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THE PHYSICS DESIGN OF THE EBR-II

by

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PREFACE

The physics design problems of the EBR-II are summarized. These include analysis of the EBR-II engineering design as well as applicable zero-power critical experiments. Pertinent reactor safety problems are reviewed. Safety considerations bearing on normal plant operation and manipulations within the reactor are emphasized. The implication of controlled in-pile meltdown experiments is considered. Irradiation damage and metallurgical phase phenomena are summarized and related to reactivity. The nuclear performance of the system is considered in terms of actual plant operation. The predicted shift of both power and reactivity from core to radial reflector is described.

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I. INTRODUCTION

The Argonne Experimental Breeder Reactor II (EBR-II) is an unmoderated, heterogeneous, sodium-cooled reactor designed to produce 62.5 megawatts (Mw) of heat.⁽¹⁻³⁾ The system complex is primarily an engineering facility to determine the feasibility of this type of reactor for central station power plant application.

The design emphasis upon high thermal performance at high temperature with optimum burnup in both reactor core and radial breeder reflector has placed severe requirements on detailed knowledge of the physics of the reactor. To this end, the neutronics of the EBR-II and similar systems have been extensively investigated.^(4,5) The correlation between the large amount of experimental data and theoretical prediction suggests that the static as well as many aspects of the kinetic behavior are generally understood. It is recognized that certain detailed reactivity and spectrum effects cannot be properly predicted. It is not always obvious whether theory or experiment or both are in error when certain measured results cannot be properly predicted.

Reactor physics calculations were originally initiated with few- and multigroup diffusion theory, and with S_N analyses in one-dimensional geometry.⁽⁶⁾ Analyses by few- and multigroup two-dimensional diffusion theory have been performed in both r, z , and r, θ geometry. In addition, space-independent static and kinetic analysis have been freely used.⁽³⁾

The physics design of the EBR-II, or of any other reactor, requires close liaison with and an understanding of many of the engineering aspects of the reactor system. For example, an error of 2% for the density of the fuel alloy appears as a 1% uncertainty in the reactivity. In loading the reactor to criticality, a 2% uncertainty in density of the fuel alloy alone suggests an uncertainty of at least 4% for the critical mass. The overall requirement for precision with respect to composition cannot be over-emphasized for good reactor design.

The critical experiment is performed principally to reduce uncertainties and thus to reduce the amount of flexibility that must be built into the reactor. Flexibility in a critical assembly is an admirable feature. In a power-producing reactor such flexibility is neither required nor

desirable, since it can lead to accident-prone situations which may not be readily recognized. Three distinct types of critical experiments have been performed with the ANL ZPR-III facility which bear on the neutronics of the EBR-II:^(5,7)

1. Fundamental Experiments

These systems consist of single-region, EBR-II-sized cores surrounded by a reflector of uniform high or low density.

2. Engineering Mockup

This system is designed to represent the best geometric mock-up with the composition of the actual reactor that may be assembled on the critical facility.

3. Clean Mockup

This system does not include detailed asymmetries of the reactor design due to geometrical arrangements or composition. This mockup is of more general interest than the Engineering Mockup. It can be included in the class of Fundamental Experiments by lifting the restriction about reflector uniformity.

All three types of experiments have been used to specify parameters for the design of the EBR-II. A study of the results obtained from such experiments has shown that, even with the detailed Engineering Mockup and its peripheral reactivity measurements, analysis is required to effect a transition from this experiment to the actual reactor. The confidence in the analysis of transition is based upon analysis of the Fundamental and Clean Mockup Experiments. These lend themselves to basic analytical investigations. Thus it might appear that all three types of experiments are necessary for a good understanding of the physics of a specific reactor design. This may not be necessary as our understanding is broadened by time and experience.

The physics of an actual reactor, such as the EBR-II, requires considerable attention to problems not faced during the general conceptual study. The precise determination of reactivity held by control mechanisms, the assurance that neutron-flux-monitoring equipment is adequate over all power ranges, and the assurance that neutron source flux is adequate for reactor startup are examples of problems peculiar to an actual reactor design. Another interesting problem requiring some precise understanding is the gradual shift of both power and reactivity from the reactor core to the radial breeder blanket. This effect comes about if fuel elements are more frequently cycled (replaced) than the inner portions of the radial blanket.

The EBR-II reactor program will be initiated with the performance of the Dry Critical Experiment.⁽⁸⁾ This experimental program with no sodium in or near the reactor core (other than that used for bonding in

the fuel elements) will give insight on both the incremental and total reactivity worth of the sodium coolant in the reactor. This information will be obtained by comparing Dry Critical data with those from the Wet Critical Experiments to be performed after coolant sodium has been introduced into the reactor. Subsequent to the Wet Critical Experiments, the reactor power will be gradually increased. The approach to power will include stability and kinetic investigations which will in fact determine the details of raising reactor power level.

II. DESCRIPTION OF THE EBR-II

The EBR-II power plant complex (see Fig. 1) is located at the USAEC National Reactor Testing Station. The plant includes a complete, remote, fuel-processing and fuel-element-fabrication facility. The EBR-II will be the first reactor in the United States Power Reactor Demonstration Program to operate on a closed fuel cycle. Partly spent or burned fuel can be pyrometallurgically reprocessed, re-enriched, and returned to the reactor after being refabricated.

Fig. 1

EBR-II Facility

The reactor is submerged in the "Primary Tank" (see Fig. 2), containing about 90,000 gal of liquid sodium at 370°C. With a maximum rate of coolant flow of about 9000 gpm, the large sodium reservoir insures that temperature transients in the bulk coolant are very slowly transmitted to the reactor. The large sodium reservoir also serves as a partial sink in case of an accidental energy release. The Primary Tank is suspended inside an airtight containment vessel (Figs. 2 and 3) which serves to confine an accidental release of built-up fission products, plutonium, and activated sodium from the primary system.

Fig. 2
EBR-II Reactor Plant

The structure of the primary system is designed to contain the energy release associated with a nuclear accident. The reactor building is designed to confine the effects of a maximum sodium-air interaction caused by a major sodium release. In a sense, the reactor is doubly contained.

The reactor is located in the reactor vessel near the bottom of the primary tank (see Fig. 3). Coolant is taken from the bulk sodium in the primary tank, passed through two 5000-gpm mechanical pumps, and is introduced near the bottom of the reactor vessel. Flow is then upward through individual fuel and breeder-reflector subassemblies. In order to achieve nearly uniform temperatures of the coolant at the outlet, the coolant flow is orificed consistent with axial and radial power density gradients and discontinuities. The hot (482°C) coolant leaves the reactor near the top of the reactor vessel and then passes through the primary heat exchanger

submerged in the primary sodium. Sodium is the working fluid of the intermediate secondary cooling system.

Fig. 3

EBR-II Primary System

Details of the reactor, the surrounding reactor vessel, and the neutron shield are given in Figs. 4 and 5. The reactor (core and breeder reflector) is a hexagonal array of subassemblies (see Fig. 5). The subassemblies are mechanically designed to prevent inadvertent interchange between enriched and depleted uranium-bearing subassemblies. Each subassembly contains the fissile or fertile material in the form of cylindrical fuel elements bonded with sodium and clad with stainless steel. The fertile material is unalloyed depleted uranium. The fissile material is contained in uranium-5 w/o fissium fuel alloy. The uranium is 48.4% enriched in U^{235} . Each fuel subassembly contains 91 fuel elements. Each fuel element contains ~67 gm of the uranium-5 w/o fissium alloy.

Fig 4
EBR-II Reactor (Vertical Section)

Fig. 5a

EBR-II Reactor (Horizontal Section)

Fig. 5b

Subassembly Locating Diagram

Reactor control is effected by moving fuel into or out of the core from below. There are twelve control and two safety subassemblies. Each of these contain 61 of the same fuel elements used in a normal fuel subassembly. Control and safety rod strokes are 14 in.; this is approximately the length of the core. In the most reactive position, fuel elements in a control rod are at the same elevation as fuel elements in a fuel subassembly. In the least reactive position, the tops of the fuel elements in a control rod are at the same elevation as the bottoms of the fuel elements in a fuel subassembly. The twelve control subassemblies are activated from the top and must be disconnected, in their least reactive positions, while fuel is being loaded (see Fig. 6). The two safety rods are actuated from below the core (Fig. 4) and are operative while fuel is being loaded. Scram signals actuate these rods depending on the "mode of operation." During "Reactor Operation," when the reactor is critical and producing power, an automatic scram signal will release the twelve control rods. With a pressure assist they move to their least reactive positions. During "Fuel Handling," when the reactor is expected to be substantially subcritical, an automatic scram signal releases the safety rods. The force of gravity causes them to move to their least reactive positions below the core. The safety rods may also be manually released during "Reactor Operation."

Fig. 6

EBR-II Fuel Handling System

The neutron-flux-monitoring equipment (fission counters and ion chambers) are located in eight air-cooled (45°C) instrument thimbles. Four of these are embedded in the neutron shield surrounding the reactor vessel (see Fig. 7). Four more are located in sodium just outside of the neutron shield. The fission counters and ion chambers are located near the central plane of the core for maximum sensitivity, but can be moved vertically inside the thimbles. The neutron shield consists mainly of graphite and borated graphite canned in stainless steel. Small amounts of sodium pass between the square shield cans. Borated graphite is used only in those shield regions where it does not interfere with instrument response.

Fig. 7

Nuclear Instrument Thimbles

There is no borated graphite inside the reactor vessel. In the neighborhood of the startup channels (see J_1 and J_2 thimbles in Figs. 5 and 7), there is no graphite inside the reactor vessel. There are eleven channels of nuclear instrumentation distributed throughout the eight instrument thimbles. Three log count data channels are operative from source power to about 2 kw. Three log flux and one linear flux channels are operative from 100 watts to full power. Three linear flux channels are operative from 60 kw to full power. One channel is operative from 6 Mw to full power and is to be used for automatic control of the reactor at power. Automatic control is not operative until the power level has been manually established.

An Sb-Be neutron source subassembly (see Fig. 5b) will be permanently located in the inner blanket of the reactor. Source activity of 100-200 curies is required for reactor startup. The Sb-Be source may be remotely disassembled by reactor fuel-handling equipment. The active Sb rod may be placed in a source shield subassembly located in the outer blanket (see Fig. 5b).

In-core instrumentation (four fission counters) is provided for the Dry Critical Experiment. If necessary, two such counters can be provided for the Wet Critical Experiments. During the Dry Critical Experiments, the in-core instrumentation will be permanently installed in the inner blanket (see Fig. 5b). The wet critical in-core instrumentation will be located in a control rod thimble.

III. EBR-II CRITICAL EXPERIMENTS

The EBR-II critical experiments with ZPR-III^(5,7) were designed to provide detailed information concerning the neutronics of the reactor. These experiments were done with mockups of the EBR-II reactor configuration simulating the normal power loading (with sodium coolant). In addition, a large number of experimental investigations were made for systems similar to the EBR-II.⁽⁵⁾ The neutronics of the EBR-II are well understood on the basis of analysis which has been well correlated with a large amount of experimental data.^(4,5)

The ZPR-III critical experiments utilized low-density aluminum for sodium in the mockup of the coolant volume of the reactor system. Gross substitution experiments with both solid sodium and aluminum of different effective densities confirmed that removal of coolant material from the reactor core and reflector regions resulted in a loss of reactivity.

A. Fundamental Experiments

Fifteen distinct critical experiments have been reported⁽⁵⁾ for EBR-II-sized cores. These experiments yielded information about reactivity effects caused by changes of core and reflector geometry, and composition. Spectral measurements, neutron flux traverses, material replacement experiments, and neutron lifetime determinations were included. These experiments were instrumental in verifying the analytical techniques applicable to EBR-II physics.

All of the experimental cores contained 14 v/o U^{235} , 16 v/o U^{238} , 12 v/o stainless steel, and 31 v/o aluminum, a composition very similar to that of the EBR-II mockup. These experiments included cylindrical, rectangular, and annular cores, with both high- and low-density reflectors.

B. Engineering Mockup

The detailed engineering mockup experiments represent the best version, with respect to geometry and composition, of the EBR-II that can be assembled on the ZPR-III. The radial properties of the EBR-II were assembled as shown in Fig. 8, which represents a cross section of the core at the central plane of the ZPR-III. The dotted hexagonal outline shows the fifth row of the reactor core superimposed on the square ZPR-III matrix. The closed hexagons indicate the control rod locations. The normal assembly matrix box (No. 1) contains 3 columns of enriched uranium, 3 columns of depleted uranium, 2 columns of stainless steel, and 8 columns of low-density aluminum. The axial properties of the mockup core are shown in Fig. 9, which represents a cross section of the plane including the central axis of symmetry. This plane is symbolic in that it passes through the safety rods, but does not show the control rods. The composition of the various core and reflector regions is given in Table I

Fig. 8

EBR-II Mockup on ZPR-III (Radial Section at Core Midplane)

Fig. 9

EBR-II Mockup on ZPR-III (Axial Section through Core Centerline)

The critical mass of the reactor configuration, which most closely resembles the composition and geometry of the EBR-II reactor, was measured to be 165 kg of U^{235} . This critical configuration can be and was predicted to within a few kg of U^{235} (calculated $k_{eff} = 1.015$).

The average reactivity worth of a reactor control rod was measured to be $0.37\% \Delta k/k$. The reactivity worth of two safety subassemblies was measured to be $1.36\% \Delta k/k$. The reactivity worth of a core subassembly, relative to low-density aluminum (simulating the displacement of sodium), is a function of radial position in the core. These measurements are reported in Table II. The reactivity worths, of inner blanket subassemblies relative to low-density aluminum are also given in Table II.

Table I
EBR-II AND EBR-II MOCKUP ON ZPR-III
(Compositions and Dimensions)

Region	Identification on Fig. 9	Mockup on ZPR-III						Reactor (61-Subassembly Core)				
		Composition (v/o)				Dimensions (in.)		Composition (v/o)			Dimensions (in.)	
		U ²³⁵	U ²³⁸	Stainless Steel	Al	Effective Radius	Length	U ²³⁵ + U ²³⁸	Stainless Steel	Na	Effective Radius	Length
Core	C-1	13.93	15.84	19.4	26.47	9.73	14.08	29.3	18.5	49.5	9.52	14.22
Control or Safety Rod	C-2	9.35	10.55	19.4	31.05	-	14.08	19.6	20.3	58.3	-	14.22
Structural Gap Regions	G-1	-	-	30.0	27.0	9.73	3.0	-	29.8	53.3	9.52	3.50
	G-2	-	-	56.3	9.0	9.73	1.0	-	52.2	47.8	9.52	0.87
	G-3	-	-	30.0	27.0	-	3.0	-	29.4	59.5	-	3.09
	G-4	-	-	40.5	22.3	9.73	2.0	-	39.5	60.5	9.52	2.06
	G-5	-	-	24.6	29.0	9.73	2.0	-	22.4	54.4	9.52	1.88
	G-6	-	-	35.1	24.6	-	2.0	-	34.8	65.2	-	1.63
Axial Blanket	B-1	0.06	29.9	19.5	17.9	9.91	19	30.3	18.5	51.2	9.52	18
Radial Blanket	B-2	0.13	61.26	19.6	6.43	15.5	30	60	20.9	19.1	30.75	55.0
	B-3	0.13	59.5	7.31	3.70	31.70	56-58					
	B-4	0.13	55.2	14.3	6.74	21.68	30					
Control and Safety Rod Followers	S-1	-	-	12.3	35.4	-	12.0	-	12.2	87.8	-	12.66
	S-2	-	-	49.1	24.8	-	8.0	-	50.1	49.9	-	6.62
	S-3	-	-	72.1	8.94	-	6.0	-	73.1	26.9	-	20.31
	S-4	-	-	69.8	13.4	-	15.0	-	73.1	26.9	-	

A number of detailed experiments were performed to determine the reactivity worths of various materials uniformly distributed throughout the reactor core and reflector regions. The importance of these is to effect a transition from the "aluminum-cooled" ZPR-III to the sodium-cooled reactor. Results instrumental in effecting this transition are given in Table III.

Table II

REACTIVITY WORTHS OF CORE AND BLANKET
SUBASSEMBLIES AS A FUNCTION
OF POSITION

(ZPR-III Mockup Experiments)

Reactor Row Number (Fig. 5b)	Type of Uranium	Reactivity Worth (% $\Delta k/k$)
1	Enriched	1.53
2	Enriched	1.48
3	Enriched	1.31
4	Enriched	1.05
5	Enriched	0.77
6	Depleted	0.18
7	Depleted	0.04

Table III

REGIONAL MATERIAL REPLACEMENT EXPERIMENTS

$[\%(\Delta k/k)/\text{kg of Material} \times 10^2]$

Material	Reactor Region*			
	Core	Axial Structural Gaps**	Reflectors	
			Radial	Axial
Aluminum	7.4	5.8	3.7	1.6
Stainless Steel	2.3	2.7	2.0	0.8
Sodium	11.6	6.4	5.8	3.7
U ²³⁵	33.5	-	-	-
U ²³⁸	1.5	-	-	-

*Core contains enriched uranium (C-i on Fig. 9)

Reflector regions contain depleted uranium (B-i on Fig. 9)

Structural gaps contain no uranium (G-i on Fig. 9)

**The axial structural gaps are located between the core and the axial reflector.

A number of distributed material replacements in the reactor core are instrumental in accounting for the reactivity effect of the 5 w/o noble metal fission products in the reprocessed fissium fuel alloy. These are given on Table IV. Measurements on synthesized fission product mockups as well as other materials of interest to fast reactor design are also given.

Table IV
MATERIAL REPLACEMENT EXPERIMENTS
(Distributed throughout EBR-II Core)

Material	Reactivity, $\rho(\Delta k/k) \times 10^2/\text{kg}$	Material	Reactivity, $\rho(\Delta k/k) \times 10^2/\text{kg}$
Molybdenum	+2.4	Physicum II*	-2.5
Ruthenium	-0.2	V	+5.9
Rhodium	-5.6	Nb	+0.29
Palladium	-2.0	Al ₂ O ₃	+13
Physicum I*	-3.3		

*Synthesized fission product mockups (see Ref. 5)

C. Neutron Source and Instrument Thimble Experiments

Special experiments dealt with the attenuation and spectral shift of neutron flux from the center of the core to the "J" and "O" thimbles (see Fig. 7) where the normal fission counters and ion chambers are located. The "J" and "O" thimbles are located about 58 and 78 in., respectively, from the reactor centerline in the central core plane. The unperturbed neutron flux can be estimated within and outside of the graphite neutron shield under the postulate of no thimble and uniform shield. The presence of the thimbles (up to 15 in. in OD) appreciably complicates such an estimate and reduces the accuracy of the predicted result. The other aspect of these investigations dealt with determining the neutron yield of an Sb-Be neutron source of the type to be employed in the EBR-II. Estimates of neutron yield are extremely uncertain (see Section IV-F). These experiments were designed to yield data relevant to:

1. the required Sb activity in the Sb-Be neutron source;
2. the geometric arrangement and the composition outside of the radial blanket and around the "J" thimbles to optimize low-power fission counter response.

The first experiment dealt with the early design of the neutron shield and reactor vessel.⁽³⁾ In the second series of experiments, several geometrical arrangements and compositions adjacent to the "J" thimbles in the final reactor configuration were investigated. The early shield configuration is shown in Fig. 17 of Ref. 3, whereas the final radial neutron shield for EBR-II is shown in Fig. 5a. The early neutron shield was completely outside of the reactor vessel which surrounds the radial outer

blanket. The final diameter of the reactor vessel is larger, and thus the stainless steel reactor vessel and its thermal baffling plates are almost in contact with the "J" thimble. Furthermore, some shielding material has been located between the reactor vessel and the outer boundary of the radial breeder reflector.

An absolute fission counter⁽⁹⁾ was placed at the core center and another at the radial core boundary during each investigation. Additional counters were placed throughout the radial blanket and neutron shield. However, the information of primary interest is that of low-power fission-counter response for the "J" thimble. To enhance instrument response, moderating material was placed in the interior of the instrument thimbles immediately adjacent to the fission counters. This was accomplished by inserting 4.1 kg ZrH in the form of 10-in.-long aluminum cans having a cross-sectional area of 9.8 square in. Other experiments included the removal of sizeable sections of normal shield material (graphite) outside of the "J" thimbles. The removed material was replaced by a better moderator (ZrH). Table V summarizes experimental fission-counter response data per watt of reactor power.

The neutron shield configuration adjacent to the "J" thimble was the object of subcritical investigations. With a 10-curie Sb-Be neutron source located at the reactor core inner blanket interface, subcritical count rates were obtained in the mockup "J" thimbles, as well as at the core-inner blanket interface. These results are also given in Table V. The experiments demonstrated that the spatial distribution of neutron flux is essentially invariant between a multiplication of 20 and a critical reactor.

These neutron-flux-attenuation experiments served to specify the details of the final neutron shield configuration for the reactor. Two "J" thimbles, containing the low-power fission counters, are immersed in a graphite Outer Shield with no graphite in the Inner Shield. Sodium is the major constituent of the inner shield region near these thimbles. All other portions of the inner shield contain graphite. These experiments showed that a 100-200-curie Sb-Be source will supply 3-6 unmultiplied counts per second in the "J" thimble startup fission counters. More sensitive in-core instrumentation is provided for the Dry Critical Experiments and is available for the Wet Critical Experiments.

Table V

SUMMARY OF EBR-II THIMBLE EXPERIMENTS

Shield Configuration	Final	Final	Final	Final	Final	Final	Final	Final	Early	Early
Thimble	J	J	J	J	J	J	J	J	J*	0
Principal Material in Region (see Figs. 4 and 5)										
Inner Shield:	Al	Al	Al	Al	Al	Al	C	C	Steel, Al, C	
Outer Shield:	ZrH	ZrH	C	C	C	C	C	C	C	
Interior of Thimble:	void	ZrH	void	ZrH	void	ZrH	void	ZrH	void	
"Bulk Sodium"	Al	Al	Al	Al	C	C	Al	Al	Al	
Count rate in thimble per watt of reactor power (cps/watt)	857	1345	731	2017	750	1841	395	826	1370	43
Cadmium Ratio in Thimble	4.5	8.8	1.9	5.5	1.9	5.2	2.4	6.9	7.6	~ 5
Subcritical Count Rate with Sb-Be Source (~10 curies)										
$k_{\text{eff}} \approx 0.97$	12.6	-	10.7	-	11.4	-	5.8	-	~26	-
$k_{\text{eff}} \approx 0.94$	6.3	-	5.8	-	5.6	-	2.9	-	~13	-

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*8-in. outer diameter; Final "J" Thimble Diameter is 15 in. OD.

D. Power Distribution

Uncertainties in the predicted power distribution may be estimated by comparing predicted with measured fission distributions from ZPR-III. Predicted fission distributions

$$F(r,z) \equiv \int_0^{\infty} \sigma_f(E) \phi(E,r,z) dE \quad ,$$

where σ_f is the fission cross section, ϕ the neutron flux, and E the energy, are obtained from a multigroup two-dimensional (r,z) analysis. This calculation, using the eleven-group constants from Ref. 4, will be described in a subsequent section. Figures 10 and 11 compare experimental with theoretical radial and axial fission distributions, respectively. Since these experimental data are insufficient for an absolute power calibration, they are normalized to the predicted U^{235} fission density at the center of the core.

The slight dip at $r \approx 11$ cm for the experimental U^{238} fission distribution in Fig. 10 is attributed to the presence of the safety rods. These were not properly described in the two-dimensional (r,z) analysis. Since mostly U^{238} fissions occur in the portions of the radial blanket shown in Fig. 10, it can be seen that dominant radial reflector fissions (U^{238}) are relatively well predicted. The data points in the outer portions of the radial reflector exhibit large uncertainties because of low power and thus of low-flux intensity in ZPR-III.

The rather striking deviations between theory and experiment in the gap regions indicated in Fig. 11 must be attributed to an inadequate understanding of the neutron cross sections of aluminum and stainless steel. However, since very few fissions ($< 2\%$) occur in the axial reflectors, these deviations between theory and experiment are not significant in determining the reactor power distributions.

E. Other Experiments

General experiments dealing with inhomogeneity, neutron spectrum and lifetime, and material replacement effects were included in the program.^(5,7) These will not be described in this report. The EBR-II mockup experiments cited in the previous sections are most significant for the reference engineering design. Additional investigations will be described here. These were designed to give insight about details of providing experimental apparatus for reactor operation.

Fig. 11

Axial Fission Distributions at Three Radial Positions

Fig. 10

Radial Fission Distributions at Core Midplane

The control mechanisms were studied in detail. Six of the control rods are located on the corners and six are located on the "flats" of the hexagonal core (see Fig. 5). The former are thus at a slightly larger distance from the reactor centerline and hold about 81% of the reactivity of the latter. (The previously cited control rod reactivity is an average.) Subsequent experiments showed that a control rod located on the corner of the hexagonal core requires the equivalent of 75 fuel elements (normal loading is 61 fuel elements) for constant reactivity worth of all twelve control rods.

The reactivity worth of a single control rod may be increased by inserting a strongly absorbing material in the sodium void follower region of the control subassembly. This region is adjacent to the fuel in the fuel subassemblies when the control rod is in its least reactive (down) position. Distributing about 1.8 kg of Ta in the void follower enhances the total worth of the control rod by less than 0.05% $\Delta k/k$. Distributing about 340 gm $B_4^{10}C$ (~ 91 a/o B^{10} in B) in the void follower was found to double the total reactivity worth of the normal control rod. If adopted, the reactivity worth of a single control rod could exceed one dollar ($\Delta k/k \approx 0.73\%$). Special control rods utilizing about 250 gm $B_4^{10}C$ are being designed as part of the EBR-II experimental program.

A rotating oscillator rod was calibrated. The mechanism is designed to replace a control subassembly. Any rotary motion is confined within a hexagonal subassembly (2.3 in. across flats). An eccentrically located $B_4^{10}C$ rod can rotate on a circle of about 2-in. diameter. A 14-in.-long rod containing 230 gm $B_4^{10}C$ was found to have a reactivity amplitude of 0.034% $\Delta k/k$. This is about the maximum amplitude that can be obtained with such a rod within the framework of the EBR-II engineering design. The amplitude may be lowered by 1) reducing the $B_4^{10}C$ content or 2) by fixing the oscillator in a less important position with respect to the core. Both of these conditions were experimentally investigated. Considerations of mechanical design suggest that a vertically moving (up and down) oscillator mechanism may be preferable. Such a mechanism will be experimentally calibrated during the Dry Critical Experiments in the EBR-II.

IV. ANALYSIS OF THE ENGINEERING DESIGN

A. Fissionable and Fertile Materials

All of the fertile material outside of the reactor core is unalloyed depleted uranium ($\sim 0.2\%$ U^{235}). A fabricated density of not less than 18.7 gm/cc was specified. The metallurgical and irradiation characteristics of this material have been extensively studied.⁽¹⁰⁾

The fuel for the first loading is uranium-5 w/o fissium alloy (Table VI) in the form of injection-cast pins weighing slightly more than 67 gm each. About one hundred small specimens and twenty full-size EBR-II fuel pins have been irradiated in the CP-5 and MTR reactors.⁽¹¹⁾ These irradiations, under a variety of temperatures to differing degrees of burnup, have been encouraging. The fuel alloy may withstand between one and two atom percent burnup* at a maximum central metal temperature of 650°C (1200°F). The dimensional stability of the alloy tends to improve as either fissium or zirconium content is increased (at the expense of uranium). It is known that zirconium content in the EBR-II fuel tends to increase with repeated recycling of the fuel alloy. Hence the ultimate potential of this fuel alloy is not exhibited by the first core loading.

Table VI

COMPOSITION OF URANIUM-5 w/o FISSIUM ALLOY

Element	Weight Percent	Element	Weight Percent
Uranium	95.00	Ruthenium	1.97
Zirconium	0.06	Rhodium	0.29
Niobium	0.01	Palladium	0.19
Molybdenum	2.48		

The phase diagram of uranium-fissium is very much like that of the U-Mo-Ru ternary system.⁽¹²⁾ The phase transformation of interest under normal operating conditions is that from the predominantly α phase to the predominantly γ phase. This transformation⁽¹³⁾ occurs at about 550°C.

An analysis of the steady-state heat transfer in the reactor at full power and full coolant flow develops the temperature distribution in each fuel element. At and near the core central plane, all of the fuel alloy is at temperatures in excess of 550°C. Near the axial extremities of the

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*When one percent of the fuel alloy atoms are destroyed by fission, one atom percent (a/o) burnup has been achieved. This represents the destruction of almost 2 a/o of the U^{235} contained in the fuel alloy.

fuel pins, the temperature of the fuel alloy is generally less than or very near to 550°C . There is an isothermal surface of revolution, nonsymmetrically tapered at each end, within the cylindrical fuel pin. Outside of the surface, all temperatures are less than 550°C ; inside the surface temperatures of the fuel alloy exceed 550°C . Under steady-state conditions the unirradiated fuel will exhibit a mixed metallurgical phase structure. Whether this complex geometric phase structure of an unirradiated or low-burnup pin causes geometric distortion remains to be observed. As the burnup of the fuel proceeds, the low-temperature α -phase material may transform to the less dense γ -phase material, even if irradiation temperatures are below 550°C .

The transformation kinetics of the 5 w/o fissium alloy are relatively slow. Complete transformation from α to γ phase would result in a fuel alloy density decrease of about 1.4%. The rate at which this transformation takes place depends on the rate at which the temperature rises and/or the stable temperature at which the transformation is carried to completion. Thermal-expansion measurements⁽¹²⁾ suggest that with a temperature rise of $1^{\circ}\text{C}/\text{min}$, the total transformation, under various conditions, requires between 60 and 100 min. Hardness tests suggest that complete transformation can be effected in about 15 min at temperatures near the temperature of the phase transformation. The transformation at bulk sodium temperature ($\sim 370^{\circ}\text{C}$) will require about 5 hr.

The metallurgical phase phenomena may be the source of a reactivity drift. Whether or not such a drift occurs with unirradiated fuel only remains to be observed. If the expansion from α to γ phase is isotropic, the 1.4% density change will cause a reactivity loss of $0.13\% \Delta k/k$.

B. U^{235} Enrichment of the EBR-II Fuel Alloy

The detailed engineering mockup of the EBR-II with ZPR-III was not precise enough to yield the required U^{235} enrichment of the fuel alloy directly. The results from the critical assembly were interpreted on the basis of both perturbation measurements and reactor analysis.

The first critical loading of the core is designed to require 67 fuel subassemblies, including control and safety rods. The firm requirement that the reactor exhibit criticality with a minimum size equal to the reference core (61 subassemblies) makes it necessary to provide six subassemblies ($< 2\% \Delta k/k$) for "contingencies."

Significant differences between the critical experiment and the EBR-II include:

1. The mockup core height is 14 in. as opposed to 14.22 in. for the reactor.
2. The mockup exhibited the radial properties of a 64-fuel sub-assembly core.

A restricted set of conditions was the basis for fixing the atomic enrichment of the fuel alloy at 48.4%. These include:

1. The fuel alloy is the five weight percent fissium alloy.
2. The as-cast density of the fuel alloy is 17.95 gm/cc. (This is the density at which the fuel pins are cut to proper length.)
3. The density of the blanket uranium is not less than 18.7 gm/cc.
4. The blanket uranium is depleted to about 0.22%.
5. The reference core will contain 67 subassemblies (53 fuel, 12 control, and 2 safety).
6. The control rods are the 61 fuel element control subassemblies.
7. With minor modifications, the configuration and compositions are as given in Table I and Ref. 3.
8. At full power (62.5 Mw) and full coolant flow (8900 gpm) about \$1 of excess reactivity is available from control rod motion.

The enrichment (U^{235} in U) was necessarily specified before significant quantities of fuel elements were fabricated. Studies with prototypes indicated that fuel alloy density would be ~ 17.95 gm/cc. Production of the reactor fuel elements indicated slightly lower densities would be realized.

The enrichment of the fuel alloy was specified on the basis of two not entirely independent approaches. The first was to apply as much of the experimental mockup data as possible with very little recourse to analytical results. The second was based principally on analytical results, and experimental information was used only when analytical results were believed unreliable. The enrichments of the fuel alloy obtained by the two methods were in substantial agreement and well within the uncertainty of the determination.

Table VII is a summary of the analytical results. The basic calculations were two-dimensional multigroup diffusion theory using the NICK-II⁽¹⁴⁾ and CUREM⁽¹⁵⁾ programs with multigroup constants from Ref. 4.

Table VII
ENRICHMENT OF EBR-II FUEL ALLOY
(Several Two-dimensional Analyses)

Problem Number:	1	2	3
Number of Core, Control, and Safety Subassemblies:	61	67	67
Computing Program:	NICK-II	NICK-II	CUM
Fuel Enrichment (U^{235} in U) for Calculation, a/o:	49.5	48.4	48.4
Calculated k_{eff} :	0.9917	0.9992	0.9928
Corrections:			
Comparison between diffusion theory and S_4 , δk :	+0.033	+0.033	+0.033
Basic errors in multigroup constants require, δk :	-0.015	-0.015	-0.015
Fissium Correction, δk :	-0.0031	-0.0031	-0.0031
Corrected k_{eff} :	1.0066	1.0141	1.0077
Add six subassemblies to obtain 67 subassembly core, δk :	+0.0217		
Provide:			
One dollar excess reactivity at full power, δk :	-0.0074	-0.0073	-0.0073
Power coefficient override ($370^\circ\text{C} \rightarrow$ full power), δk :	-0.002	-0.002	-0.002
Temperature coefficient override (Room temperature $\rightarrow 370^\circ\text{C}$) δk :	-0.0067	-0.0067	-0.0067
Corrected k_{eff} (full power, full flow with one dollar excess reactivity):	1.0122	0.9981	.9917
Δk_{ex}	+0.0122	-0.0019	-0.0083
Correction* to initial fuel alloy enrichment for just critical reactor:	-2.68%	+0.0042%	+1.82%
Final fuel alloy enrichment (U^{235} in U), a/o:	48.2	48.6	49.2

$$*\delta k \sim 0.455 \left(\frac{\delta U^{235}/U}{U^{235}/U} \right)$$

Calculation No. 1 represented the reference 61-subassembly core, and Problems 2 and 3 represented the 67-subassembly core. Control and safety subassemblies were homogenized with radially adjacent fuel subassemblies. The six core subassemblies in the inner blanket were homogenized with 24 depleted uranium-bearing subassemblies in Problems 2 and 3. It is recognized that the latter calculations would tend to underestimate the reactivity worth of the six peripheral fuel subassemblies. Corrections to the calculated reactivity were based on known errors of diffusion theory when compared with the more accurate S_N method^(4,6) as well as the known integral uncertainties in the multigroup constants.⁽⁴⁾ The fissium correction was obtained from the ZPR-III data for all constituents except molybdenum. Calculations of temperature and power coefficients are described in a subsequent section. The basic calculations predict different core alloy enrichments. Part of the cause may be ascertained by comparing Problems 1 and 2. The difference is attributed to the homogenizing of six fuel and 24 reflector subassemblies around the 61-subassembly core

boundary. This configuration is calculated to be less reactive than the more heterogeneous experimental mockup. The difference between Calculations 2 and 3 is attributed to the convergence characteristics of the two computing programs. It is not clear which of the two calculations is insufficiently converged. The predicted results are bracketed within 1.0% $\Delta k/k$. Less than three fuel subassemblies in the first row of the inner blanket represent this uncertainty.

The experimental analysis of the EBR-II mockup with ZPR-III is summarized in Table VIII. The alloy enrichment of the critical core for a 67-subassembly core was 48.4 a/o U^{235} in U. The corrections were obtained by comparing the amounts of structural "coolant" and fuel material per unit volume in the mockup with those in the reactor having the dimensions of the mockup (see Table I and Fig. 9). The distributed material replacement experiments (see Tables III and IV) were then used to effect a transition from the aluminum-"cooled" ZPR-III to the sodium-cooled reactor.

Table VIII
ENRICHMENT OF EBR-II FUEL ALLOY (EXPERIMENTAL DATA ANALYSIS)

Region	Corrections to Composition and Size of Principal Reactor Regions		
	Designation (Fig. 9)	Most Probable Reactivity Change (Inhours)*	Possible Errors (Inhours)
Core	C1 C2	-164	± 60
Upper Gap	G1 G2 G3	+ 13 14	± 50 ± 5
Lower Gap	G4 G5 G6	58 - 28	± 20 ± 20
Radial Blanket	B2, B3, B4	- 75	± 35
Total Uranium		+ 88	± 40
Structure and Coolant		32	
Inner			
Outer			
Axial Blanket	B1	+ 10	± 10
Total Uranium		+105	± 10
Structure and Coolant		+ 4	-
First 4'			
Remaining 14'			
Axial Control and Safety Rods	S1, S2 S3 S4	+ 28	± 10
Inhomogeneity Effect in ZPR III		-332	± 150
Add Three Subassemblies to Obtain a 67 Subassembly Core		+450	± 50
Temperature Coefficient (Room $\rightarrow$ 370°C, does not include sodium**)		278	± 50
Power Coefficient (370°C $\rightarrow$ Full Power)		- 83	± 80
Provide One Dollar Excess Reactivity at Full Power and Full Flow		-307	± 10
TOTAL		-673	± 500
Required Reactivity		+ 1.62% $\Delta k/k$	$\pm 1.2\% \Delta k/k$
Change in Fuel Alloy Enrichment (U^{235} in U) to Provide 673 Inhours (1.62% $\Delta k/k$)		3.35%	$\pm 2.5\%$
Reactor Core Alloy Enrichment		48.4 a/o	± 0.012 a/o

* 415 Inhours - 0.01 $\Delta k/k$ 1.0% $\Delta k/k$

** High-temperature (370°C) sodium is included in the static configuration data analysis

It is recognized that the "linearity" principle used to correct the experimental fuel alloy enrichment may not be completely valid. When the mockups are aluminum and sodium "cooled," the measured central worths of sodium were 0.06 and 0.096% $\Delta k/k$ per kilogram, respectively. However, macroscopic material replacement experiments, by substituting sodium for aluminum throughout the EBR-II mockup core, were relatively insensitive to the amount of material substituted. This same effect was analytically investigated by removing different amounts of sodium from the reactor core. It is true that the first increment of sodium introduced into a "dry" reactor has a slightly greater reactivity effect than the average sodium worth. The difference between maximum and average sodium worth does not exceed 10% for the distributed replacements.

The neutron spectrum in an aluminum-"cooled" core is not the same as in the reference sodium-cooled reactor. Natural and enriched uranium fission distributions were measured throughout the core and reflector of both of these cores. The results, shown in Figs. 12 and 13, show that the spectral difference is small and probably within experimental error.

Fig. 12

Enriched Uranium Fission Distributions in Aluminum and Sodium "Cooled" Cores

A small uncertainty is caused by the presence of U^{234} and U^{236} in the fuel alloy. Small amounts of these materials were also present in the ZPR-III fissionable materials. The EBR-II fuel alloy is "blended" by combining highly enriched ($\sim 93\%$) with slightly enriched ($\sim 1.5\%$) uranium. Table IX compares the U^{234} and U^{236} contents of ZPR-III and EBR-II fuel.

The effect of these "impurities" has been overestimated by assuming that an atom of U^{236} behaves like an atom of U^{234} . The removal of one atom percent U^{238} gives a reactivity loss of 0.034% $\Delta k/k$.

Fig. 13

Natural Uranium Fission Distributions in
Aluminum and Sodium-"cooled" Cores.

Table IX

U^{234} AND U^{236} CONTENT OF FISSIONABLE AND FERTILE MATERIAL

	Uranium			
	High Enrichment		Low Enrichment	
	EBR-II	ZPR-III	EBR-II	ZPR-III
U^{234} content (w/o)	1.01-1.09	0.94-1.06	0.0076	0.00103
U^{236} content (w/o)	0.109-0.163	0.402-0.511	0.021	0.0023

Addition of one atom percent U^{234} is estimated to add 0.22% $\Delta k/k$ to the system. This perturbation calculation assumes U^{234} and U^{238} differ only in the fission cross section. The reactor multigroup calculations, which did not include U^{234} and U^{236} , are then subject to a maximum uncertainty of 0.19% $\Delta k/k$. This represents less than one fuel subassembly in the inner blanket (Row 6 in Fig. 5b)

With the 48.4% fuel alloy enrichment, 176 kg U^{235} and 190 kg U^{238} are required for the 67-subassembly core. The available excess reactivity of \$1 suggests then that the hot clean critical mass at full power is 174 kg U^{235} .

The wet critical experiments will be conducted with a uniform sodium temperature of 260°C. Neglecting the effects of source strength, the clean system should have an excess reactivity of 1.32% $\Delta k/k$ with the 67-subassembly core.

The dry critical experiments will be conducted at room temperature with no coolant sodium in the reactor. About 50% of the core volume will be filled with air. The reflecting regions will contain similar or lesser air void volumes. The expected dry critical configuration will be compared with that actually observed. The degree of agreement between prediction and measurement will define, to some extent, the validity of the above predictions. The dry system is not expected to be critical on less than an 81-subassembly core. It is more likely that between 85 and 90 fuel subassemblies, including control and safety rods, will be required.

The major uncertainty in specifying the dry critical configuration is the total reactivity worth of the coolant sodium. The macroscopic sodium material replacement experiments tend to suggest a lower total reactivity worth of sodium than obtained from the purely analytical results. Measured sodium worth in the core is between 0.11 and 0.12% ($\Delta k/k$)/kg, whereas calculations (with multigroup constants from Ref. 4) indicate 0.14% ($\Delta k/k$)/kg. Similar discrepancies occur for the reflector regions.

C. Enrichment of EBR-II Fuel Alloy - Analysis after Fabrication

The enrichment (U^{235} content in U) of the EBR-II fuel alloy is specified at 48.4 a/o (48.08 w/o). This requirement is based on both experiment and analysis. The nuclear and thermal performance of the system depends on how well the fabricated properties of the fuel alloy meet this specification.

Fifty-seven process batches, each containing about 10 kg of the 5 w/o fissium alloy, have been prepared and analyzed. In each case, calculated alloy enrichments, based on constituents added to the 10-kg batch, were compared with sample analyses from the final process batch. The latter samples were mass spectrometrically analyzed to within 1%.

Of the 57 batches, seven were found to be less than 48.0 w/o enriched, and five were determined to be more than 48.2 w/o enriched. Three of the five registering high enrichment were 48.3 w/o enriched; the remaining two 10-kg batches were slightly less than 49 w/o enriched. The seven registering less than 48.0 w/o enrichment were all found to be 47.9 w/o enriched.

If one were to assume the extreme case, that all of the fuel were 49 w/o enriched, the reactor would exhibit 0.84% $\Delta k/k$ more reactivity than predicted. This represents between two and three fuel subassemblies at the core boundary.

The localized variation in enrichment will present itself principally as localized perturbations of the power. The magnitude of this localized power "peak" will be less (about 90%) than the percentage increase of

enrichment. A 2.0% increase in enrichment throughout a single subassembly will give a 1.8% increase in the power generated by that subassembly. This sort of uncertainty, as suggested by the two fuel alloy batches indicating close to 49 w/o enrichment, is well within the "uncertainty factors" of the thermal analysis for the EBR-II.⁽³⁾

D. Reactor Control

The physics of reactor control requires insight on the following questions:

1. What is the total reactivity worth of a single control or safety subassembly?
2. Are there any serious interaction (shadowing) effects?
3. What is the rate of inserting reactivity by control or safety rod motion?
4. What can be done to increase the reactivity worth of a single control rod?
5. What are the reactor conditions which can be protected against by the control mechanisms?

Both experiment and theory confirm that there is enough reactivity in the mechanisms for reactor control and shutdown. The conditions which cannot be protected against require simultaneous gross mechanical failure as well as loss of administrative control. These situations arise due to finite time requirements on fast shutdown (scram). This problem cannot be remedied by enhancing the total reactivity held by the control rods.

The first calculations⁽³⁾ of control rod worth used one-dimensional, cylindrical diffusion theory. The multigroup constants were from Table 13 of Ref. 5. The two-group constants (Table 12, Ref. 5) were also used. The analysis was straightforward. Fuel and void-follower sections of the control subassemblies were homogenized with core sections of fuel subassemblies. Homogenization was accomplished by cylindricizing the hexagonal reactor. Equivalent radii of the subassembly row structure were obtained by equating cylindrical and hexagonal areas.

All of the calculations yielded between 0.5 and 0.6% $\Delta k/k$ per control rod. It was recognized that this calculation was not accurate because of three distinct approximations:

1. The calculation of the cylindrical arrangement tended to remove variation in control rod worth. Six control rods are located on the "corners" and six control rods are located on the "flats" of the hexagonal core (see Fig. 5).

2. There is some effect on the calculated worth of a control rod as a result of the overall homogenization.
3. An axial buckling must be introduced into the cylindrical calculation. This provides for an equivalent core height of about 60 cm; the actual core height is about 36 cm. When removing fuel for a control rod worth calculation, 60% more fuel is removed than in the actual reactor. This is believed to be the largest error introduced by these relatively simple evaluations of the control rods.

The first uncertainty was experimentally resolved. It was found that the control rods on the corner of the hexagonal core hold about 81% of the reactivity of those on the more favorably placed "flats."

The effect of homogenization was analyzed by performing two-dimensional, two-group calculations in cylindrical (r, θ) geometry. The radial asymmetries can be treated in detail. However, the axial buckling problem is the same as in the simpler one-dimensional calculations. The r, θ calculation yielded about the same control rod worth as the one-dimensional cylindrical analysis. The effect of homogenization is found to be small.

The problem of properly treating the axial motion of the actual control rod can be studied using two-dimensional calculations in r, z geometry. These calculations do give fair agreement with experiment. Table X summarizes a variety of control rod calculations.

Table X
EBR-II CONTROL ROD ANALYSIS*

Geometry		Multigroup Constants		Computing Program		Calculated Control Rod Worth ($\% \Delta k/k$)
No. of Dimensions	Type	No. of Groups	Reference	Name	Reference	
1	Cylindrical	2	5	RE-6	16	0.59
2	r, θ	2	5	MUG-II	17	0.58
2	r, z	2	5	MUG-II	17	0.46
1	Cylindrical	11	4	RE-6	16	0.52
1	Cylindrical	10	5**	RE-6	16	0.65
2	r, z	11	4	NICK-II	14	0.36

*Measured reactivity = $0.37\% \Delta k/k$

**Table 14 of Ref. 5

Perturbation theory may be applied to the results of the one-dimensional cylindrical calculations. One may assume that the axial flux distribution can be given by $\cos k_z z$ for all groups, where k_z is the axial buckling. By restricting the perturbation to the actual core height, using a chopped cosine, this approach yields results similar to the r, z calculations. Table XI summarizes the correction factors, based on perturbation theory, which may be applied to one-dimensional cylindrical criticality calculations for the EBR-II.

Table XI

CORRECTIONS* TO ONE-DIMENSIONAL SUBASSEMBLY
REPLACEMENT CALCULATIONS (EBR-II)

Reactor Location	Reactor Row (see Fig. 5b)	Composition**		Multiplication Factor [†]
		Initial	Final	
Center	1	Fuel	Na	0.72
Center	1	Control Rod	Na	0.76
Center	1	Control Rod	Blanket	1.24
Control Rod	5	Fuel	Na	0.81
Control Rod	5	Control Rod	Na	0.83
Control Rod	5	Control Rod	Blanket	1.26
Control Rod	5	Fuel	Blanket	1.07
Inner Blanket	6	Fuel	Na	0.90
Inner Blanket	6	Fuel	Blanket	1.06

*Based on perturbation theory using six energy groups.

**Fuel = Composition of core section of fuel subassembly

Control Rod = Fuel section of control rod

Na = Sodium or sodium void follower region of control rod

Blanket = Composition of blanket subassembly

[†]Results of one-dimensional cylindrical reactivity calculations are to be multiplied by this to obtain more realistic results.

Table XI becomes very useful for semiquantitative predictions without the time and expense of two-dimensional analyses. For instance, one-dimensional control rod calculations yield $\sim 0.55\% \Delta k/k/\text{rod}$. Table XI suggests that a more realistic value would be $(0.83)(0.55) = 0.45\% \Delta k/k/\text{rod}$. While high, this is closer to the measured $0.37\% \Delta k/k/\text{rod}$. Considering the number of simplifying assumptions, including that of specifying a single k_z for all groups, the approach does improve the basic one-dimensional criticality calculation. A two-dimensional calculation and/or an experiment are required for precision reactivity determination.

Table XI may also be used to correct one-dimensional calculations of fuel subassembly removal. The one-dimensional calculated reactivity worth of the central core subassembly is $\sim 2.0\% \Delta k/k$. The correction factor (0.72) suggests that a more accurate value is $1.43\% \Delta k/k$. This replacement was measured to be $1.53\% \Delta k/k$.

The mechanical rates at which the control, safety, and fuel subassemblies are inserted into the reactor are of considerable interest. These rates determine the rate of reactivity addition under both critical and subcritical conditions. Prior to bringing the reactor critical, both control and safety subassemblies are in their least reactive positions. During fuel handling, the safety rods are in the most reactive position, but are "scrammed" prior to the critical approach. On the one hand, it is desirable if the time of approach to power does not exceed one hour. On the other hand, it is desirable if the reactor operator has at least one minute to act upon a malfunction.

Only the control rods can be actuated (other than for scram) at power. With a drive speed of about 5 in./min, all twelve rods can be individually raised in about 33 min. If it is postulated that the control rods continue to insert reactivity once the reactor has reached delayed critical, it will require about 150 sec to approach prompt critical. If a single control rod holds about twice as much reactivity as the reference control rod, about 70 sec are required to go from delayed to prompt critical.

The reactor system is instrumented with limit switches and power level trips to prevent such inadvertent ways of adding reactivity. The system is mechanically limited to the motion of a single control rod at any one time.

Table XII summarizes the total reactivity worths and reactivity insertion rates for control, safety, and fuel-handling systems. The latter is on the basis of maximum reactivity held by the central core subassembly. During fuel handling, the maximum reactivity insertion rate is about $1.8 \times 10^{-2}\%(\Delta k/k)/\text{sec}$. The reactor is at least $3\% \Delta k/k$ subcritical during fuel-handling procedures because the control rods are disengaged (see Fig. 6) and in their least reactive positions. If administrative control should be lost during fuel handling and the reactor is near delayed critical as the central fuel subassembly is inserted into the reactor core, about 40 sec are required to go from delayed to prompt critical. The reactor is instrumented to scram the safety rods on both low power level and short period. If the fuel-handling safety circuits do not function, 40 sec are available for either the reactor operator in the reactor control room or the operator of the fuel-handling system in the reactor plant to manually scram the reactor.

Table XII
NORMAL RATES OF REACTIVITY INSERTION

	Control Subassembly		Two Safety Subassemblies		Central Core Subassembly (Subassembly Loading Mechanism)	
	Predicted*	Measured**	Predicted*	Measured**	Predicted*	Measured**
Total Reactivity Worth, % $\Delta k/k$	0.5	0.37	~ 2.0	1.36	~ 2.0	1.5
Drive Speed, in./min	5	-	2	-	6	-
Effective Stroke, in.	14	-	14	-	$\sim 14^\dagger$	-
Rate of Reactivity Addition % $(\Delta k/k)/\text{sec}$						
Average	0.003	0.0022	0.0050	0.0032	0.015	0.011
Maximum	0.005	0.0038	0.0086	0.0055	0.025	0.018

*Reference 3, original one-dimensional calculations, prior to refinements. These were used to set mechanical drive speeds.

**With ZPR-III.

† Important range.

Certain experimental procedures such as stability studies will require that only eleven control rods will be available for shutdown and power regulation. An oscillator rod will occupy the twelfth control rod location. Furthermore, a future EBR-II loading may operate with a larger core. Investigations have proceeded to determine how the reactivity worth of existing control mechanisms may be enhanced. Three such approaches have been studied in detail:

1. Increase enrichment of the fuel alloy in the control subassembly fuel elements.
2. Increase the number of fuel elements in the control subassembly from 61 to 75. This provides for nonsymmetric placement of fuel elements within the control subassembly.
3. Place poison material ($B_4^{10}C$) in the sodium void-follower regions of the control subassembly.

Both experiment and analysis of the control subassembly show that the total reactivity worth is almost a linear function of the U^{235} content for a given core size. The reactivity worth is inversely proportional to core size. However, as core size increases, the control subassemblies are located at positions of greater importance, which has some bearing on the total reactivity worth. Reactor physics can barely distinguish between 1. and 2. for a given U^{235} content in the control subassembly. The first approach requires the use of fuel elements having two different enrichments of U^{235} in the remote fabrication facility; the higher enrichments are for control subassemblies, while the larger number of lower enrichment fuel elements are for fuel subassemblies. The possibility of inadvertent interchange could cause serious reactivity as well as problems of heat removal. A highly enriched central subassembly in the core would overheat and probably fail. In addition, this subassembly could hold more

than 2.5% $\Delta k/k$ total reactivity. The physics of this type of control are straightforward, but the management and engineering problems dictated that it should be relegated to the future.

Provision of more fuel elements per control subassembly obviates the chance of inadvertent interchange cited above. However nonsymmetric internals of control subassembly fabrication as well as problems of heat removal made this approach not very desirable.

The final approach shows the most promise for the initial reactor loading. Experimentally it has been demonstrated that a single control rod worth can be raised from 0.37% $\Delta k/k$ to $\sim 0.7\%$ $\Delta k/k$ by inserting ~ 350 gm $B_4^{10}C$ in the sodium void follower region of the control subassembly. (The reactivity worth may be slightly enhanced by using Ta.) The main problems of using $B_4^{10}C$ are also those of engineering design. Briefly, they are associated with high temperature or heat removal and with carburization of clad material. The primary physics problem is that concerned with long-term exposure of $B_4^{10}C$ in high neutron flux. It is also recognized that neutrons absorbed by $B_4^{10}C$ are not available for breeding. The EBR-II will be operated with all control rods the equivalent of about 80% inserted. With a 14-in. control rod stroke, it was decided to place the poison material in a region about 7 in. long. Thus the effective 28 in. of "strong" control rod are fuel (14 in.), sodium and steel void (7 in.), and $B_4^{10}C$ (7 in.). Poison material would not begin to enter the axial core extremity until half of the fuel section of the control rod has been moved below the core. Such control subassemblies are being provided for start-up experiments. They can be used to provide a relatively constant shut-down reactivity for all experiments.

E. Power Distribution and Calibration

The reactor design requires high thermal performance with fairly uniform coolant temperature at the outlet. This places severe requirements on knowing the neutron flux and power distribution throughout the reactor system. Comparison between theory and experiment has been demonstrated in Figs. 9 and 10 and in Ref. 5. However, the power distribution for heat removal is slightly different from the fission distribution. A distinction must be made between the location where energy is created and where it is absorbed. A small fraction of the energy, principally γ rays, is absorbed at some distance from the point of creation. This consideration poses some problems of heat generation in reflector regions where fission densities are low, and which are located adjacent to core regions where fission densities are high.

The power distribution for reactor analysis is obtained from the 11-energy-group CUREM⁽¹⁵⁾ calculation (Figs. 9 and 10). Figure 14 is a simplified sketch of the analysis in r, z geometry. Table XIII extends the description and gives the power produced in various regions of the reactor. This analysis assumes that all twelve control rods are fully inserted.

Fig. 14

Reactor Model for Two-Dimensional Multigroup Analysis

Table XIV demonstrates the wide variation in power density throughout the reactor. The corrections for γ heating in the inner and first row of the outer blanket are given. The additional power produced due to 0.5 a/o plutonium production in reflector material is also given.

The cycling of blanket subassemblies cannot be specified until reactor experience has been obtained. Thus there is a certain arbitrariness in specifying the plutonium concentration and consequent contribution to power density in the reflector. For orientation, the first question to be answered is: When does this problem become important? Table XV gives the time, in years, required to achieve both 0.5 a/o burnup of uranium and 1.0 a/o buildup of plutonium at the core midplane as a function of radius. Time required is based on full-power operation, a 67-subassembly core, and an assumed 100% plant factor. The ratio of the total fission power when 1 a/o plutonium is present $P(1\% \text{ Pu})$ to that when no plutonium is present [clean reactor, $P(0)$] is also given. Approximate local fission power densities are given for reference.

Table XIII

POWER PRODUCED IN VARIOUS REGIONS OF A 67-SUBASSEMBLY EBR-II CORE

(Total Power = 62.5 Mw)

Hexagonal Subassembly Row No. (see Fig. 5b)	Region of Figure 14		Fraction of Total Power (%)	Total Power (Mw)	Upper and Lower Blanket Bonus (%)*	Average Total Power per Subassembly (Mw)
1,2,4	1	Fuel Subassemblies	39.11	24.44	1.65	0.9941
3	2	Fuel Subassemblies	16.63	10.39	1.65	1.0561
		Safety Subassemblies	<u>2.23</u>	<u>1.39</u>	-	0.695
			18.86	11.78		
5	3	Fuel Subassemblies	14.68	9.18	1.22	0.774
		Control Subassemblies	<u>9.84</u>	<u>6.15</u>	-	0.513
			24.52	15.33		
6	4	Fuel Subassemblies	5.70	3.56	2.54	0.608
		Blanket Subassemblies	<u>3.21</u>	<u>2.01</u>	36.25	0.114
			8.91	5.57		
7 ff	5	Radial Blanket	6.32	3.65		
1,2,3,4	6	Upper and Lower Blanket	0.792			
5	7	Upper Blanket	0.0887			
5	8	Lower Blanket	0.0907			
6	9	Upper and lower blanket of 6 fuel subassemblies and upper and lower portions of blanket subassemblies	0.5071			
6	10	Portion of blanket subassemblies coinciding with structural gap of fuel subassembly	0.7996			

*From Regions 6,7,8,9,10

Table XIV

POWER DENSITIES IN CORE CENTRAL PLANE ($\sim z = 0$) FOR
67-SUBASSEMBLY CORE AT FULL POWER

Subassembly Row No.	Region	Maximum Power Density* (Mw/liter)					
		Fuel Subassembly	Control or Safety Subassembly	Blanket Subassembly			
				No γ Heat	With γ Heat	With 0.5 a/o Pu** (No γ)	With 0.5 a/o Pu** (With γ)
1	Core	1.23	-	-	-		
2		1.20	-	-	-		
3		1.11	0.740	-	-		
4		0.967	-	-	-		
5		0.803	0.538	-	-		
6	Inner Blanket	0.626	-	0.0832	0.120	0.0972	0.134
7		-	-	0.0391	0.0637	0.0516	0.0762
8	Outer Blanket			0.0157	0.0279	0.0234	0.0356
9				0.0045	0.0050		
10				0.0034			

*Averaged over radial area of subassembly.

**It is assumed that Pu power in blanket is in excess of 62.5 Mw. (Actual reactor power >62.5 Mw).

Table XV

PERTURBATION OF CLEAN REACTOR POWER IN REFLECTOR
MATERIAL BY PRESENCE OF BRED PLUTONIUM

Region	r(cm)*	T _{Pu} ** (years)	T _U † (years)	$\frac{P(1\% \text{ Pu})}{P(0)}$	P(0) (kw/liter)
Inner Blanket	22.4	0.85	0.93	1.22	222
	24.2	0.90	1.14	1.25	162
	26.0	0.98	1.56	1.31	111
	27.7	1.07	2.14	1.37	82
	29.5	1.18	2.8	1.43	60
	32.8	1.47	5.7	1.65	31
Outer Blanket	36.0	1.87	11.0	1.91	18
	39.2	2.43	20.0	2.22	11
	49.0	5.76	94.0	3.30	2.7
	58.7	14.5	330.0	4.0	0.89
	68.4	39.0	-	4.35	0.29
	78.1	100.0	-	4.74	0.10

*Distance from center of core in reactor midplane.

**Time required to achieve 1 a/o Pu buildup at core midplane.

†Time required to achieve 0.5 a/o burnup at core midplane.

Some details of the power distribution are given in Refs. 2 and 3.

An absolute power calibration is required to provide (1) knowledge of power level before the heat-removal system becomes operative and to (2) permit realistic power level trip settings prior to plant operation. This power calibration is performed relative to an absolute fission counter⁽⁹⁾ response. The counter, containing 0.8 mg of uranium (93% enriched), is located in the reactor midplane at the radial core boundary. This calibration has been obtained on the basis of integrating measured fission distributions from ZPR-III.⁽⁷⁾ It has also been obtained from the multigroup analysis. There is a 15 to 20% difference between the experimental and analytical power calibration.

This agreement between theory and experiment is adequate for reactor startup until the heat-removal system is operative. To a high degree of approximation, the 0.8-mg enriched uranium fission counter will record:

$$\text{Count Rate at Core Boundary (cps)} = 10^4 \frac{P}{M_c}$$

where

P = Reactor power, watts

M_C = Critical Mass, kg U^{235} .

These results are then related to the normal flux-monitoring instruments located in the J and O thimbles (see Fig. 7). The data in Table V were normalized in this way. Experimentally, this was accomplished by simultaneous count rate measurements by two different counters. The absolute counter is in the reactor core while the normal reactor instrumentation is in a ZPR-III mockup of the instrument thimble.

F. Neutron Flux Distribution and Instrument Response

The experiments designed to determine the attenuation of the neutron flux from the reactor core through the radial reflector and into the neutron shield are summarized in Table V. The experimental arrangement is shown in Fig. 15. In that figure, the symbols "W," "C," "J," "#5," "4," and "PC" refer to fission and proportional counters. The responses of these instruments, when calibrated relative to each other,⁽⁷⁾ give the spatial distribution of the neutron flux. (The early experiments with a different neutron shield configuration yielded similar information.)

Fig. 15

Shield Mockup of EBR-II on ZPR-III (Dimension in inches)

It is recognized that the attenuation of the neutron flux through the neutron shield depends upon accurate treatment of the large instrument thimble. However, it is interesting to observe that relatively simple spherical multigroup calculations do predict the qualitative features of the neutron-flux attenuation in the outer reactor regions. Unpublished 20-group constants were used in the analysis. They are similar to the 19-group constants of Ref. 18.

Figure 16 summarizes the measured and predicted U^{235} fission distributions for the graphite neutron shield. No moderating material is assumed to be present in the interior of the instrument thimble. Two calculations are shown: one has graphite in the inner shield; the other inner shield is filled with coolant and structural material. The experimental data are normalized to the calculated results at the center of the core. The qualitative features of the experimental fission distribution are predicted. However, it is quite apparent that the precision of such prediction is inadequate for precise reactor design.

Fig. 16

Calculated and Measured U^{235} Fission Distributions
to the Neutron Shield (Instrument Response)

Predicted counter responses on the J-thimble centerline are between two and five times greater than the measured values. There is some evidence from the comparison between theory and experiment, that agreement could be improved by increasing structural (Fe) absorption cross sections. However, it is also likely that the neutron spectrum is not accurately calculated. Multigroup calculations tend to predict harder spectra than are measured. Finally, the absorptions in the instrument thimble were not included in the analysis. A parametric study of the latter effect, by reducing the graphite and increasing the Fe content in the outer shield, showed that the gross features of the fission distribution are as given in Fig. 16.

These analyses and experiments are basic to the construction of the nuclear detector range charts.⁽⁸⁾ The most important considerations are:

1. Provision for adequate startup flux. About 3-6 unmultiplied counts per second will assure reliable flux monitoring at all reactor multiplications.
2. Provision for "comfortable" overlap between startup fission counters and intermediate and high power ion chambers.

Startup fission counters are located in J thimbles (see Fig. 7) with the "no graphite" inner shield. Intermediate-power-level ion chambers are located in J thimbles with the "graphite" inner shield. An overlap of a factor of about ten is provided for these instruments. High-power-level ion chambers located in the "O" thimbles are not operative below 10 kw.

G. Reactivity Effects

The determination of reactivity effects due to motion or expansion of materials in the reactor system is not easily accomplished. Small-sample substitutions, without appreciable spectral shifts due to the samples, can be treated by first-order perturbation theory. Macroscopic substitutions, with spectral shifts, are more readily handled by means of criticality analyses. The components of both the isothermal temperature coefficient and the power coefficient of reactivity are based on macroscopic substitution experiments or calculations. It would be desirable to use one-dimensional calculations as opposed to the more complicated two-dimensional analyses. The following procedure was adopted for calculating reactivity effects due to expansion and movement:

1. Devise both an "axial" and a "radial" spherical reactor calculation. In the "axial" calculation, spherical reflector shells are taken whose composition and dimensions are those along the reactor centerline (axis of symmetry). In the "radial" calculation, spherical reflector shells are taken

whose composition and dimensions are those along the central core plane (almost a plane of symmetry). The core composition is uniform and the core radius is adjusted so that the reactor is critical with about 48% enrichment of the fuel alloy.

2. Reactivity effects are calculated with these models. Where applicable, both an "axial" and a "radial" calculation are performed. For example, a calculation of sodium worth in the axial structural gaps is made with an "axial" sphere. A calculation of sodium worth in the core is performed with both an "axial" and a "radial" sphere.

3. Calculated results are then adjusted by combining the results of the "axial" and "radial" calculations. Weighting is determined by the core surface area facing in the axial and radial directions.

Where feasible, the predicted results are compared with those measured at ZPR-III (see Table III, etc.). Such comparison is not direct. To demonstrate, an important component in both power and temperature coefficient determinations is the effect of sodium density in the reactor core, predicted to be $\Delta k/k = -0.87 \times 10^{-5}/^{\circ}\text{C}$. The experimental investigation at ZPR-III dealt with removing sizeable quantities of sodium from the reactor core to effect a gross density change of sodium. The measured worth was $\Delta k/k = -1.16 \times 10^{-3}/\text{kg}$ of sodium. The mass of sodium in the core is about 29 kg and the coefficient of expansion is given as $2.9 \times 10^{-4}/^{\circ}\text{C}$. Therefore, the effect of sodium density inferred from the ZPR-III experiments is $\Delta k/k = -0.98 \times 10^{-5}/^{\circ}\text{C}$. The experiment is an attempt to simulate power reactor conditions with a zero-power, fixed-temperature machine. The calculations feature simple idealized geometric models. Comparisons between theory and experiment are encouraging. A number of such comparisons are given in Table XVI. Allowing fairly large uncertainties for those results not comparable with ZPR-III experiments, it is believed that the calculated isothermal temperature coefficient, given as $-3.6 \times 10^{-5} \Delta k/k$ per $^{\circ}\text{C}$, is known to ± 10 percent.

One-dimensional reflected cylinders and slabs were used for some calculations of reactivity effect. These calculations were "incorrect" in the same sense that the one-dimensional control rod analyses (Table X) must be "corrected" (Table XI). For example, the incremental worth of core sodium, by one-dimensional cylindrical analysis, is calculated to be ~ 2.5 times as large as that obtained from the spherical analyses. Similarly, the incremental worth of sodium in the axial structural gaps, by one-dimensional slab analysis, is calculated to be $\sim 33\%$ less than that predicted by spherical analyses.

Table XVI

COMPONENTS OF THE ISOTHERMAL TEMPERATURE AND
POWER COEFFICIENT OF REACTIVITY ($\Delta k/k/^\circ\text{C}$)

	Predicted*	Inferred from ZPR-III Measurements
<u>Core</u>		
Axial growth of fuel	$-0.39 \times 10^{-5} \Delta k/k/^\circ\text{C}$	$-0.34 \pm 0.02 \times 10^{-5} \Delta k/k/^\circ\text{C}$
Radial growth of fuel (displacement of Na)	-0.09×10^{-5}	-0.057×10^{-5}
Axial growth of structure (density change)	-0.039×10^{-5}	-0.033×10^{-5}
Density change of coolant	-0.87×10^{-5}	-0.98×10^{-5}
Radial growth of supporting structure	-0.97×10^{-5}	-0.92×10^{-5}
Doppler effect	$+0.04 \times 10^{-5}$ average	
Bowing		
<u>Gaps</u>		
Density change of coolant	-0.38×10^{-5}	-0.33×10^{-5}
Density change of structure	-0.036×10^{-5}	-0.04×10^{-5}
<u>Upper and Lower Blanket</u>		
Density change of coolant	-0.21×10^{-5}	-0.21×10^{-5}
Radial growth of uranium and jacket	-0.016×10^{-5}	
Axial growth of blanket uranium	-0.024×10^{-5}	
Axial growth of jacket	-0.021×10^{-5}	-0.0064×10^{-5}
<u>Inner Blanket</u>		
Density change of coolant**	-0.2×10^{-5}	-0.30×10^{-5}
Axial growth of blanket uranium	-0.066×10^{-5}	
Axial growth of jacket**	-0.022×10^{-5}	-0.054×10^{-5}
Radial growth of uranium and jacket	-0.07×10^{-5}	
Radial growth of supporting structure	-0.17×10^{-5}	
Bowing	0	
<u>Outer Blanket</u>		
Density change of coolant**	-0.017×10^{-5}	-0.011×10^{-5}
Axial growth of blanket uranium	-0.014×10^{-5}	
Axial growth of jacket**	-0.003×10^{-5}	-0.0012×10^{-5}
Radial growth of supporting structure	-0.034	

*Reference 3, Multigroup Constants from Tables 13 and 14 of Ref 5

**The experimental results are difficult to interpret because the radial boundaries, especially between core and blanket, are not well defined

Combinations of components of the isothermal temperature coefficient into the steady-state power coefficient of reactivity depends upon knowing the detailed temperature distribution throughout the reactor system. An attempt to combine basic reactivity effects with temperature distribution is given in Ref. 3. The power coefficient, based on the reactivity difference between an isothermal system at 700°F and full-power, full-flow operation, is predicted to be $-3.2 \times 10^{-5} \Delta k/k/\text{Mw}$.

The influence of fuel element and subassembly bowing is not included in the above result. Uncertainties with respect to bowing are not completely resolved. If bowing were to take its most pessimistic course, a relatively prompt positive component to the power coefficient of reactivity would appear. The magnitude may or may not be sufficient to yield a net positive power coefficient of reactivity at certain power levels below full power. However, it is significant to note that with pessimistic bowing analyses, there is a net reactivity loss between reactor startup and full-power operation.

The most likely situation is that the power coefficient of reactivity is negative at all powers. A small prompt positive component may be present at low powers but will not be present at or near full power. Principal evidence for this will be inferential. An increase in the magnitude of the steady-state power coefficient of reactivity with power level would be some evidence for this.

A long-term reactivity drift is caused by the burnout and subsequent irradiation growth of the fuel alloy. The reactivity loss due to uniform burnout of fuel alloy was measured (Table III) and was predicted to be

$$\frac{\Delta k}{k} \approx 0.55 \frac{\Delta M}{M} ,$$

where

$$\frac{\Delta M}{M} = \text{Fraction of } U^{235} \text{ destroyed in the core.}$$

The value of the predicted reactivity loss is relatively independent of both geometry and multigroup constants. The additional reactivity loss due to the irradiation growth of fuel alloy is difficult to estimate. There is some evidence,⁽¹¹⁾ from irradiations conducted near 600°C, that the fuel elements experience a 2% volume increase per a/o fuel alloy burnup. There is further evidence⁽¹¹⁾ which suggests that most of the irradiation growth is in the radial direction within a single fuel element. If this is true, the long-term reactivity effect is negligible. If it is assumed that the expansion is nearly isotropic, the 2% volume change per a/o burnup would appear as a 0.67%

change in length of the fuel pin. This change in length represents between 0.16 and 0.19% $\Delta k/k$. The first value is suggested by indirect ZPR-III experiment, while the latter is predicted. Therefore, between 0.71 and 0.74% $\Delta k/k$ may be required for accommodating one a/o uniform core alloy burnout and accompanying irradiation growth. This does not include the additional reactivity loss estimated at $0.1 \pm 0.05\%$ $\Delta k/k$, because of the presence of core fission products. This effect requires about 60 days of full-power operation at 80% plant factor.

Provision for transforming the fuel alloy from the more dense α phase to the less dense γ phase represents another 0.13% $\Delta k/k$.

Reactor operating experience will dictate the exact mode of cycling fuel subassemblies. However, the above estimates are indicative of the long term core reactivity effects.

V. NUCLEAR PERFORMANCE

The breeding of nuclear fuel in the EBR-II will initially be that of converting U^{235} to Pu^{239} . (True breeding, as demonstrated by simultaneous destruction and creation of Pu, must await a future loading of the system.) As U^{235} is destroyed, principally in the reactor core, Pu^{239} is created throughout the system. Most of the bred Pu ($\sim 90\%$) is distributed throughout the depleted uranium reflectors. The bred plutonium is located at positions of low reactivity worth relative to the location of the fissioned enriched uranium. If no rearrangements of subassemblies takes place during a reactor cycle, the reactivity of the system will not increase because of the bred plutonium. The overriding consideration in long-term reactivity behavior is the loss associated with the destruction of U^{235} in the core.

Plutonium buildup in the radial blanket elements does have some influence on the critical size of the second and subsequent core loadings. Suppose the first reactor core is permitted to achieve an average of one a/o burnup. It has been calculated that a second core loading, identical with the first, will require one less core subassembly in the first row of the inner blanket. This analytical result assumes that the same amount of startup excess reactivity is available in both cores. It further requires that the radial blanket is not recycled.

Reverting to the case of one a/o core burnup, the smaller second core will dictate slightly higher power densities in the radial blanket for fixed total power of the reactor. Actually, inner blanket subassemblies will initially require nonoptimum recycling because of unknown optimum irradiation exposure for unalloyed depleted uranium.

The variation of core size as a function of time has some effect on the nuclear performance of the system. Specific power (kw/kg), breeding gain, and doubling time are slowly varying functions of the time.

With a total power output of 62.5 Mw and a clean critical mass of 175 kg U^{235} , the initial average specific power in the EBR-II converter is 360 kw/kg. A Pu-fueled EBR-II with a 110-kg critical mass has an average specific power of 570 kw/kg.

An initial conversion ratio of 1.26 has been predicted for the EBR-II. This prediction is uncertain on two counts:

1. More recent cross-section data⁽¹⁹⁾ suggest that the neutron yield per fission of U^{238} is higher ($\nu U^{238} \approx 2.8$) than the analyses provided ($\nu U^{238} \approx 2.6$). This effect, although small, would tend to raise the predicted value.

2. It has long been known^(4,5,19) that the predicted value of $\sigma_{n,\gamma}^{U^{238}}/\sigma_f^{U^{235}}$ exceeds the measured value by as much as 20% at the center of the reactor core. This could mean that measured breeding or conversion ratios will be less than predicted.

About 90% of the predicted conversion is in the breeder reflector (external breeding ratio ≈ 1.16). The remaining 10% of the predicted conversion is in the core (internal breeding ratio ≈ 0.10). This is where the nuclear performance of the EBR-II differs from that of the larger (>400 -liter cores), more dilute, conceptual fast neutron systems. In the latter, almost half of the breeding or conversion may take place in the reactor core. The predicted initial breeding ratio for the Pu-fueled EBR-II is 1.7; the Pu-U-10 w/o fissium alloy is a possible fuel for such a loading. For a ternary fuel element of Pu-U²³⁵-U²³⁸-10 w/o fissium, the predicted combination of breeding and conversion ratio will be about 1.5. The upper limit of 20% uncertainty in the parameter $\sigma_{n,\gamma}^{U^{238}}/\sigma_f^{U^{235}}$ does not seriously jeopardize the predicted breeding gain of a Pu-fueled system.

The cited breeding constants are for the actual EBR-II engineering design. It is interesting to note that significant breeding parameters are less favorable than those obtained from the conceptual studies. The latter, performed for spherical geometry for an idealized reactor, do not include compromises to engineering design. Three such accommodations in the EBR-II are:

1. The axial blankets contain only about 30 v/o depleted uranium as opposed to 60 v/o in the radial reflector.
2. None of the 14 control and safety subassemblies contain any fertile reflector material.
3. Critical mass requirements are increased over those of the conceptual study. The presence of axial gaps for fuel element expansion and structural reasons reduce the axial reflector saving.

These requirements suggest that the inhomogeneous axial reflector is quite "leaky." Furthermore, cycling of axial fertile material is dictated by considerations of burnup of the core alloy. Unless a scheme is devised for reinserting the axial reflector elements, the use of the axial blanket is almost academic. When the core alloy reaches 1 a/o burnup, maximum localized burnup in the axial reflector does not exceed 0.04 a/o. The corresponding maximum localized production of plutonium does not exceed 0.1 a/o in the fertile material.

The above considerations prompted some investigations of the nuclear performance of the EBR-II with no fertile material above and below the core. It was found that replacement of the axial reflector uranium

by stainless steel tends to increase the reactivity of the system; as much as 3% $\Delta k/k$ can be obtained by such replacement. This would lower critical mass requirements. The loss in conversion ratio accompanying such replacement is less than 10%. However, this may be as much as 30% of the breeding gain. Table XVII summarizes several predicted conversion parameters.

Table XVII

INITIAL CONVERSION PARAMETERS FOR THE EBR-II

Description	No. of Groups	Geometry	Conversion Ratio				Estimated Doubling Time (years)
			Core	Axial Reflector	Radial Reflector	Total	
Reference EBR-II	11	r,z	0.125	0.109	1.03	1.26	35
Reference EBR-II	2	r,z	0.133	0.083	1.01	1.23	-
No Uranium in Axial Reflector	2	r,z	0.122	-	1.06	1.18	-
Idealized EBR-II	11	sphere	0.11	1.23		1.34	20
Idealized Pu-fueled EBR-II	11	sphere	0.21	1.63		1.84	6

VI. SAFETY CONSIDERATIONS

A thorough review of the safety considerations pertaining to the design and operation of the EBR-II requires the correlation of a number of diverse efforts and disciplines. Various aspects of this problem have been described elsewhere.(3,8,20-23) This discussion is limited to a few topics dealing primarily with the causes of a nuclear accident. Some courses which may follow the initiation of such an event are also described.

The overall safety of reactor operation depends on answering the following questions:

1. Are the mechanical components for inserting and removing reactivity reliable? If not, what may be the consequences?
2. Is the reactor system inherently stable during steady-state operation?
3. To what extent is administrative control required to avoid a nuclear event?

A. Mechanical Malfunction

The reactor system, its mechanical components, and the flux-sensing instrumentation provide a great degree of reliability.(3,8) The design is such that single mechanical failure or malfunction will not cause serious injury to the system or surroundings. For example, the remote possibility of failure of fuel-handling equipment to the point where a subassembly is actually dropped into the reactor core poses no serious hazards problem. Normal fuel-handling procedures require that the reactor be between 3 and 4% $\Delta k/k$ subcritical while fuel is inserted into or removed from the reactor core. Should a core subassembly drop into the reactor core, the rapid rate of increasing source-multiplied flux will scram the safety rods (period trip). Even if this period trip is not operative, the reactor is still subcritical after the subassembly has been dropped into the core. Such a malfunction (dropping of a fuel subassembly) becomes hazardous only if it accompanies or is accompanied by some other malfunction or loss of administrative control.

Dropping a fuel subassembly can have extremely serious consequences if the reactor is at or near delayed critical. The control rods are in their least reactive positions while fuel is being loaded; the safety rods are available for shutdown purposes. To postulate the reactor at delayed critical under these conditions requires a loss of administrative control and gross malfunction of flux monitoring equipment. (Malfunction of the latter will normally cause safety rod scram and prevent further fuel handling.) Between 3 and 4% $\Delta k/k$ must have been unwittingly added to the system to realize a critical reactor.

A gross loss of administrative control or a gross system failure is required to effect the addition of such large amounts of reactivity. The following four examples show the extent of failure or loss of control necessary to obtain a critical reactor during fuel-handling operations:

- a. About ten more than the six required fuel subassemblies must be loaded into the first row of the inner blanket (see Fig. 5).
- b. There is an unknown fuel alloy "blending" error with uranium enrichment about 6% uniformly higher than the specified 48.4%.
- c. Several subassemblies, between 5 and 10, contain highly enriched (~93%) uranium instead of the specified 48.4%.
- d. There must occur a gross failure of between 10 and 15 subassemblies with all contained fuel displaced toward the core midplane.

None of these situations should arise; if they do, they should be noted by operating personnel. Subcritical count rates during fuel handling should not exceed 30 times unmultiplied source count rates. Rapid additions of reactivity will be noted by the automatic period trips. The power level trip during fuel-handling procedures is set below 1 kw.

Six specific accidents caused by malfunction of mechanical rod drive mechanisms have been analyzed in detail⁽³⁾:

1. The two safety subassemblies are driven into (upward) a critical reactor core at 2 in./min under zero-power conditions.
2. A fuel subassembly is driven (downward) into the center of a critical reactor at a speed of 6 in./min under zero-power conditions.
3. A single normal control subassembly is driven (upward) into a critical reactor at a speed of 5 in./min under zero-power conditions.
4. A single normal control subassembly is driven (upward) into a critical reactor at a speed of 5 in./min under full-power and full-flow conditions.
5. A central fuel subassembly is driven (downward) into a critical reactor at 72 in./min under zero-power conditions.
6. A fuel subassembly is dropped into a critical reactor under zero-power conditions.

Of the six cases cited, only two (3 and 4) can occur without some gross error in reactor loading having taken place prior to the mechanical malfunction. Accidents 3 and 4, dealing with control subassembly motion,

can be effectively prevented by power level trip circuits, even though the reactor operator should note either unusual increase in power level or indication of rod motion before the trip circuits function.

The kinetic behavior of the six EBR-II malfunctions has been studied⁽³⁾ by applying the one-group space-independent model.⁽²³⁾ The objective is to obtain information concerning the time variation of the neutron flux and power. From these, the time variation of temperature is determined, which in turn specifies limits on reactor operation. Several parameters are basic to all the kinetic analyses:

- a. Prompt neutron lifetime for the EBR-II is predicted to be 8.0×10^{-8} sec. Measurements at ZPR-III (by Rossi- α) suggest a value between 7.5 and 8.5×10^{-8} sec.
- b. The effective delayed neutron fraction is $\beta_{\text{eff}} = 0.0073$.⁽⁴⁾
- c. Shutdown coefficients due to thermal expansion are based on the components given in Table XVI. Except for the full-power malfunction (Case 4), only core shutdown coefficients are operative. For rapid rates of reactivity insertion, only fuel alloy expansion is an initially effective shutdown mechanism. For extremely rapid rates of inserting reactivity, the fuel alloy expansion is inertially inhibited⁽²²⁾ during the early stages of the excursion.
- d. Reactivity insertion rates are based on the "predicted" values in Table XII. These are higher than actually expected in the EBR-II.

Even during a slow excursion, starting at zero power and flow, thermal expansion effects in the reflectors are almost negligible. If it is assumed that most of the heat produced in a given reactor region remains in that region until the excursion is terminated, the temperature rise in the reflectors is small compared to that in the core. Table XVIII gives appropriate total temperature coefficients for the various reactor regions. The temperature rise of each region per degree (centigrade) rise in the core is also given. The latter is based on detailed power-distribution calculations whose integral results are given in Table XIII.

The consequences of the five cited zero-power malfunctions depend upon the initial temperature of the system. Except for the initial critical approach, the bulk system temperature will not be less than 315°C . The melting point of the fuel alloy is near 1130°C . Another temperature of interest is the stainless steel-uranium eutectic formation temperature of 725°C . It has, however, been experimentally demonstrated⁽²⁴⁾ that the latter temperature may be exceeded for short periods of time without failure of fuel elements. The conclusions drawn from the six kinetic analyses are summarized in Table XIX.

Table XVIII

REACTIVITY AND TEMPERATURE EFFECTS
UNDER ZERO FLOW CONDITIONS*

Reactor Region (Figs. 5, 14)	$\Delta k/\Delta T$ (Region) (per °C)	$\frac{\Delta T \text{ (Region)}}{\Delta T \text{ (Core)}}$	$\Delta k/\Delta T$ (Core) (per °C)
Core	-1.28	1.0	-1.28
Inner Blanket	-0.30	0.045	-0.014
Outer Blanket	-0.03	0.010	-0.0003
Upper plus Lower Blankets	-0.26	0.02	-0.0052
Upper plus Lower Gaps	-0.42	~0	~0

*No radial expansion is included. No convection or conduction is assumed.

The first four cases in Table XIX operate on a fairly long time scale and can be terminated by the reactor operator as well as by the automatic trip circuits. The last two cases (5 and 6), dealing with high-speed subassembly insertion into the center of the core, may not be terminated by automatic trip circuits and cannot be terminated by the reactor operator. However, it must be noted that the severe consequences are reduced, and more time is available for trip circuits to function under certain conditions. In particular, available time is increased if the reactor is somewhat subcritical prior to the malfunction. However, the consequences may be more severe if the available time is still insufficient to shut the reactor down. In the gravity-induced reactivity insertion, the rate of adding reactivity when the reactor is near prompt critical is greater if the reactor is initially below delayed critical.

Further, it must be noted that the coincident malfunction of fuel-handling equipment along with the loss of administrative control must be accompanied by another coincidence. The latter refers to dropping the subassembly near the center of the core. If the reactor is at or near delayed critical and a fuel subassembly is dropped near the core boundary, the reactor will go on a period between 5 and 10 sec. Power level and period trips as well as operator action can shut the reactor down.

The reactor system is designed to cope with mechanical malfunction. However, these studies show that coincident malfunction and operational errors can lead to extremely severe consequences.

Table XIX

CONSEQUENCES OF MECHANICAL MALFUNCTION
COUPLED WITH LOSS OF ADMINISTRATIVE CONTROL

Case No. (see text)	Initial Power (watts)	Malfunction	Will Period Trip Terminate Excursion?		Will Power Level Trip Terminate Excursion?		If Trips are Inoperative Will Melting be Initiated?	
				When (sec)?*		When (sec)?*		When (sec)?*
1	8	Safety Rods	yes	<70	yes**	70	yes	120
2	8	Fuel Subassembly Loading Mechanism	yes	<25	yes**	30	yes	55
3	8	Control Rod	maybe	~100	yes**	~100	maybe	~230
4	6 x 10 ⁷	Control Rod	no	-	yes†	~17	yes	~160
5	8	Fuel Subassembly Loading Mechanism (High speed)	yes	<1	yes**	4	yes	~6
6	8	Fuel Subassembly Loading Mechanism (Gravity induced)	probably	immediately	no** maybe‡	- immediately	yes	0.13

*After insertion of reactivity is initiated.

**~10³ watts

† ~78 Mw

‡ <100 watts

B. Nuclear Accidents

The TREAT (Transient Reactor Test) meltdown program^(24,25) is an experimental investigation into the behavior of fast reactor fuel elements under conditions of transient power excursions. Mechanisms of fuel element failure and the distribution of resulting meltdown products are being investigated on an in-pile basis in an attempt to define accident conditions and to determine the parameters of importance for describing or predicting the course of a meltdown in a fast reactor.

Tests have been performed to study the failure of single, clad, and bonded EBR-II-type fuel pins. These were contained in an inert atmosphere in a stainless steel capsule. Similar experiments with clad and bonded EBR-II fuel pins suspended in stagnant or flowing sodium have been initiated.

Some accidents have been postulated on the basis of these experiments. Results obtained, although not unexpected, were different from those of out-of-pile meltdown experiments using electrical heating. The latter investigations were conducted during the EBR-II, Mark I fuel-element development program. Very significantly, both in-pile and out-of-pile experiments showed that unirradiated EBR-II fuel (5 w/o fission) can be heated without failure to temperatures appreciably in excess of 725°C for limited times.

TREAT experiments⁽²⁴⁾ dealing with single EBR-II fuel pins contained in stainless steel clad and bonded with sodium suggest that rearrangements of the fuel alloy in the EBR-II reactor leading to a gain in reactivity may occur for very large temperature excursions. Pertinent experimental data from TREAT can be separated into roughly three broad categories. These are catalogued by the maximum clad temperature registered during an excursion while the fuel element is suspended in an inert atmosphere. The temperature ranges are:

1. 950 to 970°C; Incipient failure
2. 970 to 1015°C; Mild failure
3. Above 1015°C; Energetic failure.

In order to reach such clad temperatures in the reactor, some gross malfunction must have taken place beforehand. (Maximum fuel clad temperatures under normal conditions are expected to be between 480 and 560°C. Sodium boiling is initiated at 892°C.)

The "energetic failure" results suggest that in an accident where the temperature of the fuel element clad exceeds 1015°C, molten fuel is violently ejected from the stainless steel jacket at or near the point of initial jacket failure. The ejection is caused by very high pressure attributed to the vaporized sodium. In the usual TREAT experiment, a nearly constant axial

power distribution exists along the sample, and hence failure is equally probable anywhere along the pin; experiments tend to confirm this effect. Experiments were performed with a specially shaped axial power distribution⁽²⁴⁾ closely resembling that in the EBR-II. The results show that failure would occur at the highest localized temperature, possibly near the core midplane.

If operating conditions during an accident in the EBR-II reactor lead to maximum temperatures of the fuel element at or near the horizontal midplane of the core, and if the "energetic" type of failure does occur only near the hottest point, it may be possible that fuel alloy will be propelled from the axial extremities of the fuel elements toward the center plane, leading to an increase in reactivity.

There are several unknowns in this problem. Does the hot spot always appear near the core center in overheating conditions? Does the position of maximum axial temperature at the time of failure vary considerably as a function of core radius? These two unanswered questions depend very much on reactor design and the course of the accident. Further, what happens if one or both ends of the fuel pin are not fully melted by the time failure occurs at the hot spot? And what is the effect of pre-irradiation? Does swelling lead to a different mode of cladding failure? It is not clear that release of fission gases will serve to propel the fuel alloy from its container in the same way sodium does. Laboratory experiments⁽²⁶⁾ indicate a frothing action in the escape of uranium from preirradiated EFR pins.⁽²⁷⁾

The malfunction leading to major temperature rises may be caused by reactor power level being increased because of mechanical insertion of reactivity at a rapid rate. A more likely cause of temperature increase might be obstruction or loss of coolant flow to a subassembly or to a region within a subassembly.

The estimated reactivity effects of concentrating fuel at the central plane of the core have been determined by using 2-group diffusion theory calculations in r, z geometry. It is assumed in each case that fuel originally contained in a single subassembly remains within that subassembly. Thus, all fuel is dispersed throughout a central region about 11 cm thick, displacing all sodium and some stainless steel.

The first case assumes such failure takes place in the seven central subassemblies of the core (see Fig. 5). The calculated reactivity effect is as follows:

$$\Delta k = 0.030$$

$$\Delta k/\text{subassembly} \approx 0.0043$$

$$\Delta k/\text{fuel element} \approx 0.00044$$

The second case assumes such failure takes place in the fifth row of the reactor core (see Fig. 5), where the control subassemblies are located:

$$\begin{aligned}\Delta k &= 0.026 \\ \Delta k/\text{core subassembly} &\approx 0.0013 \\ \Delta k/\text{control subassembly} &\approx 0.00086 \\ \Delta k/\text{fuel element} &\approx 0.000014.\end{aligned}$$

The final investigation assumes complete failure within the 61-element core:

$$\begin{aligned}\Delta k &= 0.1443 \\ \Delta k/\text{fuel element (average)} &= 0.000028.\end{aligned}$$

The calculations are pessimistic. It is assumed that no void formation due to sodium boiling of flowing reactor coolant has occurred.

The investigation dealing with complete core failure was repeated by assuming that the sodium density in the core has been reduced by 50%:

$$\begin{aligned}\Delta k &= 0.1143 \\ \Delta k/\text{fuel element (average)} &= 0.000022.\end{aligned}$$

This latter result suggests that the magnitude of the reactivity effect for a particular mode of failure is a function of, but relatively insensitive to, the exact disposition of the core sodium.

The "mild" and "incipient failure" experiments also suggest rearrangement of fuel alloy subsequent to failure of a stainless steel jacket as a result of eutectic formation. However, the rate and manner in which such rearrangement takes place is less pronounced than in the case of "energetic failure" cited above. There is some suggestion from the "mild failure" experiments that clad penetration takes place at the bottom of the fuel element jacket. However, the axial temperature gradients in the EBR-II suggest that the experimental implication of failure at the bottom may not apply. In any event, it is the case of "energetic failure" which poses the major hazard on two counts

1. Failure is caused by localized coolant passage obstruction within a single subassembly. Under these conditions, if all 91 fuel elements were to fail simultaneously, the reactor would be on excursion, the magnitude depending on the location of the subassembly within the reactor core. For instance, in the control rod ring (Row 5) of the reactor, a 35-sec period is the result of fuel subassembly failure whereas a 65-sec period is the result of control subassembly failure. The period

trips will not terminate this malfunction. However, power level trips are operative at all times. Near the center of the core, complete subassembly failure (91 fuel elements) would exhibit a 3-sec period which would activate both period and power level trips before the excursion could proceed to other parts of the reactor.

2. Failure is caused by rapid overheating, such as primary pump failure coupled with freezing of control rods. Primary pump failure will normally scram the reactor. The likelihood of primary pump failure coupled with inability to scram the control rods is considered remote.⁽³⁾ This remote set of circumstances is being used as a basis for generating a large-scale meltdown accident. The determination of realistic rates of reactivity addition is one of the major objectives.

These speculations are based on rather recent and incomplete experimental results. The TREAT program will continue to investigate failure mechanisms of single pins and clusters of EBR-II fuel elements in both sodium and an inert atmosphere. Additional experiments as well as analytical studies are required to confirm or to disprove the assumption that the experimental results to date are characteristic of failure mechanisms in the actual reactor.

If further studies indicate the existence of failure modes capable of producing large reactivity gains, the basic question remains: What is the rate of reactivity addition? Preliminary estimates indicate that a small fraction of a second is needed for the fuel to travel from the end to the middle of the pin. In addition, there will likely be time delays of the order of a fraction of a second or more between the first pin failure and major pin failure in other subassemblies during a loss of coolant flow accident. It will be possible to determine the consequences of such fuel element failure only after all such effects are understood and considered in detail.

C. Core Meltdown

The two previous sections describe situations which might conceivably give rise to a nuclear event. The discussion of malfunction demonstrates how temperature excursions may be initiated. The implication of the TREAT experiments demonstrates how the core may lose its mechanical rigidity under temperature excursions of various kinds. The consequences of such core failure are determined by the rate at which reactivity is added to the system. Some rather large reactivity effects are implied by the TREAT experiments. Without knowing the rate at which reactivity is inserted, an estimate of a nuclear burst can be obtained by considering the following two problems.

1. Subsequent to loss of core integrity, what is an extremely rapid way of adding reactivity?
2. What are the mechanisms for limiting the extent and magnitude of the nuclear burst?

Detailed answers to these two questions were originally given by Jankus and Okrent,⁽²²⁾ and more recent analyses have been provided by Jankus.⁽²⁸⁾ Without implying any credence to the accident, the following pessimistic situation was postulated:

- a. The sodium has boiled away from the center of the core.
- b. The uranium from the middle of the core has trickled down into the lower part of the core and is retained there, producing a region abnormally dense in enriched uranium at the core bottom, with a large gap at the core center.
- c. At the worst possible moment, the upper portion of the core falls as a single unit, producing a prompt critical configuration at the highest possible rate of reactivity insertion.

After the excursion is initiated, the mechanism of reactivity reduction is caused by expansion of the reactor.⁽²²⁾ Table XX gives the predicted explosive energy yield for the EBR-II as a function of reactivity insertion rate for the postulated excursion.

Table XX

EXPLOSIVE ENERGY YIELD AS A FUNCTION
OF REACTIVITY INSERTION RATE

Reactivity Insertion Rate (\$/sec)	Explosive Energy Yield (lb of TNT)
200	370
600	850
1000	1240

The two highest rates of reactivity insertion cited require many coincidences in timing and fuel disposition prior to and during the excursion. If this pessimistic accident were realized, the \$200/sec is a more probable rate of reactivity insertion. There is considerable evidence⁽²¹⁾ that the primary containment system is capable of withstanding at least 400 lb of TNT. In addition, there is the comprehensive reactor plant containment. The latter is designed to withstand a maximum sodium-air reaction.

D. Reactor Stability and Kinetics

Operating experience with the three EBR-I cores (Mark I, II and III)⁽²⁹⁻³¹⁾ had some influence on the design of the EBR-II. The most important consideration dealt with the elimination of a prompt positive power coefficient of reactivity. This is accomplished by reducing or eliminating bowing of the fuel subassembly as well as bowing of the fuel within the subassembly. If bowing does exist, the design of the reactor is such that such motion is principally outward, thus giving a negative component to the power coefficient.

Basically, the fuel and blanket subassembly support structure makes the reactor a solid unit. However, clearances are necessary to effect insertion and removal of subassemblies. The subassemblies located in the grid-plate are fixed, with respect to radial motion, at the bottom with essentially no restraining forces at the top. If there are thermal gradients to induce bowing of the subassembly walls, the ultimate net motion is outward. There may be increments of inward motion at some low powers. Such motion, if any, is further restricted by the presence of protrusions on the subassembly walls which tend to take up the clearances that may exist.

The subassembly bowing phenomenon may be qualitatively examined. The subassembly grid support structure (see Fig. 4) is at constant temperature ($\sim 350^{\circ}\text{C}$). Radial subassembly motion pivots on the upper grid plate (see Fig. 4). Initially, the subassemblies are free to move at the top, near the outlet plenum (see Fig. 4). Bowing-induced motion will increase the core radius until the top end clearances are taken up. Further bowing will effectively reduce the core radius because subsequent motion is constrained at the top and the bottom of the subassembly. Reduction of the core radius will cease as soon as there is an effective contact between the subassembly and its inward neighbor. From this point on, all motion is outward, effectively increasing the core radius.

The fuel pins within the hexagonal subassembly are packed tightly through the use of helical spacer wires. Rigidity is further enhanced by indentations on the subassembly wall which restrict motion of the fuel pin where the spacer wires are not in contact with the wall.

It is not likely that a large delayed negative temperature coefficient is present in EBR-II. Such a coefficient in EBR-I is attributed to heating of structural components above the core. There are no such components in EBR-II.

Maximum bowing effects can occur in the control subassemblies. These effects are reduced by the incorporation of a flow "twister" just above the fuel elements. This tends to take the coolant from one side of the control subassembly and bring it to the other side. Thus, where the

coolant at higher temperature is in contact with the subassembly wall in the core, the coolant at lower temperature is in contact with the same wall just above the fuel element. High and low temperature here refers to temperature distribution within a single control subassembly.

Detailed stability analyses for the EBR-II have been reported.^(32,33) These studies find that the reactor exhibits no anomalous kinetic behavior during steady-state operation.

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