4 52013	20.32 17.95	51.67 43.88	28.01 38.17	876 1218	54942. 127537	150.6 383.11X	8192 22821.7	66.822 287.55	1.733 4.022	EBR I-M-5 REF. DESIGN	

Δ INCLUDES CENTER TUBE & AREA OUTSIDE CAN X INSIDE CAN

Fig. 5. Comparison of Core Cross Sections

3. The cross sections for nuclear reactions in the cycle fluid are so small that fluctuations in the density of the fuel from 100% liquid to 100% vapor do not create uncontrollable fluctuations in reactivity.

4. The forced-flow pump is the first variable to respond to a change in power, and the reflectors follow to hold the vapor leaving the reactor at the desired amount and pressure.

5. The reactor can operate with any of the following liquid metals as the boiling fluid: mercury, cesium, rubidium, potassium, sodium, or lithium. The condition of pressure and temperature will be that concomitant with the metal used, but the power can vary with demand.

III. TECHNICAL ASPECTS

A. General

The design of this type of reactor is without precedent. Many variables of the interacting nuclear and thermal phenomena are pressure- and temperature-dependent, and their relationships as functions of space and time are not understood to a satisfactory extent. Concurrent analytical and experimental work is believed essential to success. Such a situation dictates that the core and auxiliaries have as much flexibility and instrumentation designed into them as possible without jeopardizing the objectives of demonstrating the dynamic characteristics of the proposed core within the limits of the structural integrity of the supporting and containing structure.

The largest areas of uncertainty in the estimated performance lie in material technology, in heat-transfer phenomena, and in the dynamics of the system represented by the core and the boiling sodium. Therefore, the performance estimates are based on conservative appraisals of physical properties and phenomena in these areas.

The materials are of primary concern because they will determine the temperature at which liquid metal can be boiled. In light of recent research and development it is hoped that columbium can be fabricated into the desired shapes shown in Fig. 1 and that it will retain its structural integrity at 2000°F. If a 2000°F operating temperature cannot be reached, the tests will be cut back to a temperature at which structural integrity can be maintained.

In the area of heat transfer, the phenomena of two-phase flow, heat-transfer rates during boiling and condensing of the NaK, together with the burn-out fluxes of a liquid metal at the proposed conditions of this test, have not been investigated to the point where it is considered advisable to use any but the most conservative values. These conservative values are used throughout the design. Therefore, the heat transfer should be equivalent to the estimated performance.

As a preliminary experiment, an electronic analog of the reactor system will be built to coordinate the analog into the design, with the hope that the desired reactor heat-transfer function can be designed into the physical system.

Another area of the design that involves a small range of uncertainty is the fluid-flow system. This system constitutes a turbine-driven recirculating pump, whose impeller is immersed in the sodium, which involves fluid dynamics, thermal and radiation shielding, containment, and structural design. A high order of skill is required because

of the combined effect of pressure and temperature stresses, expansion, creep, and fatigue.

The area of controls is very important and will require considerable research and development work on neutron-flux instruments, together with pressure- and temperature-sensing devices (see Fig. 6). The test unit is designed so that the sensing elements of the instruments can be replaced.

The extent of revision to the existing EBR-I is very small, but some structural work will have to be altered to accommodate the reflector control mechanisms.

The nuclear calculations are conventional except for the determination of the effect upon reactivity caused by the possible interaction between the control systems, the existing cup, and the movable reflector. This effect can easily be determined when an analog is constructed. The motion and worth of the existing cup and of the reflectors may be arranged and controlled as desired. The required rate of motion is not known at this time. Hot and cold critical experiments will be conducted. Some performance calculations on the steady-state nuclear, thermal, and mechanical phenomena were made for the in-core boiling condition only. The results of these calculations are listed in Table I.

TABLE I. Characteristics of the Core

Mode of Operation	Boiler	Boiler or Heater	Heater
Thermal capacity of the test core, kW	1000		5000
Power density, MW/liter (core)	0.06		0.30
Length of the test unit, in.		104	
Diameter of the test unit (max), in.		22.8	
Dimensions of the core (cyl), in.		11 x 11	
Core composition by volume percent	$\left\{ \begin{array}{l} \text{U}^{235} \\ \text{U}^{238} \\ \text{C} \\ \text{Cb, 1\% Zr} \\ \text{Na} \end{array} \right.$	22.19	
		11.09	
		18.39	
		20.32	
		28.01	
Neutron flux (avg)		2.0×10^{14}	
Sodium density coefficient, $(\Delta k/k)/(\Delta \rho/\rho)$		0.01	
Core expansion coefficient, $(\Delta k/k)/(\Delta V/V)$		0.323	
Sodium saturation temperature, °F		2000	
Fuel temperature (max), °F		1790	
Sodium velocity inlet to core, ft/sec		7.2	
Sodium pressure at exit from core, psia		75	
Core heat transfer surface, ft ²		59	
Heat flux (avg), Btu/hr-ft ²	57,840		289,200
Average exit vapor quality by weight percent	1.6		
Average exit vapor volume fraction, %	87.0		
Sodium recirculating pump pressure ratio		1.36	
Sodium flow rate through core, lb/sec		38	
Shutdown coolant requirement (max), kW		150	
Helium flow rate, lb/sec	1.0		5.0
Helium pressure (max), psia		77	
Helium-compressor pressure ratio		1.1	
Helium temperature entering the compressor, °F		600	
Heat-transfer area of condenser, ft ²		78

Fig. 6. Schematic of Flow Diagram Showing Instrumentation Sensing Points

B. Nuclear

Nuclear phenomena were calculated for steady-state conditions by means of Code RE-122 and the 16-group cross sections of Hansen and Roach.² These calculations were made for the test core with sodium as the coolant. Some characteristics of the core are given in Table I.

This proposed experiment involves a small, reflector-controlled fast reactor with enough reactivity in the reflector (as a function of position and time) to exceed the combined excess reactivity resulting from the undesirable operating conditions and accidents.

The two halves of the transversely split cylindrical reflector shell (see Fig. 7) move axially in opposite directions with respect to the traverse centerline of the core, reducing the number of neutrons reflected into core. Thus the movement of the reflector controls the reactivity, and the removal of the reflector scrams the reactor. When the core is completely encompassed by the reflector, 100% reflector effectiveness is considered to exist.

When the reactor is functioning as a liquid-metal heater, the two halves of the reflector are actuated by the same signal and move exactly the same amount, at the same rate but in opposite directions. When the reactor is functioning as a boiler, the two halves of the reflector operate independently. The upper half of the reactor is actuated by the temperature of the uranium carbide fuel. The lower half is actuated by the vapor pressure of the boiling liquid.

A hypothetical core with reflector is chosen to study the effects of different liquid-metal coolants and different operating conditions. It is an exploratory, order-of-magnitude study, and the numbers do not pertain to the specific test-core design. The composition of this core and its dimensions are given in Table II. These numbers are associated with criticality existing for the conditions of 100% reflector effectiveness and 50 w/o lithium vapor in an infinite cylindrical core. The reactor is supercritical when it is full of liquid metal in the 100% liquid phase and the reflector is in the 100% effective position. The reactivity as a function of reflector effectiveness with the percent of lithium vapor by weight in the core as parameter is shown in Fig. 8. It is subcritical when more than 50% vapor is present in the core. Thus if it is intended to operate as a liquid-metal heater the reflector effectiveness must be limited to something less than 100%. How much less depends upon which liquid metal is in the core and what percentage of it, if any, is vaporized. If the core were designed to be minimal critical when full of liquid metal in the 100% liquid phase, it would not operate as a boiler, for as soon as vapor formed, the reactor would go subcritical due to both core expansion and void formation. The calculation is independent of the dynamic characteristics of the system. The reactivity worth as a function of reflector effectiveness, percent vapor in core, and time will be determined by analog.

Fig. 7. Enlarged View of Core Region

TABLE II. Dimensions and Composition of a Hypothetical Core

Core dimensions (cyl), cm: 24 x 24					
Core volume, liters: 11					
Core Composition					
Material	v/o	Atom/cc of Material	Atom/cc of Core	Volume of Material, liters	Weight of Material, kg
U^{235}	26.28	0.0473×10^{24}	0.01243×10^{24}	2.89	54.35
U^{238}	13.13	0.048×10^{24}	0.00630×10^{24}	1.44	27.15
C	21.79	0.0838×10^{24}	0.01826×10^{24}	2.40	6.72
Nb-1% Zr	23.81	0.0545×10^{24}	0.01298×10^{24}	2.62	22.0
Liquid metal (Li)	15.50	0.0463×10^{24}	0.00718×10^{24}	1.71	0.9
Reflector					
Material	Atom of O per cc of Reflector	Atom of Be per cc of Reflector	Volume of Material, liters	Weight of Material, kg	
BeO	0.0728	0.0728	24.9	71

Fig. 8. Reactivity vs Reflector Effectiveness

The change in reactivity, $\Delta k/k$, as a function of the change in volume of the core, $\Delta V/V$, is shown in Fig. 9. This $\Delta V/V$ due to thermal expansion of the core provides a prompt negative feedback of reactivity. After boiling starts, the heat-transfer rate from fuel to fluid decreases with an increase in the amount of vapor formed. Consequently the temperature of the fuel and the core structure increases with an increase in vapor formation. The temperature of the boiling liquid metal remains constant, but the feedback of negative reactivity due to voids increases. As a result, there is a prompt negative and a delayed negative feedback, which increases with an increase in the amount of vapor.

The change in reactivity, $\Delta k/k$, as a function of the change in density of the liquid metal within the core, $\Delta \rho/\rho$ (see Fig. 10), is determined by varying the density of liquid metal within a critical core. The time lag of this feedback will also have to be determined by analog.

The difference in the calculated values of reactivity between the core with 100% reflector effectiveness, the core full of liquid, and the core full of vapor is 0.007. Therefore it is theoretically unlikely that the fluctuations associated with boiling can cause any trouble. The average density of the boiling liquid metal can oscillate due to flow instability resulting from

(1) the manner in which the operating conditions are varied, (2) operating at a resonant condition, and (3) operating at an excessive power density. The dynamic phenomena of the first two causes can be analyzed and controlled, and that of the third analyzed and avoided.

Fig. 9. Change in Reactivity as a Function of Change in Volume of Core

Fig. 10

Change in Reactivity as a Function of Change in Density of Liquid Metal within Core

All the liquid metal proposed have different cross sections. Calculations were made by holding everything constant and filling the core with each liquid metal. The relative reactivity worth of each liquid metal thus determined is as follows:

Lithium	1.0000
Sodium	0.996766
Potassium	0.991267
Mercury	0.988078

The relative worth in reactivity of each metal pertains only to this core and does not necessarily indicate what this relative worth may be in cores of different dimensions and composition. The estimates are preliminary and are considered to indicate orders of magnitude and relative relationships. An additional calculation in a spherical geometry using the DSN code was made as a check. The 16-group cross-section set of

Hansen and Roach was used² with the assumed conditions of liquid sodium and 100% reflector control. Only the first 10 groups were used, including energy down to 30 eV.

The fuel was assumed to be UC, with the atom ratio:

$$\frac{0.032 \times 10^{24} \text{ C/cc}}{0.029 \times 10^{24} \text{ U/cc}}$$

The reactor had $k = 1$ at the following volume fractions: fuel, 0.5167; clad (Mo), 0.2036; sodium 0.2801. The sodium was then replaced with void, and the k found to be 0.9907. A $\Delta k/k$ of -27.602% resulted when the BeO reflector was removed. A more detailed calculation of the reflector worth as a function of distance moved was not attempted at this time.

Since the total worth of full-density sodium in the available void space is only about 1%, it seems unlikely that boiling fluctuations can cause the reactivity to exceed prompt critical. The density of the boiling sodium in the core can change from two causes: a change in temperature due to a change in pressure or inlet temperature, and a fluctuation in void fraction due to flow instability. The latter is considered to be the major cause.

To obtain the change in reactivity due to volumetric expansion of the core from any cause, the radius was increased in increments of $1/2$ up to 2%. The core mass and the thickness of all other regions were held constant. The most likely cause of change in a core volume is expansion or contraction of the core structural materials, and this can be controlled by design. The coefficient of thermal expansion used was $6.8 \times 10^{-6} \text{ cm}/(\text{cm})(^\circ\text{C})$.

The beryllium oxide reflector creates a small power spike at the outer edge of the core, but is too thin to permit the production of much low-energy flux. The power distribution in the spherical model (shown in Fig. 11) is a qualitative indication of how the radial power distribution in the actual core will be affected by the beryllium oxide reflector.

Fig. 11. Radial Power Distribution as a Function of Cylinder Radius

C. Heat Transfer

The calculations pertaining to the thermal and fluid-flow phenomena in the core and in all components of the test unit were made in accordance with accepted engineering practice.³ The intent is to be conservative in estimating the heat-transfer and pressure-drop characteristics in view of the uncertain behavior of the physical properties of sodium. Conservative estimating is also necessary since the heat

transfer and pressure-drop correlations used in the calculations have yet to be verified experimentally for the design conditions of the proposed reactor.

Sodium is the liquid proposed for the test core. The conditions existing with mercury, cesium, rubidium, potassium, sodium, and lithium present are estimated. The heat-transfer and fluid-flow computations for this fast core were made assuming (1) high vapor-volume fractions can exist in the core without oscillations, or (2) if oscillations do occur, they will be of such magnitude as not to affect reactivity significantly. This problem is not associated with this core alone, but is shared by any system in which a liquid is boiled in the core of a nuclear reactor.

The vapor quality can be estimated from the presently available correlations for the flow of a two-phase mixture. However, only the actual operation of the reactor can determine what quality is compatible with stability. The pumping rate may need to be increased to reduce the vapor-volume fraction and/or subcool the sodium before it enters the core.

For a given system pressure, the power density of the boiling core is primarily limited by the burnout heat flux and, to a lesser degree, by the pressure drop. Therefore, for the same system pressure, the boiling core will have a lower power density than a core that only heats the liquid metal.

When the test core is operating as a heater, the estimated axial power distribution is as shown on the HEATER curve of Fig. 12; when the test core is operating as a boiler, the estimated power distribution is as shown on the BOILER curve of Fig. 12.

Fig. 12. Axial Power Distribution as a Function of % Axial Length

With the present arbitrary limitation of 2000°F placed on the temperature of the uranium carbide, the power density when the test core is

operating as a heater is estimated to be four to five times as great as when the core is operating as boiler. If it is possible to operate nearer the critical temperature of any of the liquid metals, this ratio of four to five will become smaller. The ratio decreases because the ratio of forced-convection heat-transfer rates of the liquid and vapor phases of a metal for the same mass flow approach one as the operating pressure approaches the critical pressure. The ratio of one cannot be approached now, because the saturation temperatures at the critical pressures are far in excess of the 2000°F limitation. For the present operating conditions, as the quality of the vapor increases, the convection heat-transfer rate decreases; it is therefore necessary to reduce the power density along the uranium carbide rod by reducing the effectiveness of the neutron reflector surrounding the core.

The average heat flux and average power density that may be expected with some assurance of stability and safety, when the reactor is operating as a liquid heater, are considered to be 450,000 Btu/ft²-hr and 0.45 MW per liter of core volume.⁴ The test core should approach these values cautiously until the operating characteristics are established.

With reference to the hypothetical core, consider two extremes of the spectrum of the liquid metals suggested for the power cycle, mercury and lithium. The specific heats of mercury and lithium at 1800°F are 0.038 and 0.985 Btu/lb-°F, respectively.⁵ The products of flow rate W (lb/sec), specific heat, and temperature difference in the liquid ΔT are as follows:

$$\text{Mercury: } W \times 0.038 \times \Delta T = 4743 \text{ Btu/sec;}$$

$$W \times \Delta T = 124,816 \text{ lb-°F/sec,}$$

$$\text{Lithium: } W \times 0.985 \times \Delta T = 4743 \text{ Btu/sec;}$$

$$W \times \Delta T = 4815 \text{ lb-°F/sec.}$$

The average heat flux and average power density associated with the reactor, when it is operating as a boiler, is considered to be of approximately 100,000 Btu/ft²-hr; this value results from studying several documents.^{3,6-9} The burnout heat flux for boiling liquid metals as determined by investigation varies from 0.45 to 55×10^6 Btu/ft²-hr. The variation in these predictions is a result of disagreement among investigators as to the importance of the influence of the thermal diffusivity of the liquid phase on burnout heat flux. The determination of the actual values of the heat flux and lower density for the various liquid metals flowing through the prototype of this core should be the first experimental work done in connection with this concept.

As a boiler, the reactor is capable of putting 1.0 MW of heat into the liquid metal. The heats of vaporization of mercury and lithium at

1800°F are 120.5 and 9000 Btu/lb, respectively. The products of the flow rate W (lb/sec) and the latent heat h_{fg} (Btu/lb) are as follows:

$$\begin{aligned}\text{Mercury: } W \times h_{fg} &= 960 \text{ Btu/sec;} \\ W &= 7.9 \text{ lb/sec.}\end{aligned}$$

$$\begin{aligned}\text{Lithium: } W \times h_{fg} &= 950 \text{ Btu/sec;} \\ W &= 0.106 \text{ lb/sec.}\end{aligned}$$

The liquid metal enters at a temperature close to that of saturation. The major portion of the heat transferred from the uranium carbide rods to the liquid metal is the heat of vaporization.

The difference in heat-transfer characteristics of the various liquid metals that may possibly be used apparently does not significantly affect the value of the heat flux.³ The rate of heat transfer q (Btu/ft²-hr) is expressed by the relationship $h = q/A(T_s - T_l)$ Btu/ft²-hr-°F.

For boiling water, numerous empirical formulas exist for calculating h from the physical properties and fluid dynamics. These formulas should not be assumed to be valid for any of the liquid metals. Some experimental determinations of the relationships between q and $(T_s - T_f)$ exist for mercury and sodium. These determinations indicate that there is an optimum temperature difference and that the temperature difference value is strongly dependent upon the additives that affect the wettability of the surface of the rods. In this design the surface is niobium-1% zirconium, and no data exist for surfaces of this material that can indicate the optimum $(T_s - T_a)$ or the relationship between q and $(T_s - T_f)$. Boiling-liquid-metal data have a wide variation, as seen from the Liquid Metals Handbook.⁵

The effects of the following factors cannot be calculated: (1) geometry of the surface, (2) finish of the surface, (3) amount and type of additives used to enhance wettability, (4) pressures of the liquid metal, and (5) heat flux in the boiling region influenced by physical properties of the liquid metal. These factors must be determined experimentally for the specific design. For example, the heat flux from a plane surface will not be the same as that from a surface in the form of a bundle of round rods.

The power leaves the heater as a hot liquid and is available only in the form of heat ($W \times C_p \times \Delta T$). If W_f is small, ΔT is large and the temperature of the rejected heat in the power cycle is low, and if this is a space installation the radiator will be large.

The power leaves the boiler as a liquid-metal vapor and is available as heat or as kinetic energy. The liquid metal will return to the reactor

minus the heat of condensation and with a few degrees of subcooling. If vapor is expanded, the liquid metal will return at a much lower temperature but not as low as if an intermediate heat exchanger were in the cycle.

Some of the physical properties of liquid metals in both the liquid and vapor phases are questionable, and with the large number of these properties involved in the calculations and the lack of experience, the heat-transfer rates and pressure losses calculated for this concept are subject to corrections. An effort has been made to ensure that these calculated values are conservative. Three categories of heat-transfer rate may exist along a flow channel through the core. They are the ones associated with (1) transfer of heat by forced convection from a solid surface to the liquid, (2) boiling, and (3) vapor phases of the liquid metal flowing across the surface. The heat-transfer rates for the liquid and vapor regions of the core can be calculated with some confidence, but the length and surface of each region will depend on the value of the heat flux in the boiling regions.

The heat-transfer rate and pressure loss through the core when it is operating as a liquid-metal heater (4743 Btu/sec for heating the liquid) are given in Table III.

TABLE III. Heat Transfer and Pressure Loss Associated with Liquid-metal Heater

Q (total heat): 4743 Btu/sec ΔT assumed: 400°F Q/ ΔT : 11.86 Btu/sec-°F T exit: 1832°F						
Liquid Metal	Hg	Cs	Rb	K	Na	Li
W, lb/sec	297.	208.	148.	59.	38.	12.
W/ ρ , ft ³ /sec	0.423	2.08	1.90	1.52	0.84	0.444
v, ft/sec	6.94	34.1	31.0	24.8	13.7	7.29
P out, lb/in. ²	3500.	140.	105.	90.	40.	1.00
T out, °F	1832.	1832.	1832.	1832.	1832.	1832.
ρ avg, lb/ft ³	700.	100.	78.	39.	45.	27.
K avg, Btu/hr-ft-°F	10.0	10.7	8.0	19.5	31.	29.
C _p avg, Btu/lb-°F	0.040	0.057	0.080	0.201	0.316	0.989
μ avg, lb/ft-hr	1.6	0.70	0.30	0.26	0.34	0.50
Pr, C _p μ /K	0.0050	0.0040	0.0030	0.0036	0.0035	0.0170
Pe, DV ρC_p /K	590.	550.	720.	410.	190.	204.
Re x 10 ⁻³ , VD ρ / μ	92.	176.	236.	113.	55.	12.2
D/K x 10 ³	0.84	0.79	1.05	0.58	0.27	0.29
f x 10 ²	0.50	0.45	0.38	0.48	0.55	0.75
h, Btu/ft ² -sec-°F	3.30	3.50	4.15	4.80	9.5	8.9
ΔP , lb/in. ²	3.4	10.7	5.8	2.4	1.0	0.25
G x 10 ⁻³ , lb/ft ² -sec	2.43	1.66	1.22	0.49	0.32	0.10

The formulas used in the calculation are⁶

$$Nu = 0.93 + 10.81R - 2.01R^2 + 0.0252R^{0.273} (\psi Pe)^{0.8};$$

$$\Delta P = \frac{fL\rho_f V_f^2}{2gD}.$$

Estimating h and ΔP is difficult because the geometry of the surface, the finish of the surface, and the additives used to enhance the wetability, all influence the questionable values of the physical properties.

The heat-transfer rates and pressure losses through the core when it is operating as a boiler (950 Btu/sec for vaporizing only) are given in Table IV. The values listed⁵ should be taken only as indicative of the relative merits of the liquid metals considered.

TABLE IV. Heat Transfer and Pressure Loss Associated with Liquid-metal Boiler

Q (total heat): 950 Btu/sec q (heat flux): 100,000 Btu/ft ² -hr T _{exit} : 1,800 °F											
Liquid Metal	Hg		Cs		Rb		K		Na		Li
P sat, lb/in. ²	3500.		140.		105.		90.		40.0		1.0
h _{fg} , Btu/lb	120.		187.		321.		800.		1620.0		8850.
W _g , lb/sec	7.91		5.08		2.95		1.19		0.59		0.11
G, lb/ft ² -sec	65.		42.		24.		9.8		4.8		1.1
Phase	f	g	f	g	f	g	f	g	f	g	
ρ, lb/ft ³	700.	30.	100.	0.7	78.	0.4	39.	0.15	45.	0.04	
W/ρ, ft ³ /sec	0.025	0.510	0.113	16.2	0.086	16.7	0.068	17.8	0.029	32.3	
v, ft/sec	0.205	4.4	0.927	131.	0.705	136.	0.558	147.	0.238	265.	
K, Btu/hr-ft-°F	10.0	0.010	10.7	0.004	8.0	0.005	14.5	0.009	31.0	0.017	
C _p , Btu/lb-°F	0.040	0.030	0.060	0.060	0.080	0.100	0.200	0.210	0.320	0.620	
μ, lb/ft-hr	1.60	0.40	0.70	0.07	0.30	0.07	0.26	0.05	0.34	0.06	
Pr, C _p μ/K	0.05	1.20	0.04	1.16	0.03	1.30	0.036	1.25	0.035	2.10	
Pe × 10 ⁻² , DV _p C _p /K	0.007	121.	0.016	447.	0.017	305.	0.090	156.	0.039	157.	
Re × 10 ⁻² , DV _p /μ	27.1	101.	40.0	399.	55.4	234.	25.	125.	9.5	60.	
f × 10 ⁺² friction factor	1.40	0.80	1.00	0.60	0.90	0.65	1.30	0.70	1.50	0.90	
Re ^{0.8} × 10 ⁻²	5.60	16.	7.6	48.0	10.	32.	5.24	18.8	2.4	10.6	
Pr ^{0.4}	0.300	1.08	0.276	1.06	0.246	1.11	0.249	1.09	0.262	1.34	
D/K × 10 ²	0.084	84.	0.079	227.	0.105	156.	0.058	93.	0.027	49.	
h × 10 ⁻² , Btu/ft ² -hr-°F	46.	0.470	61.	0.520	54.	0.530	0.52	0.510	53.5	0.57	
ΔP, lb/in. ²	<2	<2	<2	<2	<2	<2	<2	<2	<2	<2	

The heat transfer and pressure loss are calculated by the following formulas:³

Liquid: $Nu = 0.93 + 10.81R - 2.01R^2 + 0.0252R^{0.273} (\psi Pe)^{0.8};$

Boiling: $q = 100,000 \frac{\text{Btu}}{\text{ft}^2\text{-hr}}, \text{ when } \Delta T_{s-f} = 10^\circ\text{F};$

Vapor: $Nu = 0.023Re^{0.8}Pr^{0.4};$

Friction: $\Delta P = \frac{fL\rho V^2}{2gD}.$

A literature search revealed little information on the pressure loss associated with the two-phase flow of liquid metals. The pressure loss due to friction may be no greater than that for 100% vapor flowing the entire length of the core.

Lithium cannot be boiled at 1800°F because the saturation temperature is known to be about 3000°F.

Heat transfer is an important aspect of this design, and the need for basic work is paramount.

D. Dynamics

This fully automatic, nuclear power unit constitutes a closed-loop, nonlinear system with internal and external feedbacks. The internal feedbacks are designed into the system. The variable external feedbacks are designed to be functions of the operating conditions.

The cylindrical-shell neutron reflectors encompassing the core are designed to move in opposite directions from the transverse centerline of the core. Thus the core is uncovered, permitting neutron leakage. These two halves of the reflector have the same worth and can move in unison or independently of each other.

In the event of an accident, the two halves are rapidly and completely removed from the core. This combined reactivity worth is such that their removal scrams the reactor for every postulated type of accident.

During normal core operation as a heater, these two halves will float about a position of 99% effectiveness; when the core is operating as a boiler, one-half of the reflector will be at 99% and the other at <99% effectiveness.

The mechanism for moving the two halves of the reflectors are rack and pinion drives. These mechanisms are discussed in Section III-F on Mechanics.

The block diagram of the control system is shown in Fig. 13. The internal-feedback loop is within the area bounded by the dotted lines.

The principal sources of internal feedback in this core are the volumetric expansion of the core fuel and the variations of the density of the fluid flowing through the core.

The feedback due to the Doppler phenomenon is estimated to have little effect because of the small volume of this core. The variation of reactivity as a function of volume changes and of liquid-metal density is shown in Figs. 9 and 10, respectively.

Fig. 13. Block Diagram of Control System

The feedback due to expansion or contraction of the core is essentially prompt, except for inertial effect of mass increments along the fuel rods in response to temperature-dependent expansive and contractive forces.

The core can have the desired amount of movement both axially and radially in response to an increase in temperature of the fuel. This can be accomplished by designing the core so that it expands as a cylindrical volume against an elastic restraint. The elastic restraint is just sufficient to hold the radially-spaced fuel elements together as an integral cylinder and yet allow this cylinder to move radially and axially.

The feedback due to density variation is oscillatory and inconsistent because the boiling phenomenon is inherently unstable but controllable by design to within a narrow range. Therefore, the effect of this feedback is minimized by making the fluid volume in the core small at the expense of pump power.

The fluid density changes, primarily because of the formation of vapor and to a lesser extent because of the coefficient of expansion of the fluid.

The feedback from the fluid-density variations will lag the power variation. The amount of lag cannot be determined until a thorough dynamic analysis is made. The reactivity associated with the density of the fluid is less than the reactivity associated with the volumetric changes in the core.

The fluid density and its concomitant effect upon reactivity can be varied at will during operations by varying the power and/or the speed of the forced-circulation pump.

The change in reactivity due to an increasing rate of vapor formation is negative. Conversely, if the boiling rate is decreasing, the reactivity feedback is positive. An increase in system pressure suppresses boiling and causes a positive feedback of reactivity, the rate and direction of a pressure change determining the nature of the feedback. The interaction and combined effect of varying any or all of the components of the total feedback, in any desired manner, can readily be seen on an analog of the reactor system.

Another secondary cause of reactivity variations is the fluid-volume change in the flow channels caused by the expansion and contraction of these channels, which produces a feedback due to space-dependent neutron diffusion. This reactivity variation has a small effect because of (1) the small change in volume, (2) the low absorption and scattering cross sections in a fast spectrum, and (3) the low volume percentage of fluid in this core.

The volumetric expansion of the core is most likely to be caused by temperature; the fluid volume and density variations can be caused by pressure, temperature, and/or fluid dynamics.

The core constitutes a sensitive, nonlinear, closed-loop system with possible positive components of feedback but a predominantly negative, nearly prompt, combined feedback in response to increasing power for the anticipated operating conditions and hypothetical accidents. Whether the reactor core is stable at an attractive power density will depend on the magnitude and phase relationship of internal feedbacks that are designed into it. The core can have a high power density and a short time constant, and under this condition it is extremely sensitive to reactivity feedback. The dynamic characteristics of this core will be studied before it is built.

The nonlinear closed loop of the core is within the external closed loop of the reactor, which couples a loop with fixed characteristics to a loop of variable characteristics. This system can be simulated or studied on an analog.

E. Choice of Liquid Metal

Comparing the merits of liquid metals on the basis of some isolated parameter is easy and often misleading. All the parameters involved must be comprehended in order to select the liquid metal that is best for a specific system in a specific environment. In the EBR-I, Mark V modification, one must determine by physical test which liquid metal, when vaporized in this core of a solid-fueled fast-neutron reactor, will have characteristics that are comparable with the highest degree of dynamic stability and thermodynamic performance.

We propose to investigate mercury, cesium, rubidium, potassium, sodium, and lithium. The core and test equipment are designed for a wide range of operating conditions centering around the conditions concomitant with the characteristics of sodium, because in this relatively unexplored area of technology, the characteristics of sodium are best known.

The first consideration in the choice of a liquid metal for use in the reactor core depends on the pressure of the environment. The pressure of the fluid within the core shell should be greater than the pressure of the environment but not greater than 400 lb/in.^2 , since both the core shell and the reactor-containing shell are designed, with a safety factor of three, for 200 lb/in.^2 .

A metal that has low cross sections for nuclear reactions is desirable so that fluctuations in the number of atoms of this liquid metal in the core at a given time and place do not cause significant changes in reactivity.

The enthalpy, latent heat, and specific heat are important. If these values are large, less mass flow of metal is required to provide the thermal power, which means less volume and weight of reactor, heat exchangers, and machinery, and fewer atoms to adversely affect the reactivity.

Temperature has a far-reaching effect upon the design, since all the physical properties of the liquid metal, the nuclear materials, and the structural materials are functions of temperature. At present, the maximum liquid-metal temperature of 2000°F is limited by the structural and chemical integrity of the uranium carbide rods. The next limiting factor, by a few hundred degrees, is the structural material. As the temperature limit is lifted, the use of liquid metal in a nuclear reactor becomes more attractive as the operating temperatures approach those associated with the critical pressure. Furthermore, the performance of the power plants, and especially the performance of those that may be used in space missions, will improve.

The characteristics of the metals compared herein as possible cycle fluids are presented in Reference 5. Vapor pressure vs temperature, and density vs saturation pressure are plotted in Figs. 14 and 15, respectively.

Some considerations in the choice of a cycle fluid are as follows:

1. At any given temperature, lithium has a lower vapor pressure than sodium, potassium, rubidium, cesium, or mercury, thus requiring less thickness of pressure parts and reducing the weight of the reactor vessel, the piping, turbine, pumps, and radiator, but at the expense of large volume.

Fig. 14. Vapor Pressure vs Temperature

Fig. 15. Density vs Saturation Pressure

The vapor pressures (psia) at 2460°R for the liquid metals considered are as follows:

<u>Li</u>	<u>Na</u>	<u>K</u>	<u>Rb</u>	<u>Cs</u>	<u>Hg</u>
2	75	140	240	200	6500

The highest cycle temperature compatible with material reliability over the expected lifetime of the power system under consideration is 2000°F, corresponding to a pressure of 75 psi for sodium.

2. The temperature of the cycle fluid on the reject part of the power cycle should be as high as compatible with good efficiency, especially for power systems that reject heat by radiation. The weight of the radiator can be a large percentage of the total weight of the power plant. Therefore, the higher the rejection temperature the better. For a given condenser pressure, the ratios of radiator areas for several liquid metals with respect to the area required for sodium are as follows:

<u>Li/Na</u>	<u>Na/Na</u>	<u>K/Na</u>	<u>Rb/Na</u>	<u>Cs/Na</u>	<u>Hg/Na</u>
0.2	1.0	1.9	2.6	2.4	35.7

On the basis of the first two considerations, lithium appears to be the choice. However, the vapor pressure of lithium corresponding to the highest permissible temperature, which we have arbitrarily set for liquid metal at 2000°F, is 2 psia.

For convenience, some physical properties of the various liquid metals are presented in Table V for comparison at atmospheric pressure.⁵

TABLE V. Physical Properties of Liquid Metals⁵

	Hg	Cs	Rb	K	Na	Li
Atomic weight	200.	132.91	85.48	39.10	22.997	6.94
Melting point, °F	-38.	83.	102.	145.8	208.	351.
Boiling point, °F	674.	1260.	1270.	1400.	1618.	2437.
Density of liquid, lb/ft ³	794.	104.7	82.15	41.38	46.4	31.8
Density of vapor, lb/ft ³	0.26	0.096	0.060	0.0305	0.0165	
Thermal conductivity, Btu/hr-ft-°F	8.1	10.65	11.8	18.	30.1	27.2
Specific heat (liquid), Btu/lb-°F	0.0323	0.0572	0.0877	0.1868	0.305	0.98
Specific heat (vapor), Btu/lb-°F	0.0248	0.0372	0.0579	0.1267	0.214	0.716
Latent heat of fusion, Btu/lb	5.07	6.90	11.79	25.5	48.7	186.0
Latent heat of vaporization, Btu/lb	125.7	213.8	347.5	851.	1609.	9108.

F. Mechanical Design

The mechanical aspects of the proposed experimental reactor are the control system, the pumps, and the structure. Each has been explored adequately so that there should be no insurmountable obstacle in the detailed design work.

The functions of the reactor control system will be (1) positioning the reflector in response to operating demands and (2) ensuring the safety of the reactor in the event of an accident involving the input of an excessive amount of reactivity into the core.

1. Control Mechanisms

The control mechanisms associated with EBR-I, Mark III, which consist of the external safety rods, the scram plug, and the elevator platform, are considered in two parts. The first

part is concerned with a description of the existing mechanism associated with the EBR-I, Mark III operation. The second part is concerned with the modification and calibration of these mechanisms so that they are compatible with the proposed test reactor.

a. Existing EBR-I, Mark III, Control System. The external safety-rod drive is shown in Fig. 16. The mechanism consists of a rack, a 3-in. pinion gear, an electric clutch, and a gear motor. As the gear motor slowly drives the rack upward at approximately 17 in./min for insertion, it winds up a flat, spiral-wound motor spring approximately $4\frac{1}{2}$ times, which exerts a torque of approximately 150 lb against the pinion gear. Upon breaking the electric circuit to the magnetic clutch, this spring drives the rod downward with a maximum velocity of 100 in./sec. Deceleration at the bottom end of the stroke is accomplished by driving the connecting shaft into a close-fitting, safety-rod guide cylinder, trapping an air cushion between the cylinder and connecting shafts.

Fig. 16. External Safety-rod Drive

The safety-rod velocity and acceleration vs travel and the safety-rod travel vs time are shown in Figs. 17 and 18, respectively. The safety plug and drive are shown in Fig. 19.

Fig. 17. Safety-rod Acceleration and Velocity vs Travel

Fig. 18. Safety-rod Travel vs Time

Fig. 19. Safety Plug and Drive

ion in the bottom of the cylinder. The acceleration vs travel, velocity vs travel, and travel vs time are shown in Figs. 20, 21, and 22, respectively.

The elevator mechanism (Fig. 23) consists of a large hydraulic piston centered directly beneath the two platforms that support the lower shield section, safety-plug drive, and external safety-rod drive. This assembly can be raised or lowered by a hydraulic pump mounted in an adjoining area.

Starting at the down position, the elevator advances at 24 in./min until a total rise of 50 in. is reached, when the elevator is stopped. When the elevator is raised above this 50-in. height, it advances at 2 in./min until it contacts the adjustable jackscrews at approximately 80 in. At this point the elevator may be accurately positioned up to $4\frac{1}{4}$ in. by the four, motor-driven jackscrews on the under side of the ceiling in the conveyor room.

In the event of scram operation, the elevator descends the first 30 in. with an acceleration of approximately 0.8 times gravity, after

The safety plug is centered at the base of the cup directly beneath the reactor tank in the "in" position. This is a uranium plug weighing 81 kg, jacketed with stainless steel. The movement of the plug is determined by a pneumatic cylinder resting on the lower elevator platform. The cylinder has a 4-in. bore and a 7-in. stroke. The plug is advanced to the "up" position before the elevator is raised, and is held in this position by air pressure in the cylinder. In the event of power failure, the valves assume the normally open position and drive the plug down away from the reactor. The up stroke of the plug is slow and metered at 28-lb pressure; the down stroke or scram is driven by 75-lb pressure at 55 ft/sec, and scram time is approximately 0.25 sec per 6-in. travel. The plug is brought to rest by a 2-in. cush-

Fig. 20. Safety-plug Acceleration vs Travel

Fig. 22. Safety-plug Travel vs Time

Fig. 21. Safety-plug Velocity vs Travel

Fig. 23. Elevator Mechanism

which the acceleration decreases and allows the conveyor to rest against the lower stops. If power fails, the elevator descends in the same manner as for scram operation. The shock absorbers are designed for use only in the event of piping or gasket failure.

Fig. 24. Beryllium Oxide Reflector Drives

platform and the beryllium oxide reflector can be raised or lowered independently of the other three drives. This drive and location arrangement will be very accurate and will permit the operator to position the reflector to within 0.001 in. of the desired location.

b. Additions and Modifications of Control System for Proposed Reactor. The proposed beryllium oxide reflector drives are shown in Fig. 24. The external safety drives will be used to move the beryllium reflector and will be referred to henceforth as the beryllium oxide reflector drives. Only four of the eight existing drives will be used. These drives are mounted to the underside of the upper platform located 90° apart on a 21-in.-diam circle. The upper ends of the connecting rods will be modified so that they may be connected to the beryllium reflector. This reflector will take the form of a 3-in.-thick annulus, which encircles the inner tank. The reflector will have an inside diameter of $17\frac{7}{8}$ in. and an outside diameter of $23\frac{7}{8}$ in. The movable reflector will be 11 in. high and will be divided into four equal quadrants, connected to the beryllium oxide reflector drive. Operation will be similar to the EBR-I, Mark III system. Some modifications may be necessary in the gear motor arrangement to reduce the insertion speed to approximately 2.5 in./min.

One of the four beryllium oxide reflector drives will be mounted to a platform connected to the underside of the upper platform by three motor-driven jackscrews. This

2. Result of Preliminary Analysis

Preliminary investigations indicate that the scram data and curves used with the Mark III reactor will not change appreciably because of the experimental modifications. Therefore, pertinent data contain scram characteristics similar to those listed for External Safety Rod Drives of Mark III.¹⁰

3. Calibration of Existing EBR-I Control Mechanisms

The present arrangement of safety-plug and elevator mechanism is satisfactory, and no modifications are contemplated at this time.

The following connotations concerning calibration of controls have been taken from ANL-5023¹⁰ and list the effectiveness of four control rods, eight external safety rods, two internal safety rods, and one safety plug:

"The pile was originally built with four types of controls, consisting of eleven safeties and four running controls as follows:

1. Four control rods (described in ANL-4420)^{11*}
2. Eight external safety rods (described in ANL-4420)^{11*}
3. Two internal safety rods (described in ANL-4420)^{11*}
4. One safety plug (described in ANL-4554)^{12*}

"Their calculated effectiveness (ANL-4356)^{13*} was as follows:

Four controls plus eight external safeties	- \$3.80 (1064 ih)
Two internal safeties	- \$3.15 (882 ih)
Safety plug	- \$1.60 (448 ih)

"It was estimated that dropping the external blanket would cause a reactivity change of \$11.00 (3080 ih).

"When the pile was started up it was found that the controls were not as effective as calculated, necessitating some changes in the design of the control system.

"The four running controls were calibrated (see Figure 4**), and their total effectiveness was found to be 40 ih. This also showed the external safety rods to be worth 80 ih, since the rods themselves are identical.

*Reference number refers to reference at end of present report, rather than a reference in ANL-5023.

**This figure appears in ANL-5023 and is not repeated in the present report.

"It was found that a single control rod run at high speed could effectively control the pile on automatic control; so one rod was left as designed. The other three rod drives were reworked by removing the differential gear box and replacing the motors with constant speed motors to speed up the travel of the rods so they could be used for a coarse control. The control panel was also changed so these rods could be run either individually or ganged.

"The internal safety rods containing the enriched boron were designed to compensate for the negative temperature coefficient. These rods were found to be worth only 30 ih apiece, which was too little control for the contemplated use. These were therefore removed from the pile, and blanket rods were installed in their place. It was necessary to design another control system to compensate for the temperature coefficient. This was accomplished in the following manner:

"It was found that the external blanket was worth 350 ih in the top four inches of its travel. This being sufficient to compensate for temperature coefficient, a system was designed to accurately control the travel of the cup during the last 4-1/4" of its travel. This was accomplished by installing four screw jacks on the elevator stops, driven by a motor through a worm gear train, to slowly raise or lower the external cup during this last portion of its travel. The jacks are of sufficient size to overcome the hydraulic pressure on the ram holding the external cup, so no changes were necessary on the elevator hydraulic system to incorporate this control system. The elevator is raised until it reaches the jacks; then the jacks take over the control of the cup position to finally locate the cup in the running position. Figure 5* shows the calibration of this control.

"The complete removal of the external cup was found to be worth 3220 ih. This is in agreement with the theoretical estimate.

"The safety plug in the bottom of the external cup was found to be worth 27 ih. This safety control was left as designed.

"The pile, therefore, as it is presently built has the following controls:

1. Three coarse and one fine control worth 40 ih total.
2. Eight external safety rods worth 80 ih total.
3. One safety plug worth 27 ih.
4. External cup jack control worth 350 ih.
5. The complete external cup worth 3220 ih. (The entire cup is dropped away from the core on a scram.)

*This figure appears in ANL-5023 and is not repeated in the present report.

"This gives a total of 390 ih in the controls. The fast acting safety controls can remove 107 ih in 100 milliseconds. The slower acting cup removes 3,220 ih in a time determined by the fact that the downward acceleration of the cup is slightly less than 'g'."

4. Additional Design Features.

The forced-circulation pumps may be either electromagnetic or multiphase,¹ driven by electric motors or turbines. An electromagnetic pump will be simple and reliable, but the necessary electrical supply presents problems of weight and volume. If multiphase pumps are used, there are the problems associated with bearings and seals but relatively simple compact drives in the form of motors or turbines.

The reference design uses a helium-driven turbine centrifugal pump, capable of carrying the full power load.

The maximum shutdown heat is 150 kW, which can be easily handled by the system. The pumps must be kept in operation at all times once the core has been put into operation.

The structural arrangement of this design is shown in Fig. 1.

The core consists of uranium carbide rods clad with niobium-1% zirconium. These rods are welded into tube sheets (see Fig. 4), and the subcritical bundle inserted into a cylindrical shell of niobium-1% zirconium (see Fig. 3). This core is then suspended in a box frame of columbium, which also serves as the structural base for the control mechanisms, controls, and pumps. The box frame is inside the stainless-steel encompassing shell. The shell annulus is a vacuum; therefore the reactor and all structural work are in an atmosphere of helium.

Both the core shell and the encompassing shell are designed to withstand an internal pressure of 200 psia at 1500°C. Thus, if the power unit was intended to operate in a deep-sea environment where the pressure was 3000 psia, the pressure of the helium would be 3200 psia and the pressure of the mercury in the core would be 3400 psia. If the power unit is in space, the pressure of the helium could be 10 psia and the pressure of the lithium in the core 20 psia.

G. Materials of Construction

Although a boiling, liquid-metal reactor will be dynamically stable at some power density, the materials constraining the cycle fluid set the top temperature limit of the cycle. A high power density and a high cycle temperature are the objectives; therefore the dynamic characteristics designed into the reactor system and the compatibility of its materials are essential to a successful design.

Materials considered promising for structural use are (in the order of their preference): columbium, tantalum, and vanadium. Some of their properties are listed in Table VI.

TABLE VI. Physical Properties of Structural Materials

	Cb	Ta	V
Density	8.6	16.6	5.7
Melting point, °C	2415	2996	1710
Thermal conductivity, cal/(sec)(cm)(°C)	0.131	0.131	0.074
Coefficient of thermal expansion $\times 10^6$, °C	6.8	6.5	8.6
Specific heat	0.0065	0.0036	0.0019
Electrical resistivity, microhm-cm	14.2	12.4	25
Modulus of elasticity, kg/cm ² $\times 10^{-5}$	11.2	18.9	14.0
Tensile strength, kg/cm ² $\times 10^{-3}$	70	100	80
Hardness Rockwell	B100	B95	B93
Workability (cold-worked)	Easy	Easy	Easy
Machinability	Like steel	Like steel	Like steel
Weldability	Special methods	Special methods	Poor
Corrosion resistance (at room temp.) (at 1000°C)	Good Unknown	Good Unknown	Very good Unknown

Columbium is preferred for the following reasons:

1. Stronger at high temperatures than vanadium or tantalum.
2. Lowest thermal-neutron-capture cross section of the refractory group.
3. Most easily sintered, forged, swaged, rolled, machined, and welded.
4. Low rate of work-hardening.
5. Reasonably compatible with most uranium-base fuel materials.
6. Not attacked by most molten metals.

The promising present and potential applications for columbium in the nuclear field for fuel-element cladding has led producers of the materials to hold undesirable impurities to several hundred parts per million. The most important undesirable impurities found in reactor-grade columbium are carbon, oxygen, hydrogen, and nitrogen. In service at high temperature, these elements cause embrittlement as well as decreased corrosion resistance. The present reactor-grade columbium has excellent corrosion resistance, and short-term tests indicate that it is suitable for use with alkali metals at 2000°F. The chemical composition of reactor-grade columbium is shown in Table VII.

TABLE VII. Chemical Composition of
Reactor-grade Columbium¹⁴

Cb	99.6%	H	20 ppm
Ta	0.1%	N	300 ppm
C	0.02%	Zr	Trace
O	0.05%		

Columbium can be fabricated as easily as cold-rolled steel. Tubing can be swaged, bent, spun, stamped, forged, rolled, and drawn. Columbium can be welded to itself, to tantalum, and to zirconium by gas-arc welding or resistance welding.

In the past $2\frac{1}{2}$ years, nearly 100 tons of the columbium-1% zirconium alloy have been melted. This has been converted to bar, sheet, tubing, piping, wire, and die-impressed forgings. Sheet has been produced up to 36 in. in width, and seamless tubing in 1/8-in. OD, 0.006-in. wall has been cold-drawn on a production basis. Forming operations such as bending, hydrospinning, deep-drawing, and machining have all been successfully performed. Insert-gas shielded-arc welding has been completely successful on sections up to 1/2 in. thick with weld efficiencies of from 90 to 100%.

Tantalum exhibits unique corrosion-resistant properties, and like columbium it is sensitive to atmospheric contamination at elevated temperatures. The chemical composition of reactor-grade tantalum is shown in Table VIII.

TABLE VIII. Chemical Composition of
Reactor-grade Tantalum¹⁴

Ta	99.74%	O	0.07%
Cb	0.10%	H	0.01%
C	0.05%	N	0.03%

Tantalum has been used extensively in the chemical industry for heat-exchanger and condenser tubes which are in contact with chlorine, chlorides, hydrochloric acid, and nitric acid. It has not been used in the boiling alkali metal but may be satisfactory.

Vanadium, in group VA of the Periodic Table, is the most recent refractory metal to be manufactured in small-diameter, close-tolerance tubing. Little information on vanadium is available as of this date.

H. Shielding

1. Neutron Fluxes

A ten-group calculation with the DSN code was used to secure the neutron flux incident on the shielding. An 8-cm-thick beryllium oxide reflector and a 17-cm air gap separate the graphite from the core.

The neutron-flux distributions in the shield regions were determined by the RE-34 shielding code using six energy groups. The distributions are shown in Fig. 25.

2. Core Gamma Rays

The core gamma-ray calculations were based on the following assumptions and procedures:

- a. The gamma rays were grouped into 1-, 2-, 4-, and 6-MeV energy groups.
- b. The gamma-ray fluxes at the surface of the core were determined for 1-MW power operation and a core radius of 13.52 cm.
- c. Prompt fission and fission-product gamma rays, as a function of energy, were as follows:
 - 8.61 MeV/fission of 1-MeV photons;
 - 4.82² MeV/fission of 2-MeV photons;
 - 1.357 MeV/fission of 4-MeV photons;
 - 0.256 MeV/fission of 6-MeV photons.

Fig. 25. Neutron- and Gamma-flux Distribution in Shielding

Table IX lists the gamma-ray absorption cross sections used in gamma-ray flux calculations. The core gamma-ray flux in the regions outside the core was determined by the following formula, which effectively includes a linear buildup factor:

$$\phi_{\gamma} \left(\sum_{i=1}^i \right) t_i = \frac{Q_2}{2} \left(\frac{r}{r_0} \right) \exp \left(- \sum_{i=1}^i \mu_i t_i \right),$$

where

μ_1 = gamma-ray linear-absorption coefficient for the i^{th} region, cm^{-1} ,

$\frac{Q_2}{2}$ = radiating surface source, $\text{MeV}/\text{cm}^2\text{-sec}$,

r = radius of spherical source, cm ,

$$r_0 = r + \sum_{i=1}^i t_i,$$

and

t_i = thickness of i^{th} region, cm .

TABLE IX. Gamma-ray Constants¹⁵

E, MeV	Gamma-ray Linear Attenuation Cross Section, μ , cm^{-1}			
	Fe	BeO	Graphite	Ordinary Concrete
1	0.468	0.165	0.102	0.149
2	0.333	0.115	0.0710	0.105
4	0.259	0.078	0.0486	0.0745
6	0.239	0.064	0.0392	0.0630
8	0.232	0.054	0.0341	0.0571

3. Capture Gamma Rays from the Shield Regions

The gamma rays from regions outside the cone due to neutron captures were calculated from the equations listed in ANL-6000.¹⁵ The thermal neutron flux was represented by an exponential shape, and the appropriate equations were used to calculate the capture, gamma-ray flux at either surface of each source region and through specified exterior shield regions. Table X lists constants used in the capture, gamma-ray source calculations.

Based on the calculations, the total radiation dose outside the concrete during 1-MW operation is estimated to be 30 mR/hr. Nearly all of this is from capture gamma rays originating in the concrete and steel. This should be the worst possible case since an air gap is between the reflector and the graphite inner surface.

TABLE X. Constants for Capture Gamma-ray Calculations

	N_γ (MeV/Neutron Capture)			
	2 MeV	4 MeV	6 MeV	8 MeV
Steel	0.55	0.82	0.84	3.78
Graphite	-	-	4.95	1.34
Ordinary concrete ($p = 2.33 \text{ g/cm}^3$)	1.66	2.71	1.27	1.34

I. Fuel Material

The high operating temperature, coupled with the high fissile contents required of the fuel material to be employed in this core, excludes metallic alloy systems. Service requirements necessitate the selection of a fuel material from a limited number of uranium compounds about which physical, mechanical, and nuclear data have been developed. Of these compounds, UO_2 and UC are the best known, and available data indicate that these materials are potential nuclear fuels for this experiment. Uranium monosulfide and uranium nitride are other compounds which may be interesting.

Uranium carbide is selected as the reference fuel. This fuel has a higher melting temperature, a higher uranium concentration, and a higher thermal conductivity than uranium oxide. Because of insufficient knowledge, uranium sulphide is not seriously considered.

Uranium carbide fuel elements ranging in carbon content from 4.3 to 5.3 w/o have been prepared and jacketed with Type 304 stainless steel with sodium or NaK bonds between fuel and jacket. These slugs have been irradiated to burnups up to 25,000 MW-days per ton of uranium (2.8% metal atom burnup) at center temperatures up to approximately 1200°C and heat fluxes to 410 W/cm^2 .

Under irradiation conditions of approximately 7,000 MWD/MTU and center temperatures of approximately 800°C , little difference in behavior is noted in these uranium-carbide alloys containing from 4.6 to 5.3 w/o carbon. Fission-gas release was slight, and diameter changes were less than 1%. Irradiation of the same alloys under more extreme conditions, 20,000 MWD/MTU at 930°C , also showed excellent stability. However, hyperstoichiometric uranium carbide decarburized from the center with subsequent carburization of the stainless-steel cladding. In some cases the cladding carbon content increased to greater than 1 w/o. Preliminary analysis of the data from these experiments indicate that fission-gas release after 25,000 MWD/MTU at a center temperature of approximately 930°C will be less than 2% and probably about 1%. Less

than 3% diameter change may be expected for uranium-carbide slugs after 25,000 MWD/MTU at a center temperature of approximately 930°C. At a center temperature of 1200°C, a diameter increase of about 10% is indicated by extrapolation.

Current status of knowledge indicates satisfactory performance of unsectioned uranium carbide up to about 1300°C at burnups of about 10,000 MWD/MTU. Experience with enhancement of irradiation performance by the addition of fuel jackets leads to an expectation of even higher temperature performance.

The compatibility of uranium carbide with the niobium-zirconium alloys has been studied at various laboratories. Present data indicate compatibility of these materials at 1000°C and perhaps somewhat higher. Poor performance with hyperstoichiometric uranium carbide is anticipated. Carburization of the strongly carbide-forming columbium jacket at higher temperatures could present problems. If metallic bonding of fuel to cladding becomes necessary, interaction between fuel and bond will have to be determined and mass transfer studied.

The technology for fabricating uranium monocarbide and for the evaluation of the fabricated material is in the process of being developed. Commercial methods used to react uranium and carbon produce a heterogeneous mixture. This material is suitable melt stock for arc melting and casting; however, it is not suitable for powder fabrication. The stoichiometry of the uranium carbide is difficult to control, and the effect of either excess metal or carbon has not been fully studied.

The Chemical Engineering Division at ANL is developing a method for producing uranium carbide. This method, which precipitates uranium carbide from a zirconium-magnesium molten alloy, yields a good, almost stoichiometric material. This procedure permits close control over the particle size and the carbon content of the carbide. Uranium carbide formed by this method would appear to be suitable for powder fabrication. Initial tests have been promising, but high-density sintered carbide has not yet been produced from this material.

The solid-fuel slugs could be formed from uranium carbide by arc-casting, swaging of high-density material, vibratory compaction, pelletizing and sintering, or extrusion.

The tubular fuel might be formed by extrusion; however, this method has never been applied to thin-wall tubes. Of the methods listed, pelletizing and sintering presently appears to be best suited for fabrication of the small-diameter slugs or pins required for this loading.

A development program necessary to form uranium carbide by powder techniques would involve the following:

1. Developing a method for producing a uniform, sinterable, uranium-carbide powder.
2. Studying the compaction characteristics and sintering behavior of this powder.
3. Studying methods of plasticizing the carbide for either room temperature or hot extrusion.
4. Developing methods of producing a high-density sized grain, and evaluating vibratory compaction.

The technology for uranium sulfide is similar to that for uranium carbide. Uranium sulfide is more easily formed than uranium carbide and can be sintered to high densities. For this reason, uranium sulfide should be considered as a potential fuel. The development program for uranium sulfide would be almost the same as that suggested for uranium carbide.

Pressing and sintering techniques have been applied to the fabrication of uranium nitride pellets.

The technology for fabricating uranium oxide and the evaluation of its pertinent properties are well developed. In this category, only a minor amount of work specifically applied to the core would be necessary. The solid-fuel element could be fabricated by pressing and sintering pellets, extrusion, or vibratory compaction of high-density oxide. The density of either the pressed or extruded oxide is approximately 93 to 95% of theoretical density. The density of compacted uranium oxide is approximately 90 to 92% of theoretical density. The Ceramic Group is well versed in pelletizing or extruding uranium oxide of this diameter, and no problems are anticipated outside of tooling up.

The tubular element could be fabricated from uranium oxide by extrusion. The core pin could be made by pelletizing; however, the tube portion has too thin a wall for pressing. Because of the particle size range that would be needed, vibratory compaction would not be recommended. Extrusion of thin-walled tubes of uranium oxide has not been accomplished, and would present some development problems in die design and plasticizers. This would require time, but should not be a major problem.

Compatibility between the fuel and metal cladding should be evaluated at temperatures above those anticipated in the reactor.

IV. SAFETY

The most likely sources of accidents are those resulting from nuclear excursions and those resulting from the release of thermally hot, radioactive sodium.

The anticipated accidents will be the same as those provided for in cores that have been operated in EBR-I. The provisions for detecting and safeguarding against nuclear accidents are those existing now and will not be diminished but possibly augmented. Since this reactor boils the liquid metal, the amount of reactivity due to the presence of sodium in the core is less by design than the delayed-neutron fraction, β . The core is designed to expand as a solid cylinder; therefore it has a large, prompt, negative reactive feedback with increasing power.

If the double-wall pressure vessel and pipes leak or rupture, the high-temperature sodium will go into the primary NaK. The mixture will be 30 gal of saturated sodium at 2000°F expanding into 4000 gal of NaK at 600°F. The pressure, temperature, or reactivity level of the primary NaK will not rise any noticeable amount.

Operation of the helium-driven axial-flow pumps within the pressure vessel will eliminate the possibility of accidents resulting from the position of the electrically driven centrifugal pumps on top of the reactor shielding.

V. PROGRAM OF ANALYTICAL WORK

In all areas of analysis, there will be many groups of calculations as the design evolves in each area. The assumptions made throughout the calculations, the reasons for choosing the physical properties used, and the choice of boundary conditions will be clearly stated.

The coordinated analysis will produce a mathematical model from which the characteristics of the core can be determined so that the model and the core will have the same response to any given signal.

The principal areas of analytical work are considered to be:

- A. Nuclear
- B. Thermodynamic
- C. Mechanical
- D. Dynamic.

Before the start of the experimental phase, a test program and the design and hazards reports must be written.

A. Nuclear Analysis

The calculations associated with the nuclear analysis are in three sets. The first or immediate set is the type that establishes first-order approximations in the shortest time for use in the initial phase of the other areas of analysis. The second or intermediate set is for the purpose of initiating the design. The third or final set is of the highest caliber and will constitute the finished design, which will be used in the Design Report and the Hazards Reports.

1. Immediate Calculations

The immediate calculations are approximations and they will be used to enable the other analytical groups to rough out their work.

The following is a list of the work in this set of calculations, listed in the order of need:

- 1. Develop first-stage physics methods.
 - a. Choose cross section.
 - b. Estimate applicability of one-dimensional transport codes to the core.

2. Check out first-stage physics methods.
 - a. Correlate pertinent unclassified criticality data.
 - b. Check cross sections vs criticality and danger-coefficient data.
 - c. Check methods vs similar-size experimental criticals.
3. Determine the core dimensions, critical mass, and power distribution.
4. Determine the relationship between reactivity and a change in $\Delta V/V$, $\Delta R/R$, and $\Delta Z/Z$. Average values for the core will be satisfactory for the first set of calculations.
5. Determine the relationship between reactivity and a change in density of the sodium in the core. Average values of the core will be satisfactory for the first set of calculations. This calculation will require cooperation with the thermodynamic analysts.
6. Determine β_i , λ_i , and ℓ .
7. Determine the worth of the reflector alone as a function of position, and then the worth of the cup alone, both relative to the same reference.
8. Determine the worth of the Nb-Cb-Fe strata between the core and the reflector.
9. Calculate the following (approx.):
 - a. Shutdown heating.
 - b. Gamma heating in shell reflector, cup, and shielding.
 - c. Level of activity in the control room.
 - d. Time required for the reactor to cool to the power that will permit work on the test unit.
 - e. Sodium activation.
 - f. Amount of fission gas formed in the fuel elements.
10. Design a suitable source.
11. Design an oscillator, and determine its best location. The drive is to be located above the shielding.

12. Determine the effectiveness of the present EBR-I shielding.
13. Determine the optimum location for the ion chambers.
14. Plan critical experiments.

2. Intermediate Calculation

The intermediate calculations are used to initiate the design. This set of calculations is a two-dimensional, four-group PDQ.

The following is a list of work in this set of calculations, listed in the order of need:

15. Develop second-stage physics methods.
 - a. Analyze spectrum of the core.
 - b. Choose four-group energy spacings.
 - c. Collapse ten-group to four-group cross sections.
 - d. Develop two-dimensional, unsymmetric representation of the core.
16. Check out second-stage physics methods.
 - a. Correlate pertinent unclassified (and possibly classified, i.e., Pratt & Whitney) criticality data.
 - b. Establish equivalences between:
 - Ten-group Sn and four-group Sn calculations
 - Four-group Sn and four-group PDQ calculations
 - Four-group PDQ and experimental criticals.
17. Recalculate items 3 through 9, and this time calculate items 4 and 5 as a function of position in the core.
18. Calculate the variation in reactivity and the change in isotopic abundance as a function of irradiation time for constant full power and a variable flux. Then hold flux constant, and show how power varies.
19. Calculate the Doppler effect, and show why it produces a small feedback relative to that produced by $\Delta V/V$ and $\Delta \rho/\rho$ in this core.

3. Final Calculations

Several types of final calculations are required: general physics calculations performed with highest possible accuracy, calculations of secondary importance but inherently intricate (i.e., burnup), and calculations which by their nature can be performed on the finalized design only (i.e., Hazards Report). It is anticipated that two-dimensional transport-theory codes will be used in transient analysis.

The following is a list of the work in this set of calculations, listed in the order of need:

20. Develop three-stage physics calculations.
 - a. Check out TDC (Two-dimensional Code).
 - b. Compare TDC's with experimental critical data.
21. Recalculate items 1 through 19, and put items in finalized form, properly indexed, referenced, and explained.
22. Investigate the effects of the beryllium oxide reflector on neutron lifetime.
23. Make a thorough analysis of the effect of time dependence in the nuclear phenomena. This will in effect be a critical analysis of the mathematical model of the system and the techniques for transforming it to computer circuitry.
24. Analyze and correlate the critical experiments.
25. Plan the loading procedure.
26. Plan the startup procedure.
27. After careful consideration and analysis of all aspects, plan the experimental program. This is a coordinate effort of all analytical and experimental sections. It is very important to know what must be measured and recorded in this experiment before the experimental hardware is built.

B. Thermodynamic Analysis

The following calculations are considered important in the thermodynamic analysis (they are listed in an order of need that best coordinates them with those of the other categories of analysis):

1. Establish the heat-transfer rate to be used in the liquid section of the core.

2. Establish the heat-transfer rate to be used in the boiling section of the core.
3. Establish the heat-transfer rate to be used in the condenser.
4. Establish burnout heat flux to be used.
5. Determine the amount of sodium vapor that can be produced by this core and the recirculation rate for 100, 50, and 25% of full power.
6. Calculate the pressure drop through the core, the condenser, and the ducting, in the sodium circuit. Also calculate the pressure drop in the NaK circuit.
7. Calculate the incremental pressure drop along the core channel for 100, 50, and 25% fuel power.
8. Calculate the steady-state temperature at the centerline of the fuel rods and at the outside surface of the columbium cladding for 100, 50, and 25% power. Plot these temperatures along the length of the fuel rods and reflector.
9. Calculate the pressure and flow requirements of the recirculating pump over the range of operation.
10. Calculate the effect of subcooling at the inlet to the core.
11. Determine whether the liberating surface at the liquid-vapor interface is sufficient. If it is not, determine how much the supersaturated mixture must be subcooled.
12. Design the sodium-vapor condenser.
13. Determine the heat-transfer rate through the double walls of the container.
14. Calculate the heat generation in the beryllium oxide reflector and cup, and design a system for cooling these.
15. Develop the equations for transient temperature and dimensional variation in the core.
16. Develop the equations for the fluid dynamics of the system.
17. Determine if the present thermal insulation and shielding in the EBR-I installation has sufficient means of cooling.

The space- and time-dependent temperature and elasticity equations can be handled by making time a continuous function and by slicing the core volume into shells ΔR and ΔZ with an average heat generation in each shell of q_{ij} (see Fig. 26). Within each such volume element there will be a number of fuel rods surrounded by flowing sodium. These rods are sliced into shells Δr by ΔZ (see Fig. 27).

Fig. 26. Diagram Illustrating Subdivision of Core

Fig. 27. Diagram Illustrating Subdivision of a Fuel Rod

Heat is generated in the uranium and transferred throughout the reactor by conduction and convection, causing the materials of the reactor to undergo physical changes which affect the reactivity.

The heat generation throughout the volume of the core is determined from the solution of the diffusion equations as a function of space only. It is assumed that the curves of heat generation in the axial and radial directions do not change shape with a change in power, but that the radial curve shifts vertically by an amount directly proportional to the neutron density.

The core consists of a bundle of clad rods bound by a cylindrical shell. Thus the core represents a statically indeterminate, elastic system. A change in the fuel temperature will change the radial and axial dimensions of each individual rod, and their combined effect changes the dimensions of the core and thus the reactivity. The amount of change will be that concomitant with elastic equilibrium.

The uranium carbide fuel rod is clad with a niobium tube. The absolute temperatures in the uranium carbide are higher than those in the niobium, and the coefficient of expansion of the uranium carbide is greater than that of the niobium. Therefore, the expansion of the fuel is going to be restrained by the cladding, since the fuel will be in compression and the cladding in tension. Strain-energy methods are used to determine how much the fuel can expand axially and radially against the restraining influence of the cladding. Equilibrium exists when the strain energy of the fuel equals the strain energy of the cladding and core structure. The increase in length and diameter of each rod and the core will be that concomitant with this condition of elastic equilibrium.

The clad rod presents a three-dimensional problem in elasticity, but because of extreme complexities in the equations will be handled as a two-dimensional one, as is the core.

The effect of axial expansion is to increase the length of the rods and of the containing shell. The rods at the center of the core will try to expand axially more than the rods at the periphery, thus causing relative movement between rods. Any plane that was perpendicular to the center-line of the core axis at the steady-state condition, except that plane through the peak of the neutron buckling along the axis, will appear as a surface of revolution during the change in power and possibly afterward. The deflection of these planes at intervals ΔZ and of the radial movement of points ΔR on it must be known for a clear understanding of space- and time-dependent variations in reactivity.

The relative movement between rods is dependent upon temperature variation and the restraining effect occasioned by contact with adjacent rods or structure. Each rod attempting to move through the compacted bundle of rods is restrained to some degree. This restraining effect to axial motion may be occasioned by viscous friction, coulomb friction, or a rigid boundary.

The amount of friction and the nature of the friction between adjacent rods depend upon the degree of tightness occasioned by controlled internal clamping and by the interface contact pressure between adjacent longitudinal elements due to restrained radial expansion.

Since there are radial spacers between the rods, the centerlines of the rods remain essentially in the same relative position, and the force of thermal expansion is expressed as strain energy in the metals of the core. As the temperature of the metals increases, there is compression of the rods and spacers but, at the same time, stretching of the shell containing them; consequently the diameter of the core increases, and the reactivity decreases.

The movement of the uranium carbide is a function of temperature and of the structural features of the core. Temperature is a function of heat generation, heat transfer, coolant dynamics, and the physical properties of the core materials. The amplitude of the power response to a signal is determined by the steady-state gain of the feedback, which in turn is dependent upon the relationship between reactivity and power change. The frequency-dependent time constants and time lags are functions of heat generation, coolant flow, the elastic properties of the materials, and the mechanical phenomena of the structure.

It is desired to know the values of $\Delta V/V$, $\Delta R/R$, $\Delta r/r$, and $\Delta Z/Z$ as a function of position in the core and of time-dependent temperature.

Figures 26 and 27 indicate a method of handling the space dependency, thus leaving time as the continuous independent variable in the equations for temperature and elastic phenomena.

The dynamics of a boiling fluid in parallel channels will be analyzed by the simulation of an analogous electrical system. The flow circuit is shown in Fig. 28.

Fig. 28. Schematic Diagram of Flow of Coolant through Reactor

The equations pertain to a single loop. The system consists of n parallel channels; therefore there are n degrees of freedom. This system can be simulated on an electronic analog and will consist of n resonant circuits in parallel.

Heat is added to the flowing fluid in the channels over a length Z . The vapor formed in the channels separates from the liquid in header A, is condensed in B and returned to header D by pump C. For any given heat input to the system, the pump C produces the desired ratio between fluid and vapor. The desired ratio is that which provides stable boiling conditions.

C. Mechanical Analysis

The mechanical analysis is concerned with the physical aspects of the core and the test unit, both of which must have the proper physical arrangement to produce and constrain the desired nuclear and thermodynamic phenomena. They must be constructed of materials suitable to the service for which they are intended, be controllable by reasonable means, and have structural integrity.

After the design objectives and physical boundaries have been set, the following work is required in the order listed:

1. Design the fuel element and cladding.
2. Design the core.
 - a. Fuel assembly.
 - b. Supporting structure.
 - c. Method of holding assembly together.
 - d. Means of providing instrumentation.
 - e. Loading and unloading features.
 - f. Meltdown characteristics.
 - g. Vibration characteristics.
 - h. Flow-channel shape.
 - i. Source and drive.
 - j. Oscillator and drive.
3. Design the mock-up of the fluid-flow circuits.
4. Design the desired feedback characteristics into the core. This requires close cooperation with all persons connected with the analysis.
5. Design the containment shell.
 - a. Perform stress analysis.
 - b. Investigate material compatibility.
 - c. Investigate fabrication techniques.
 - d. Investigate possible difficulties caused by variations in coefficient of thermal expansion.

6. Design the condenser.
 - a. Investigate best condenser arrangement (coil type vs vertical-tube type).
 - b. Perform stress analysis.
 - c. Investigate material compatibility.
 - d. Investigate fabrication techniques.
 - e. Design condenser supports.
7. Design a sodium-recirculating pump driven by an electric motor. Design a sodium-recirculating pump driven by a helium turbine. Ask for bids on each pump. Investigate the bearings and seals thoroughly.
8. Design the helium system.
 - a. Determine the thermodynamic conditions of the helium for the test unit when the sodium pump is driven by:
 - i. An electric motor.
 - ii. A helium turbine.
 - b. Determine the leakage rate of helium into the sodium space, and design a system for controlling the helium blanket above the sodium vapor.
 - c. Design the helium-to-air heat exchanger, and ask for bids.
 - d. Design the helium compressor, and ask for bids.
 - e. Determine whether it is more desirable to use helium or a vacuum in the double-wall construction, and design a control system for either case.
9. Determine if radiation shields are needed with the double-wall construction.
10. Check the capacity of the existing NaK system to determine if the system will provide NaK at the desired conditions in the test unit.
11. Design the instrument leads, the means of supporting them, and the means of bringing them in and out of the pressure shell.
12. Design the system for actuating the reflectors. This is to be done in conjunction with the dynamics analysis.

13. Check the system for actuating the cup and safety rods, and then coordinate the check with the system for actuating the reflectors.
14. Study the fuel-removal facilities at EBR-I, and if impossible to redesign to suit this fuel assembly, then design a new one.
15. Redesign the present cup at EBR-I to work with the reflector.
16. Check possible necessary building alterations.
17. Calculate the shutdown or emergency cooling conditions.

D. Dynamic Analysis

The dynamic analysis is dependent upon the work in all other categories and is in effect the coupling of all time-dependent phenomena in their correct phase relationships.

The work program is as follows:

1. Construct a block diagram of the system represented by the test unit.
2. Analyze the phenomenon in each block, and write the equations that describe this phenomenon. The blocks of equations are considered to be:
 - a. Space- and energy-dependent neutron-diffusion equations.
 - b. Space- and time-dependent neutron-kinetics equations.
 - c. Space- and time-dependent thermal equations.
 - d. Space- and time-dependent equations for the dynamics of a boiling fluid flowing in parallel channels between common headers.
 - e. Space- and time-dependent equations of the mechanical phenomena, which include the following subblocks:
 - (1) Thermal expansion.
 - (2) Elastic phenomena (energy, stress, and deformation).
 - (3) Acceleration and inertia of masses.
 - (4) Static friction, viscous friction, and viscous damping.
 - (5) Fluid-density variation.
 - (6) Metal-phase changes.

3. Couple the blocks of equations, making sure that boundary conditions and phasing are compatible.
4. Design and write the equations for the control system.
5. Write the equations for the worth and speed of the reflector, cup, and safety rods.
6. Construct the ideal mathematical model of the system.
7. Effect a compromise between the ideal model and the computer equipment available.
8. Transfer this model into electronic circuitry (using digital or analog machines where best suited).
9. Test each block separately; then couple all the blocks to form a closed-loop system. Testing blocks of equations is best done on the PACE analog, but the coupled blocks will be coded for the IBM-704.
10. Determine the dynamic characteristics of the system, and make any design changes that will improve its response to the signals it will receive in the service for which it is intended.
11. Make sure that the equipment for recording the power trace is of an accuracy commensurable with that required for a high-caliber analysis. This is important.

This program will require the cooperation of those responsible for the nuclear, thermodynamic, and mechanical equations, and it will form the basis for the Hazards Report and the Design Report.

VI. SUPPORTING RESEARCH AND DEVELOPMENT

A. General Discussion

The success of this investigation will depend upon good analysis, substantiated by research and development. A partial list of matters that should be investigated concurrently are listed in the order of their importance.

1. The characteristics of the material that will come into contact with the boiling-liquid metal are considered to be the most important question to be answered. On the basis of what is known today from the work of Pratt and Whitney, Battelle, Superior Tube, and Babcock and Wilcox, it appears that columbium with 1% zirconium has more promise than any of the other materials considered. It is suggested that this material be tested thoroughly at the conditions of operation. The metallurgical and fabrication characteristics of this material for the condition of this test unit should be thoroughly understood. Even though tests indicate that the material is satisfactory, the test unit and all piping in its closed loops will be double wall with provisions for leak-testing in the annulus. These precautions are considered essential at this stage in the development because of the serious consequence of a structural failure, which may allow high-temperature radioactive liquid metal to escape.

2. Fuel-element fabrication and tests are important. The most likely fuel materials are considered to be uranium oxide and uranium carbide.

3. Critical experiments on the cold core with and without the liquid metal is essential.

4. Detailed experimental studies on heat transfer and thermodynamics can be done with models.

5. A mathematical analysis and analog simulation of the dynamic system of the test unit should be made before construction so that the desired feedback characteristics can be designed into the unit with provision for adjusting them to take care of small inaccuracies in the analysis. The analog and test should be run concurrently to provide greater insight into the physical phenomena occurring in the core.

6. A full-scale test of the turbine-driven liquid-metal circulating pump should be made at the operating conditions. This test should be preceded by separate tests of bearings, shaft seals, turbine wheels, cavitation of impellers, and speed-control response.

7. The metal-vapor condenser and the helium heat exchanger should be tested at three times operating pressure with hot NaK. Since this could

be a spectacular test, it should be performed in a sealed container surrounded by an inert atmosphere.

8. The liquid level in the test unit must be known and controlled; therefore a level indicator must be developed.

9. Thermocouples, flux-measuring devices, and flowmeters must be developed.

10. If the boiling metal is to be used eventually in a direct-cycle turbine system, test programs should be initiated on bearings, shaft seals, and turbine wheels in a metal-vapor environment.

B. Critical Experiments

The critical experiments referred to in paragraph A3 above include the following items:

1. The principal uncertainties requiring clarification by experiments are those associated with the reflector control. The worth of the reflectors as a function of axial position will be measured. Radial and axial U^{235} fission distributions will also be determined for various axial reflector positions. It may be desirable to vary the reflector thickness.

2. The pressure vessel size will be fixed before critical experiments can be performed. The critical experiments will serve to fix an appropriate fuel concentration for the size core it is possible to build. The worth of complete columns of fuel added radially to the core will be determined, as well as the worth of axial slices of fuel at various heights and at the top and bottom of the core.

3. The axial effect of variation of liquid metal and the effect of reflector position on metal worth will be determined.

4. Capture-rate measurements will be made in the reflector to aid in the determination of the gamma heating rate.

5. Flux measurements will be made in a mock-up of the shield, particularly with the reflector away from the core.

C. Heat Transfer and Thermodynamics

Studies of heat transfer and thermodynamics referred to in paragraph A4 above include the following items:

1. The pumping system will be designed, mocked up, and tested.
2. Two-phase flow phenomena will require the following investigations:

a. Vapor-volume fractions for boiling sodium will be measured to determine the feedback due to density variation. Because of the effect of "voids" on reactivity, there must be adequate methods of predicting vapor-volume fractions in the reactor coolant channels. Because of the complexity of two-phase flow, these measurements must be made empirically.

b. Two-phase frictional pressure losses will be investigated to determine flow characteristics through the system. Frictional pressure losses for the reactor core will be determined accurately.

c. Burnout heat-transfer limits for boiling sodium will be investigated because one of the limits on core performance is the "burnout" limit. This peak or critical heat flux limits reactor power density. For optimum core design, this limit must be established.

d. Void variations in single and multiple channels should be investigated. One problem in a reactor cooled by a boiling alkali metal is that these systems must run at pressures at which the vapor specific volume is large. This means that the reactor core will have a large vapor-volume fraction. The vapor-volume fraction and pressure variations in single and multiple channels must be investigated, as the vapor-volume fraction at which the core can be run stably directly affects the amount of liquid that must be pumped through the core.

e. Investigation of flow distribution through the core is connected with the burnout studies. The distribution of coolant through the core will affect the vapor-volume fraction through the core, and the flow distribution through the core must be investigated.

D. Dynamics

In connection with the dynamics work referred to in paragraph A5 above, it is suggested that an analog of the closed loop, consisting of the reactor core, the forced-circulation pump, and the reactor controls be constructed to simulate the proposed design, and that the analog and the reactor be coordinated.

When the tests start it is almost certain that the analog will have to be modified to make the electronic system and the physical system actually similar. Hopefully these modifications will be only in the area of phasing.

The three main sets of equations in the system are the neutron kinetics, the fluid dynamics, and the mechanical.

The neutron-kinetics equations are well known and easy to represent.

The fluid-dynamics equations are complex and extensive. Recently, they have been written and checked against an actual physical system

represented by boiling a liquid in a closed-loop, pressurized system.⁴ This set of equations has a good representation of the actual phenomenon in the physical system. These equations, with their constants and boundary conditions changed, will be used to define the fluid-dynamic conditions in the reactor.

The mechanical equations are concerned with the reactivity feedbacks and the interaction of the reflector controls. Since this core is similar to that of EBR-I, Mark III, an analysis of the core should result in a similar set of equations. The equation defining the worth of the reflector and its effect upon feedback is a function of space and time and will require thorough analysis. The transfer function can be altered by physical changes to the pump or reflectors, preferably before the test unit is built.

This analog should be left intact until after the test results have been analyzed. The analog should be available when needed during this period.

VII. TEST PROGRAM

The intent of the program is to test components and systems under conditions that demonstrate the characteristics of both the physical phenomena and the materials under the conditions of operation.

The program consists of a mock-up of the entire test unit and out-of-system tests on samples, sections, and components.

The program lists the desired tests. Each test listed, and possibly some additional ones, will have to be worked up in detail with manpower requirements and costs.

The supporting test work is considered to be in the following categories:

- A. Mechanical
- B. Thermodynamic
- C. Nuclear
- D. Metallurgical
- E. Safety

The tests are intended to give immediate answers to questions arising out of this design. More generalized investigations will probably follow.

A. Mechanical Tests

1. Mock-up of the Fluid Dynamic System

When the design drawings are complete, exact duplicates of the shell, core, pump condenser, and jet will be constructed and assembled. These duplicates will probably be made of carbon steel and/or aluminum. The fluid will be water, a few degrees below saturation temperature. All the instrumentation will be installed on this mock-up primarily for the purpose of seeing how much trouble it is to install and maintain the instrumentation.

The procedures for loading and unloading the fuel elements will be developed on the mock-up.

2. Sodium Pump

The sodium pump is a helium turbo-driven pump unusual with respect to its fluid dynamics and the materials of construction. The pump will be tested under the actual conditions of operation.

This pump will be purchased after a research and development program, acceptance being based upon demonstration that performance is satisfactory for the desired conditions of operation.

The tests made under item 1 above will demonstrate that the design is correct so that the actual pump, constructed of columbium, can be fabricated with confidence that only minor alterations may be necessary to meet the performance.

The acceptance tests will be made with sodium at the operating conditions.

The sodium pump will be tested in the actual columbium pressure vessel with all flow and thermodynamic conditions exactly as they will be in the experimental program at EBR-I. Performance curves over the range of operation will be required.

3. Control Assembly

A mock-up of the reflectors, cup, and safety rods will be made of aluminum to check the speed and precision of the actuating mechanisms.

4. Oscillator Rod

A mock-up of the oscillator rod with its actual driving mechanism and recording instruments will be made. The actual sinusoidal input and power traces must be measured, and not some combination of unaccounted-for errors due to sloppiness in the electronic and mechanical components. Precision in this system is considered to be imperative.

5. Helium System

The helium system will be tested insofar as controlled leakage is desired.

6. Sodium-detecting System

A system for detecting sodium that leaks into the surrounding helium shell will be developed and tested.

7. Pressure-vessel Closures

The closure of the pressure vessel will be tested. The material and bolts will be of the actual material, and the fluid will be helium at the temperature and pressure of the operating conditions.

8. Hydrostatic Tests

Hydrostatic tests will be made on the finished pressure vessel and the condenser tubes by raising pressure from 75 to 225 psia in steps of 50 lb. The strain-gauge readings will be recorded on the surface of the cylinder, and the deflections measured. Welds will be X-rayed after each 50-lb step in pressure.

9. Structural Integrity Tests

The core assembly will be subjected to mechanical load and to vibration to determine how it will stand up under severe operating conditions or accidents.

The entire assembly will be subjected to vibration tests simulating what might occur if dynamic instability is reached during operation. This could occur because of the critical speed of the pump or because of fluid instability due to operating conditions.

B. Thermodynamic Tests

1. Heat-transfer Data

Construct experimental apparatus for determining the following characteristics at a temperature of 1100°C and a pressure of 100 psia:

- a. Heat-transfer rate for liquid sodium.
- b. Heat-transfer rate for boiling sodium.
- c. Heat-transfer rate for condensing sodium.
- d. Burnout heat flux for liquid and boiling conditions.
- e. Heat-transfer rate between niobium shell and stainless steel shell.

2. Pressure-loss Data

The pressure variation along a typical channel is determined experimentally using boiling sodium at operation conditions. A generalized study of two-phase flow in boiling liquid metal should be very extensive. This very specialized case is required for the immediate needs of the program.

C. Nuclear Tests

The nuclear tests are concerned with critical experiments consisting of two parts. The first part is concerned with the determination of the flux distribution in the cold critical mass. The second part is concerned with a check of the first part, the determination of the control system, and the determination of the feedbacks due to expansion, sodium, and Doppler effect. The second part has not been detailed.

The primary purpose of the initial experimental program is to establish a critical mass for the core. The design of the current core and the surrounding reflector is complex and novel enough so that theoretical calculations require a safe allowance for possible critical mass underestimation; at the same time, the dimensional restrictions are such that a "safe margin" cannot be provided. At least critical mass must therefore be determined before the core and fuel element designs are finalized.

Figure 29 is a cylindrically symmetric view of the core and surrounding reflectors. The volume fractions and total amount of materials

NOTE: ALL DIMENSIONS IN CENTIMETERS

Fig. 29. Zone Arrangement of Core

are listed on page 77. Because of the lack of proper materials or because of other practical restrictions, reproducing the shown configuration exactly might be difficult. Therefore, the features that cause the main theoretical difficulties and should be emphasized are outlined below in the order of their importance.

a. The Low-density Reflector (Zone II) Surrounding the Core and Separating it from the High-density Reflectors. This reflector allows neutron streaming, cuts off the softer spectrum flux diffusing back from the beryllium oxide reflector, and, in general, has a large influence on core reactivity. This influence is difficult to evaluate exactly since almost all available experimental criticals have immediately adjacent high-density reflectors and do not lend themselves as checks for the theoretical calculations. This reflector (Zone II) is composed of a number of concentric columbium and stainless-steel vessels with sodium, NaK, or void between the vessels. It would be preferable if this feature (allowing axial neutron streaming) could be mocked up roughly.

b. The Movable BeO Reflector (Zone III). Calculations have indicated that this reflector has a worth of $\sim 0.07 \Delta k/k$, which is made uncertain, however, by the large void spaces (Zone VII) above and below the reflector. Since beryllium oxide is not available, this reflector can be mocked up with pure beryllium. Experimental evidence has shown that the difference in reactivity caused by interchanging beryllium and beryllium oxide reflectors is slight.

c. Radial-core Dimensions and U^{235} Density. Both these parameters are practical maxima and cannot be exceeded.

d. Core Materials. The fuel in the core is fully enriched uranium carbide, and the structural material is columbium (27% by volume for a total of 40 kg). Since that much columbium is not available, area substitutions will have to be made. The overall worth of columbium for this core is expected to be negative and on the order of -0.05 to $-0.02 \Delta k/k$.

e. Axial Reflectors (Zone V). This is an extension of the core configuration where the uranium carbide has been replaced by beryllium oxide. Thus there are vertical channels for sodium flow which will enhance neutron streaming. This effect might be especially significant for the upper reflector, which immediately adjoins a large sodium plenum. The axial sodium channel feature should therefore be mocked up roughly.

Calculations have indicated that the shown configuration should be supercritical by 0.02 - $0.04 \Delta k/k$. If this is correct, the critical-mass determination will be simple and criticality can be approached either by keeping core dimensions constant and increasing U^{235} density (up to a maximum of 0.28% by volume) or by interchanging radially reflector

material by core material (up to a maximum radius of 14 cm). If the final configuration is reached and the core is not critical, the following steps can be taken:

- a. The length of the core can be increased to 38 cm.
- b. The length of the axial reflectors can be increased to 10 cm.
- c. The axial length of the radial (Zone III) reflector can be increased by 10 cm.

The compositions, weights, and volumes of the zones shown in Fig. 29 are listed below:

<u>Zone I: Core</u>		
	<u>Weight, kg</u>	<u>Volume, liters</u>
U ²³⁵	90.81	4.856
U ²³⁸	6.17	0.337
C	4.96	2.97
Nb	39.93	4.643
Na	2.61	3.677
Void		0.757
	<u>144.48</u>	<u>17.24</u>

<u>Zone II: Nb-Na-SS Reflector</u>		
	<u>Weight, kg</u>	<u>Volume, liters</u>
Nb	48.42	5.63
Na	27.54	38.79
SS	23.67	29.96
Void		21.91
	<u>99.63</u>	<u>96.65</u>

<u>Zone III: BeO Reflector</u>		
	<u>Weight, kg</u>	<u>Volume, liters</u>
Be	47.64	25.79
O	84.69	23.51
	<u>132.33</u>	<u>49.294</u>

<u>Zone IV: Heavy Blanket</u>		
	<u>Weight, kg</u>	<u>Volume, liters</u>
U	1544.6	82.6
SS	73.5	9.3
Void		8.1
	<u>1618</u>	<u>100.03</u>

<u>Zone V: End Reflectors</u>		
	<u>Weight, kg</u>	<u>Volume, liters</u>
Nb	13.95	1.622
Na	1.08	1.527
Be	2.88	1.559
O		2.008
	<u>17.91</u>	<u>6.157</u>

<u>Zone VI: Upper Na Volume</u>		
	<u>Weight, kg</u>	<u>Volume, liters</u>
Na	8.96	12.62

Zone VII: Void
Volume = 87.8 liters

<u>Zone VIII: Upper Radial Plenum</u>		
	<u>Weight, kg</u>	<u>Volume, liters</u>
SST	370.35	46.88
Void		20.09
	<u>370.35</u>	<u>66.97</u>

<u>Total Weight of Materials, kg</u>	
Enriched U:	96.98
Depleted U:	1544.6
SST:	483.4
Nb:	102.30
C:	4.96
Be:	50.52
O:	23.51
Na:	40.19

D. Metallurgical Tests

The metallurgical experimental program is concerned with materials compatibility under the condition of irradiation and strain expected in the test unit. The metallurgical tests have not been considered thoroughly. However, a partial list of tests is presented below.

1. Materials Compatibility

a. Investigate the effect of sodium in contact with niobium at temperatures up to 1100°C.

b. Construct a materials couple, and expose it to 1100°C. The purpose of this test is to determine what effects, if any, are encountered during the intimate contact of the columbium alloy and its protective sheath metal at 1100°C. For example, if a eutectic should form on contact between the alloy and a sheath of Inconel, we would want to know this so that it would not bias our test results.

c. Construct containment capsules, measure physical dimensions of refractory metal capsules, and take samples for metallographic examination. Load capsules with sodium under an inert atmosphere. Assemble complete units with protective outer capsule and seal. Place samples in furnaces, and heat. Immerse the lower ends of the capsules in the furnace, and cool the upper ends to permit refluxing of the sodium. Remove the capsules individually at successively longer periods of time for examination. Cool and disassemble capsules for examination. Measure physical dimensions, and observe failures. Clean out capsule, section, and examine for effects of corrosion. Make metallographic and chemical analyses.

2. In-pile Tests

a. Design in-pile loop for irradiation testing of fuel element.

b. Set up first loop in a mock-up area for test purposes. The heat would be supplied to the loop by electrical heating.

c. Test effectiveness of instrumentation selected for operation of the loop, and observe gross behavior of the mock-up.

d. Construct loop(s) capable of steady-state operation in CP-5.

e. Assemble equipment, and insert loop in CP-5 with fuel element in position for long-term irradiation.

f. Observe fuel under irradiation. Leave fuel in reactor for selected period of time until desired burnup is achieved.

g. Remove loop and element from reactor. Transport to cave for examination.

h. Remove fuel element from loop, and examine. Take physical dimensions, photograph, and section for analysis and metallography.

i. Prepare selected number of fuel elements for irradiation in MTR.

j. Remove elements from capsules. Remotely clean and examine fuel specimens. Place in specially designed furnace under inert atmosphere, and heat to 1100°C for about 4 hr to anneal elements. Allow to cool, and remove from furnace. Examine.

k. Allow activity of elements to decay to bring activity back down. Insert into TREAT, and expose to predetermined transient. Record data. Disassemble capsule in cave, and examine specimen. Summarize results.

3. Flowing-sodium Tests

a. Design loop for containment of fuel specimen. This loop would be similar to the one to be installed in CP-5.

b. Expose to TREAT transient. Collect data.

c. Disassemble loop, and examine condition of element. This last step may have to be done in a dry box under an inert atmosphere due to the pyrophoricity of uranium carbide and sodium mixture.

E. Safety Tests

Safety-oriented experiments on the EBR-I conversion program should prove valuable, both in leading to increased understanding of the safety aspects of the system, and in providing guidance for development of advanced versions of the system design. Roughly, one may divide such experiments into tests on the system at or near conditions of design operation, and tests on severe "accident" conditions. Tests of the first type have been outlined previously, for example in Sowa's memo of January 31, 1962.¹⁶ Both types of experiment place severe demands upon instrumentation.

Safety is concerned with experiments of the second type and instruments for such experiments. Instrumentation for these tests overlap that desired for normal operation, and safety experiments in general offer a means for testing the progress in instrument development.

The following list of experiments forms a basis for setting up a safety-related program of tests on the proposed system. Since this system represents a new area of work, the list presents a scope of effort, and the program adopted should be flexible enough to include experiments based on results that develop in the course of the testing.

1. Dry Transient Experiments in TREAT

Aims of these experiments would include definition of temperature limits on fuel and on acceptable amounts of fuel-cladding reaction. In addition, element failure and fuel movement mechanisms would be studied. This phase would include, but probably not be limited to, about six to ten dry opaque tests and about eight to 12 dry transparent tests. Experiment analysis would be primarily based on postmortem inspections, high-temperature fast-response cladding thermocouple records, internal high-temperature thermocouple records, and high-speed color photographic data. Two or three dry clusters might be exposed in TREAT, although it would seem that the most interesting experiments on gross fuel movement would be performed with coolant. With minor changes, existing equipment designs could be used.

2. Single-element Transient Experiments with Flowing Coolant

A few, perhaps four to eight, single-element tests with flowing coolant would provide an opportunity to study coolant-expulsion characteristics, fuel failure associated with coolant expulsion, and fuel movement in the coolant-present condition. These tests probably should be specified only after out-of-pile loop and relatively long-term experiments in CP-5 had provided some data on coolant behavior. Analysis would include post-mortem inspections, coolant-thermocouple record, coolant-pressure records, and internal-element thermocouple records. Equipment might well be designed on the basis of facilities now under development. However, with the exception of coffins and auxiliary circuits, which might be used with little or no changes, it appears that major design changes are required. Parenthetically, for purposes of orientation, rough flux-depression calculations show that a single 3% enriched EBR-II element could be given an energy input up to ~150 kW-sec in TREAT using the package sodium loop now being developed.

3. Small Clusters in Flowing Coolant

These experiments would be done to mock up more fully flow and temperature distributions.

Although these experiments would not represent an exhaustive program, they would yield significant data. Further, once given equipment,

samples, instrumentation, and actual operating experience, the above program could be expanded relatively easily. The actual reactor time for the above experiments is expected to be modest, and capable of being fitted into the TREAT schedule without disrupting other programs. (Note that the experiments may be run in conjunction with other, similar experiments, thus minimizing reactor time demands.) Supplementing these experiments, there should also be a comparable set of tests on preirradiated elements.

4. Instrumentation

Instrumentation jobs* for the experimentation would thus be:

- a. High-temperature thermocouples for fuel and/or cladding temperature measurements.
- b. High-temperature thermocouples for coolant temperature measurements.
- c. Pressure-measuring techniques, steady-state and transient pressures under high-temperature conditions.
- d. Coolant flow measurement.

Also of interest, but of lower priority would be investigations into methods of determining:

- e. Material movements.
- f. Coolant voids.

*These should be interpreted as being in addition to those associated with the conversion, although there is some overlap.

VIII. COST ESTIMATE

The cost of fabricating and testing is estimated and presented in three parts. These are the cost of the fabricated components, the cost of assembling these components into a test unit, and the cost of supporting research and development work, preliminary to the start of testing.

A. Cost of Fabricated Components

Containing shell	\$210,000	
Core and fuel	100,000	
Reflector and mechanism	80,000	
Condenser	60,000	
Oscillator mechanism	15,000	
Recirculating pump	155,000	
Source-drive mechanism	15,000	
Heat exchanger	5,000	
Compressor	25,000	
Instruments	75,000	
Shielding change	20,000	
Fuel-handling equipment	35,000	
	<u>\$795,000</u>	\$ 795,000

B. Cost of Assembling Test Unit (Labor)

Cost of assembling test unit	\$850,000	\$ 850,000
------------------------------	-----------	------------

C. Cost of Research and Development

In-core tests of fuel elements	\$100,000	
Fabrication techniques	25,000	
Turbopump testing	50,000	
Instrument development	75,000	
Heat-transfer tests	50,000	
Materials testing	100,000	
Preliminary dynamic analysis	200,000	
	<u>\$600,000</u>	\$ 600,000
		<u>\$2,245,000</u>
Contingency of 20%		449,000
		<u>\$2,694,000</u>

ACKNOWLEDGMENT

The substantial assistance of M. Novick, R. P. Carter, R. O. Haroldsen, D. F. McGinnis, A. E. McCarthy, and K. Almenas is largely responsible for the context of this report.

REFERENCES

1. Adams, H. E., *Pumping Boiling Fluids When the Available Static Head Is Zero or Less*, ASME Paper 63 WA-220 (Aug 30, 1963).
2. Hansen, G. E., and Roach, W. H., *Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies*, LAMS-2543, Los Alamos Scientific Laboratory (Nov 1961).
3. Lottes, P. A., *Nuclear Reactor Heat Transfer*, ANL-6469, p. 88 (Dec 1961).
4. Anderson, R. P., Bryant, L. T., Carter, J. C., and Marchaterre, J. F., *Transient Analysis of Two-phase Natural-circulation Systems*, ANL-6653 (Dec 1962).
5. *Liquid Metals Handbook Sodium-NaK Supplement Third Edition*, TID-5277, Atomic Energy Commission and Department of the Navy, Washington, D. C. (July 1955).
6. *International Developments in Heat Transfer*, p. 527, Part III, International Heat Transfer Conference August 28-September 1, 1961, Boulder, Colorado, USA, published by ASME, 29 W. 39th St., New York 18, N. Y.
7. Weatherford, W. D., Jr., Tyler, John C., and Ku, P. M., *Properties of Inorganic Energy Conversion and Heat Transfer Fluids for Space Applications*, WADD Technical Report 61-96 (Nov 1961).
8. Noyes, R. C., *An Experimental Study of Sodium Pool Boiling Heat Transfer*, NAA-SR-6769, Atomics International, Canoga Park, Calif. (Dec 1960).
9. McAdams, W. H., *Heat Transmission*, 2nd Ed., McGraw-Hill Company, Inc., New York, London (1942).
10. Lichtenberger, H. V., Novick, M., Cerutti, B. C., Cameron, R. A., McGinnis, D. F., Pettit, E. N., Whitham, G. K., Haroldsen, R. A., and Koch, L. J., *Progress Report on the Experimental Breeder Reactor, April 1, 1951 through January 31, 1953*, ANL-5023 (Feb 20, 1953).
11. Lichtenberger, H. V., *Experimental Breeder Project, Report for the Period March 1, 1949 through January 31, 1950*, ANL-4420 (March 13, 1950).
12. Lichtenberger, H. V., *Experimental Breeder Project Report for the Period February 1, 1950 through March 31, 1951*, ANL-4554 (April 1, 1951).
13. Zinn, W. H., Lichtenberger, H. V., and Novick, M., *Fast Neutron Pile for a Test of Conversion; Feasibility Report*, ANL-4356 (Oct 14, 1949).
14. Tipton, C. R., Jr., edited by, Battelle Memorial Institute, *Reactor Engineering Handbook, Second Edition, Vol. I Materials*, Interscience Publishers, Inc., New York (1964).
15. Grotenhuis, M., *Lecture Notes on Reactor Shielding*, ANL-6000 (March 1959).
16. Sowa, E. S., private communication (Jan 31, 1962).