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EXPERIMENTAL BREEDER REACTOR-II (EBR-II)
SHIELD DESIGN

by

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ABSTRACT

A description of the EBR-II shield and the methods employed in arriving at the final design are presented. The major shield design problems for that reactor are enumerated and discussed.

I. INTRODUCTION

The EBR-II system is primarily an engineering facility to determine the feasibility of the fast-neutron breeder reactor for central-station power-plant application. It is a part of the Atomic Energy Commission's program for the development of power reactors. The EBR-II will be the first reactor in the United States to operate on a closed fuel cycle.

The primary objective of the EBR-II, aside from the fuel-cycle aspect, is to demonstrate the engineering feasibility of a large central-station-size plant of this type. Major objectives include achievement of high temperature of the reactor coolant at the outlet, a significant breeding gain, and the utilization of prototype components applicable to large-sized plants with little or no modification.

The calculations for the EBR-II shield design were begun in 1954, at which time computational methods used for shield design were not well established for any materials except possibly configurations of iron and water. The early computations for the EBR-II shield design were thus quite crude because of a lack of knowledge and experience. The final results, presented in this report, represent the advancement of the art during the period of the design of the reactor. They do not represent the ultimate in methods to be used on the same or similar reactors.

The purpose of this report is to describe the methods of attack on the shield design problems for this particular fast reactor. It will serve to record the results and make them available to those who may find them useful.

II. EBR-II DESCRIPTION

A. Plant Description

The Argonne Experimental Breeder Reactor II (EBR-II)^{1,2} is an unmoderated, heterogeneous, sodium-cooled reactor. The power output of the reactor is 62.5 Mw of heat, which generates 20 Mw of electricity through a conventional steam cycle. The EBR-II system is an integrated nuclear power plant, including a complete fuel-processing and fuel element-fabrication facility in addition to the reactor, the heat transfer systems, and the steam-electric plant. The plant is located at the National Reactor Testing Station in Idaho.

The preliminary design of this plant was described at the First International Conference on the Peaceful Uses of Atomic Energy in 1955.³ Although the basic design concept and objectives have remained unchanged, many modifications have been made in the detailed design, including the separation of the facility into several buildings and providing containment for the reactor system.

The general arrangement of the buildings in the EBR-II facility is shown in Fig. 1. The reactor and primary sodium coolant system are housed in the Reactor Plant (see Figs. 2 and 3), which is a steel containment vessel 80 ft in diameter and approximately 140 ft high. The power plant contains the turbine, generator, and the associated steam and electrical equipment. It also contains the control room for the reactor. The Fuel Cycle Plant (see Fig. 4) contains the facilities for processing and fabricating the fuel. The Sodium Boiler Plant houses the sodium-to-water steam-generating equipment. In this building, sodium is received, stored, processed, and pumped into the primary or secondary systems. The buildings are interconnected by sodium or steam lines; however, the design guarantees that no sodium reaches the Power Plant and no primary sodium reaches the Sodium Boiler Plant.

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¹Koch, L. J., et al., Construction Design of EBR-II, Proceedings of Second UN International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 9, 323 (1958).

²Koch, L. J., et al., Hazard Summary Report on EBR-II, ANL-5719, (May 1957); see also ANL-5719-Addendum (1962).

³Barnes, A. H., et al., The Engineering Design of EBR-II, Proceedings of the First International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 5, 330 (1955).

Fig. 1. EBR-II Plant Arrangement

Fig. 2. Reactor Building

Fig. 3. Reactor Plant - Operating Floor and Balcony Equipment Layout

Fig. 4. Fuel Cycle Facility

B. Reactor Description

The EBR-II reactor (see Figs. 5 and 6), the primary coolant system, the control and safety drive systems, parts of the fuel-handling system, and other associated components are located within the double-walled Primary Tank, which is 26 ft in diameter and 26 ft deep. The tank contains liquid sodium which acts as the primary coolant for the reactor. This primary sodium is drawn into a plenum below the reactor, pumped through the core, and then through the primary heat exchanger. This unit is also immersed in the primary sodium. After giving up heat to the secondary sodium, the primary coolant returns directly to the bulk sodium in the primary tank.

Fig. 5. Primary System

Fig. 6. Fuel Handling System Principal Mechanisms

The reactor core is made up of fuel pins containing enriched uranium, jacketed with stainless steel, and bound in clusters of ninety-one. The core is surrounded by blanket rods of depleted uranium. Control of the reactor is achieved by vertical motion of special fuel assemblies. The core is very compact and the blanket is heavily loaded with uranium to capture a maximum number of leakage neutrons. For analysis, the reactor is divided into several regions which will be referred to later. The control rod region has the movable fuel assemblies which control the reactor. (There are also two safety rods which are not specifically referred to in the shielding analysis.) The inner blanket is structurally similar to the core but contains depleted rather than enriched uranium. The core and blanket are contained in the reactor vessel. The neutron shield, which is located between the blanket and the primary sodium, is partly inside and partly outside of this reactor vessel.

C. Shield Description

There are two major shielding regions in the EBR-II Reactor. The neutron shield, between the blanket and the bulk sodium, serves to reduce the neutron flux leakage into the bulk sodium, thus keeping activation of the components and of the secondary sodium to a minimum. It also serves to some extent as a reflector, conserving a relatively small number of neutrons. The biological shield, which is located around the sodium tank, keeps radiation levels within acceptable limits at the humanly accessible locations around the reactor.

1. Neutron Shield

The many components in the sodium tank, pumps, heat exchangers, etc. (see Figs. 5 and 6), must not be permitted to become excessively radioactive; therefore, a neutron shield is needed between the outer blanket and the bulk sodium (see Figs. 6 and 7). The purpose of this shield is to moderate and absorb the neutrons that escape from the blanket before they get into the sodium tank, where they may be captured by the steel in various components or by the secondary sodium as it passes through the heat exchanger.

Boron and carbon were selected for the neutron shield. These are found in a convenient compound, boron carbide, but, because its cost is high and boron is not needed in such large quantities, 3% boron carbide (by weight) in graphite was selected for the boron-containing material. Hydrogenous materials, which make the most efficient neutron shields, are ruled out by virtue of the high-temperature incompatibility with sodium and considerations of reactor safety.

An early design of the neutron shield involved six rows of stainless steel cans, each 4-in. square, filled with graphite. Three per cent boron carbide was included in the fourth and sixth rows. In the final design, the number of rows was increased to seven, with the first two rows located inside the reactor vessel and the remainder outside. The arrangement is shown in Fig. 8.

NOTE: FOR PLAN VIEW SEE FIG. 4

Fig. 7. Reactor and Neutron Shield

Fig. 8. Neutron Shield

The fifth and seventh rows are borated, although the boron carbide has been omitted in areas immediately adjacent to the neutron-detecting instruments in order to increase their response. Figure 9 shows the plan of the primary tank, indicating locations of nuclear instrument windows and other components.

The heat exchanger (see Fig. 10) is surrounded by boron steel to reduce the number of neutrons captured in the secondary sodium. Since the secondary sodium must leave the Reactor Building, the activity of this sodium must be kept below the level that requires the sodium pipe to have a gamma-ray shield. A shell of borated steel nominally one-inch thick and having 1.5% boron concentration was recommended to surround the heat exchanger.

2. Biological Shield

The radial shield beyond the sodium tank does not need to be a particularly efficient shield. Space is not at a premium, and there are no areas located radially around the reactor that must be occupied continuously during operation. The large amount of sodium around the reactor, approximately 7 ft thick, does not serve as an efficient shield; however, it must be considered as a significant portion of the shield because there is such a large amount of it. The remaining portion of the radial shield is not a critical part of the shield design in material or thickness. The primary function of the biological shield is to attenuate gamma radiation from the bulk sodium and from neutron captures in the neutron shield and reactor vessel. Thus, the material which was chosen for this shield is ordinary concrete (see Fig. 11) and the thickness is a somewhat arbitrary 6 ft.

The shield above the core also consists of a large amount of sodium, as may be deduced from Fig. 6. Space, or building height, was at a premium in this direction; therefore a more efficient shield was desired. The basic material chosen for this shield beyond the sodium was heavy concrete. The layer of material directly above the sodium tank could not be heavy concrete because of temperature limitations and, therefore, is made up of steel balls. The decision to use steel balls in this region did not come about directly. As the design of the reactor progressed and it became clear that it would be necessary to protect the heavy concrete from excessive heating, the lower section of the top shield was first specified to be layers of graphite and steel. As a result of the large number of penetrations in the top shield (see Fig. 12), it soon became obvious that the laminar construction was not economical, nor was it simple. Therefore, the lower region of the top shield was specified to be of steel balls and powdered graphite. At this time, more detailed calculations of the neutron flux in the sodium indicated that the original estimates of the fluxes were quite conservative. It was, therefore, decided to omit the powdered graphite.

Fig. 9. Location of Nuclear Instrument Thimbles

Fig. 10. Primary Heat Exchanger

Fig. 11. Blast Shield and Biological Shield

Fig. 12. Top Shield Penetrations

In any reactor, a significant portion of the material surrounding the core is chosen and placed for reasons other than those based on radiation shielding. Nuclear, mechanical, and structural requirements leave only certain regions which may be specified by the shield designer. In the analysis, however, each layer of material, regardless of its primary function, is treated as a component of the shield. Tables I, II, and III give the nominal centerline thicknesses and compositions of the regions in the radial, axial upward, and axial downward directions from the core, respectively. The basis of the material thicknesses given is that of the reactor as constructed. Most of the neutron calculations applied to the shield design were based on an earlier configuration. Because the reader may like to compare the results of calculations with the exact composition used, the material thicknesses based on the early configurations are given in Tables IV, V, and VI.

Table I
FINAL RADIAL SHIELD COMPOSITION

Region	Thickness (cm)	Outer Radius (cm)	Vol-%				
			Fuel Alloy	Sodium	Steel	Carbon	C+ 3 wt-% B ₄ C
Core	18.82	18.82	31.8	48.7	19.5	-	-
Control Rod Ring	5.36	24.18	21.3	57.9	20.8	-	-
Inner Blanket	10.69	34.87	60.0	22.4	17.6	-	-
Outer Blanket	43.24	78.11	60.0	22.4	17.6	-	-
Inner Neutron Shield	33.65	111.76	-	33.1	10.0	53.5	-
Vessel Region	9.208	120.97	-	58.6	41.4	-	-
Outer Neutron Shield	57.942	178.91	-	8.33	9.59	46.3	30.9
Sodium	217.29	396.20					
Steel	1.27	397.47					
Gap	12.70	410.17					
Steel	1.27	411.44					
Steel Wool	15.24	426.68					
Gap	17.78	444.46					
Steel	0.64	445.10					
Gap	10.16	455.26					
Blast Shield	67.95	523.21					
Ordinary Concrete	182.88	706.09					

Table II
FINAL AXIAL SHIELD COMPOSITION ABOVE THE CORE

Region	Thickness (cm)	Outer Radius (cm)	Vol-%				
			Fuel	Sodium	Steel	C+ 3 wt-% B ₄ C	Gas
Core	18.06	18.06	31.8	48.7	19.5		
Gap	8.89	26.95		55.7	29.6		14.7
Gap	2.21	29.16		43.1	56.9		
Blanket	45.72	74.88	32.0	47.06	20.4		
Gap	11.59	86.47		58.9	30.0		11.1
Plenum	60.96	147.4		77.6	21.1		
Steel	5.72	153.2					
Neutron Shield	71.63	224.8		9.0	9.6	77	
Steel	3.33	228.1					
Sodium	290.8	518.9					
Argon	45.72	564.6					
Steel	5.08	569.7					
Steel Balls	71.12	640.8					
Air	5.08	645.9					
Steel	1.27	647.2					
Foam Glass	15.24	662.4					
Air	13.34	675.8					
Steel	5.08	680.8					
Heavy Concrete (280 lb/ft ³)	121.9	802.7					
Steel	7.62	810.3					

Table III

FINAL AXIAL SHIELD COMPOSITION BELOW THE CORE

Region	Thickness (cm)	Outer Radius (cm)	Vol-%			
			Fuel or Alloy	Sodium	Steel	Gas
Core	18.06	18.1	31.8	48.7	19.5	-
Gap	5.23	23.3		58.7	41.3	-
Gap	4.78	28.1		51.8	26.1	22.1
Blanket	45.72	73.8	32.0	47.6	20.4	-
Gap	6.68	80.5		73.9	26.1	-
Steel	5.99	86.4				-
Sodium Steel	32.10	119		78	22	-
Steel	10.16	129				
Sodium	45.70	175				
Steel	6.35	181				
Sodium	35.56	216				
3% Boron Steel Plate	1.27	218				
Sodium	7.62	225				
3% Boron Steel Plate	1.27	227				
Sodium	7.62	234				
Steel Tank Bottom	3.81	238				
Gap	10.16	248				
Steel Plate	1.91	250				
Rib	24.13	274				
Steel Plate	1.91	276				
Insulation (Steel Wool)	15.24	291				
Gap	20.32	312				
Steel Plate	0.64	312				
Rib	15.24	327				
Gap	10.80	338				
Steel	1.27	340				
Ordinary Concrete	396.24	736				

Table IV

PRELIMINARY RADIAL SHIELD COMPOSITION

Region	Thickness of Region		Outer Radius		Volume %				
	in.	cm	in.	cm	U Alloy	Na	Steel	C	C + 3 wt % B ₄ C
Core	7.41	18.82	7.41	18.82	31.8	48.7	19.5	-	-
Control Rod Ring	2.11	5.36	9.52	24.18	21.3	57.9	20.8	-	-
Inner Blanket	4.21	10.69	13.73	34.87	60.0	22.4	17.6	-	-
Outer Blanket	17.02	43.24	30.75	78.11	60.0	22.4	17.6	-	-
Vessel Region	7.12	18.1	37.87	96.21	-	70.2	29.8	-	-
Gap	2.00	5.08	39.87	101.27	-	100	-	-	-
Neutron Shield	27.09	68.81	66.96	170.08					
Canned Graphite	13.64	34.65	53.51	135.92		10.8	5.0	79.3	
Canned C + 3% B ₄ C	4.54	11.53	58.05	147.45		9.9	5.1	-	80.
Canned Graphite	4.54	11.53	62.59	158.98		8.9	5.1	81.0	
Canned C + 3% B ₄ C	4.37	11.10	66.96	170.10		5.6	5.3	-	84.
Sodium	89.04	226.16	156.00	396.2					
Steel	0.50	1.27	156.50	397.47					
Gap	5.00	12.70	161.50	410.17					
Steel	0.50	1.27	162.00	411.44					
Insulation (Steel Wool)	6.00	15.24	168.00	426.68					
Gap	7.00	17.78	175.00	444.46					
Steel	0.25	0.64	175.25	445.10					
Gap	4.00	10.16	179.25	455.26					
Blast Shield	26.75	67.95	206.00	523.21					
Steel	0.50	1.27	179.75	456.53					
Concrete ($\rho = 0.25$)	4.00	10.16	183.75	466.69					
Steel	0.50	1.27	184.25	467.96					
Concrete ($\rho = 0.25$)	4.00	10.16	188.25	478.12					
Steel	0.50	1.27	188.75	479.39					
Concrete ($\rho = 0.25$)	4.00	10.16	192.75	489.55					
Steel	0.50	1.27	193.25	490.82					
Concrete ($\rho = 0.25$)	4.00	10.16	197.25	500.98					
Steel	0.50	1.27	197.75	502.25					
Glass Wool	3.62	9.21	201.37	511.46					
Steel	0.50	1.27	201.87	512.73					
Glass Wool	3.62	9.21	205.49	521.94					
Steel	0.50	1.27	205.99	523.21					
Ordinary Concrete	72.00	182.88	277.99	706.09					

Table V

PRELIMINARY AXIAL SHIELD COMPOSITION ABOVE THE CORE

	Thickness of Region		Outer Radius, cm	Volume %				
	in.	cm		Fuel or Alloy	Na	Steel	Graphite + wt-% B ₄ C	Gas
Core	7.11	18.06	18.06	31.8	48.7	19.5		
Gap	3.50	8.89	26.95		55.7	29.6		14.7
Gap	.87	2.21	29.16		43.1	56.9		
Blanket	18.00	45.72	74.88	32.0	47.6	20.4		
Gap	4.563	11.59	86.47		58.9	30.0		11.1
Plenum	24.00	60.96	147.4		77.6	21.1		
Steel	2.25	5.72	153.2					
Neutron Shield	28.2	71.63	224.8					
Canned C + B ₄ C	4.54	11.53			10.0	5.0	80.0	
Canned C + B ₄ C	4.54	11.53			10.0	5.0	80.0	
Canned C + B ₄ C	4.54	11.53			10.0	5.0	80.0	
Canned C + B ₄ C	4.54	11.53			10.0	5.0	80.0	
Canned C + B ₄ C	4.54	11.53			10.0	5.0	80.0	
Canned C + B ₄ C	4.54	11.53			10.0	5.0	80.0	
Steel	1.31	3.33	228.1					
Sodium	114.5	290.8	518.9					
Argon	18.0	45.72	564.6					
Steel	1.5	3.81	568.4					
Insulation	6.00	15.24	583.7					
Gap	6.00	15.24	598.9					
Steel	1.85	4.80	603.6					
Graphite	4.00	10.16	613.8					
Steel	4.00	10.16	623.9					
Graphite	4.00	10.16	634.1					
Steel	4.00	10.16	644.3					
Graphite	4.00	10.16	654.4					
Steel	4.00	10.16	664.6					
Graphite	4.00	10.16	674.7					
Steel	4.00	10.16	684.9					
Graphite	4.00	10.16	695.1					
Steel	3.75	9.52	704.6					
Heavy Concrete	37.00	93.98	798.6					
Steel	2.00	5.08	803.6					

Table VI

PRELIMINARY AXIAL SHIELD COMPOSITION BELOW THE CORE

	Thickness of Region		Outer Radius, cm	Volume %					
	in.	cm		Fuel or Alloy	Na	Steel	C	C + 3 wt-% B ₄ C	Gas
Core	7.11	18.06	18.06	31.8	48.7	19.5	-	-	-
Gap	2.06	5.232	23.29		58.7	41.3	-	-	-
Gap	1.88	4.775	28.07		51.8	26.1	-	-	22.1
Blanket	18.00	45.72	73.79	32.0	47.6	20.4	-	-	-
Gap	2.63	6.68	80.47		51.8	26.1			
Steel	2.36	5.994	86.46						
Sodium-Steel	12.64	32.11	118.6		78	22			
Steel	4.00	10.16	128.7						
Sodium	15.00	38.1	166.8						
Steel	2.50	6.35	173.2						
Neutron Shield	21.0	53.34	226.5						
Canned Graphite	4.54	11.53			10.8	5.0	79.3	-	-
Canned Graphite + B ₄ C	4.54	11.53			9.9	5.1	-	80.1	-
Canned Graphite	4.54	11.53			10.8	5.0	79.3	-	-
Canned C+3% B ₄ C	4.54	11.53			9.9	5.1	-	80.1	-
Steel	2.00	5.08	231.6						
Gap	4.00	10.16	241.8						
Steel	.500	1.27	243.0						
Gap	10.00	25.4	268.4						
Steel	.500	1.27	269.7						
Insulation (Steel Wool)	6.00	15.24	284.9						
Gap	8.00	20.32	305.3						
Steel	.250	.635	305.9						
Rib	6.00	15.24	321.1						
Gap	4.25	10.80	331.9						
Steel	.500	1.27	333.2						
Ordinary Concrete	156.0	396.24	729.4						

III. RADIATION CALCULATIONS

A number of preliminary calculations were performed during the early design stages in order to determine composition and thickness requirements of specific critical parts of the shield and to attempt to predict the radiation levels in various parts of the system. Some of these calculations were published in the progress reports of the Reactor Engineering Division.^{4,5,6} Many results were reported as internal memoranda. Most of these reports have little more than historical value because of design changes and new analysis techniques.

A. Neutron Sources

The primary source of radiation from a reactor is from nuclear fission in the core. A knowledge of the secondary gamma rays is quite essential but is still closely related to the fast-neutron leakage from the core, which therefore must be known. A blanketed fast reactor, such as EBR-II, has an essential difference from the more familiar reflected but unblanketed thermal reactor. That is, most of the fast-neutron leakage into the shield comes from the blanket rather than the core. This makes knowledge of the fission distribution in the blanket most essential. Since the fission distribution in the EBR-II blanket is not a constant, and for that matter not very amenable to being represented as a constant, the problem of computing the fast-neutron leakage is made appreciably more complicated than for the typical thermal reactor.

A representative power distribution in the core and blanket of EBR-II is illustrated in Fig. 13, from which it may be seen that the spatial power distribution in the blanket can be approximated by an exponential function. The power distribution given in Fig. 13 was computed by means of the UNIVAC RE-7 diffusion code⁷ with twenty energy groups of neutrons (see Table VII). The cross sections employed were those reported by Loewenstein.⁸

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⁴Grotenhuis, M., and A. J. Ulrich, "Neutron Shield," Reactor Engineering Division Quarterly Report, ANL-5371 (Jan 1955), p. 247.

⁵Fricke, E. F., "Shielding Analysis," Reactor Engineering Division Quarterly Report - Section I October, November, December, 1955, ANL-5561 (April 1956), p. 155.

⁶Grotenhuis, M., H. I. Kraig, and A. E. McCarthy, "EBR-II Shield Design Calculations," Reactor Engineering Division Quarterly Report - Section I, ANL-5571 (July 1956), p. 146.

⁷Butler, M. K., and J. M. Cook, UNIVAC Program for the Solution of Multigroup Reactor Equations, ANL-5437 (1954). (RE-122 is the same program executed on the IBM-704.)

⁸Loewenstein, W. B., private communication.

Fig. 13. Power Distribution in the Core and Blanket

Table VII

NEUTRON ENERGY GROUPS FOR TWENTY-GROUP PROBLEMS

Group No	Energy Band	Fraction of Fission Neutrons Born in Band	Group No.	Energy Band	Fraction of Fission Neutrons Born in Band
1	- 1.353 Mev	0.573	10	300 - 3350 ev	0
2	1.353 - 0.1979 Mev	0.295	11	90 - 300 ev	0
3	0.4979 - 0.3020 Mev	0.065	12	43 - 90 ev	0
4	0.3020 - 0.1832 Mev	0.034	13	34 - 43 ev	0
5	0.1832 - 0.1111 Mev	0.017	14	22 - 34 ev	0
6	0.1111 - 0.0674 Mev	0.008	15	18 - 22 ev	0
7	0.0674 - 0.0248 Mev	0.006	16	9 - 18 ev	0
8	0.0248 - 0.00912 Mev	0.002	17	6 - 9 ev	0
9	0.00912 - 0.00335 Mev	0	18	0.7 - 6 ev	0
			19	0.085 - 0.7 ev	0
			20	0 - 0.085 ev	0

Calculations of the fast-neutron leakage by means of removal cross sections were performed in terms of the above source distribution. Although removal cross sections are not truly applicable in this situation, better procedures were not readily available at the time. Later work showed that inaccuracies were not great enough to change the conclusions. The results of the early calculations are presented in Table VIII.

Table VIII
EBR-II FAST NEUTRON SOURCES

Source Region	Power (Mw)	Relative Neutron Flux at the Neutron Shield
Core	32	4.7
Control Rod Ring	20	6.6
Inner Blanket	6	2.3
Outer Blanket 1 } Outer Blanket 2 }	4	{ 107 122

It will be noted that the core itself is not the primary source of fast-neutron leakage by at least an order of magnitude. Nearly all the fast neutrons leaking from the blanket were from the outer 20 cm of the blanket. The neutrons fluxes existing at deep penetrations of the shield are directly dependent on this fast-neutron leakage, which shows the importance of knowing the fission or power distribution in the blanket.

It should be recalled that the fission distribution in the blanket is not a constant quantity with time. It changes as the plutonium content builds up and as the blanket subassemblies are moved to new locations or replaced by new ones. It is estimated, however, on the basis of core and blanket physics studies⁹ that the changes thus induced in the EBR-II blanket will not materially affect the shield design, provided that highly loaded subassemblies are not deliberately placed in positions near the outer part of the blanket.

B. Neutron Fluxes

When the design of the EBR-II had progressed to the point where it was relatively final, it was time to compute the neutron distributions in the shield as completely as possible for that configuration of materials. To do this adequately, multigroup procedures were essential. At that time reliable multigroup problems had been done for core and blanket physics calculations. These were utilized to describe the power distribution in the core and blanket in order to provide a source for the deep-penetration problems. The calculations for the twenty-energy-group core were not suitable for deep-penetration computations because of limitations in the program itself. Other available multigroup diffusion codes were limited as to number and dimension of the regions, boundary conditions, and methods of slowing down. The calculation of neutron

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⁹Loewenstein, W. B., private communication.

distributions out through the sodium tank was subcontracted to Nuclear Development Corporation of America (NDA).[†] The multigroup diffusion code known as STAR was utilized for these calculations.

A summary of the deep-penetration problems requested from NDA is given in Table IX. The ten-neutron-energy-group structure used is given in Table X. The problems were based on an idealized spherical geometry with the shields having thicknesses that matched those on the appropriate centerline direction from the core. Some of the regions were homogenized for convenience. The materials and thicknesses on which the ten-group STAR problems were based are those given in Tables IV, V, and VI.

Table IX

A SUMMARY OF THE TEN-GROUP NEUTRON-FLUX PROBLEMS

A. Borated* neutron shield problems

1. Axial centerline above the core
2. Axial centerline below the core
3. Radial centerline
  - a. Neutron shield followed by sodium
  - b. Neutron shield followed by sodium and the heat exchanger with a borated** steel wall
  - c. Neutron shield followed by sodium and the heat exchanger with a nonborated steel wall

B. Nonborated neutron shield

1. Axial centerline above the core
2. Axial centerline below the core
3. Radial centerline
  - a. Neutron shield followed by sodium
  - b. Neutron shield followed by sodium and the heat exchanger with a borated** steel wall
  - c. Neutron shield followed by sodium and the heat exchanger with a nonborated steel wall

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*Boration: 0 - 0.6 - 30% B₄C (by weight)

**Boration: 1.25% (by weight)

Table X

NEUTRON ENERGY GROUPS FOR THE TEN-GROUP PROBLEMS

Group	Energy Range	Group	Energy Range
1	10 Mev - 1.32 Mev	6	24.5 kev - 3.35 kev
2	1.32 Mev - 0.5 Mev	7	3.35 kev - 454 ev
3	500 kev - 180 kev	8	454 ev - 8.4 ev
4	180 kev - 67 kev	9	8.4 ev - 0.27 ev
5	67 kev - 24.5 kev	10	0.27 ev - 0

The most critical use for the neutron fluxes thus determined was to estimate the secondary sodium activation. Representative problems were included in which the heat exchanger was assumed to be at the midplane of

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[†]Now United Nuclear Corporation, White Plains, New York.

the reactor. It is actually located appreciably higher than this, and therefore the assumption was considered conservative insofar as secondary sodium activation is concerned. There are two regions where boron could be added to reduce the activation of secondary sodium: the graphite neutron shield, and a boron-steel shell surrounding the heat exchanger. As seen in Table IX, calculations were run with various concentrations of boron in the neutron shield, and both with and without the boron boundary around the heat exchanger. From these calculations the value of the boron can be assessed with respect to secondary sodium activation.

A summary of the results of the calculations for the NDA ten-group neutron attenuation is given in Figs. 14 through 20. A brief description and the purpose of each figure follows.

Figure 14 gives the general magnitude of the neutron fluxes in the radial shield through the heat exchanger. The neutron energy groups listed in Table X have been condensed into three groups as follows: the sum of groups one through five is labeled fast flux, the sum of groups six through nine is called intermediate flux, and group ten is referred to as slow flux. Combining groups in this manner permits all the radial neutron fluxes to appear on one figure. It can be seen that the neutron fluxes at the outer diameter of the sodium tank are quite low.

The absorption of neutrons is largely a thermal or low-energy neutron process. In order to estimate the value of boron in the neutron shield and heat exchanger with respect to the neutron absorptions in the heat exchanger, the behavior of the thermal neutron flux will suffice. Figure 15 compares the slow-neutron flux (group 10) for the three cases: borated neutron shield, borated heat exchanger wall; unborated neutron shield, borated heat exchanger wall; and unborated neutron shield, unborated heat exchanger wall. Two nominal boron concentrations in the neutron shield were chosen for analysis: 0.6% and 3% boron carbide, by weight.

More exact calculations of the sodium activation require that epithermal neutron absorptions must be added to those that occur at thermal neutron energy. The total sodium capture distribution in the heat exchanger is shown in Figs. 16 and 17. These curves show the comparative results of borating the neutron shield and the heat exchanger wall. Figure 16 compares the sodium capture distributions in the borated heat exchanger for the cases of borated and unborated neutron shield. Figure 17 shows the relative effect of borating the neutron shield for the case of an unborated heat exchanger wall.

A general view of the neutron fluxes in the axial shield above the reactor is given in Figs. 18 and 19. The energy groupings are the same as for Fig. 14.

Fig. 14. Radial Three Group Neutron Flux; 3% Borated Neutron Shield and Unborated Heat Exchanger Wall

Fig. 15. Radial Slow Flux

Fig. 16. Total Sodium Captures in the Heat Exchanger with a Borated Wall

Fig. 17. Total Sodium Captures in the Heat Exchanger with an Unborated Wall

Fig. 18. Axial Neutron Fluxes Above the Core

These figures give the magnitude of the neutron fluxes in the axial direction and illustrate the neutron-attenuating properties of sodium. In addition, the magnitude of the neutron fluxes at the upper sodium surface are seen to be quite low. Where necessary, the stainless steel structure in these regions was taken into account by assuming that the appropriate volume fraction was mixed with the sodium. The steel was assumed to be pure iron for calculational purposes.

The neutron fluxes in the radial concrete shield are given in Fig. 20. The two neutron-energy groups represent the results of a calculation by removal theory in conjunction with a one-group diffusion calculation to determine a thermal flux. This was considered adequate treatment in view of the fact that the material is hydrogenous and that the neutron fluxes were quite low at the beginning of the concrete and therefore a less crucial quantity. The fast flux outside the shield through 6 ft of concrete is $8.8 \times 10^{-3} \text{ n}/(\text{cm}^2)(\text{sec})$.

Fig. 19
Axial Neutron Fluxes
Above the Core; 0.6%
Borated Neutron
Shield

Fig. 20
Radial Neutron Fluxes
in Concrete

Figures 21 and 22 show the neutron fluxes in the axial direction below the core. The calculations for Fig. 21 were done by the RE-122 diffusion code with the use of twenty neutron-energy groups. They differ from the ten-group STAR problems for the same location in that they represent the final material arrangement given in Table III. The two neutron-flux curves given in Fig. 21 are the fast-neutron flux, a sum of the neutrons in groups one through seven of Table VII, and the thermal-neutron flux, group twenty. The fast-neutron flux is comparable with that given in Fig. 14. The thermal flux, however, is forced down in the outer steel region, which in this case formed the outer boundary for the calculation. In reality, neutrons slow down and scatter back from the concrete, yielding a thermal-neutron flux peak at the outer side of the steel region.

Fig. 21. Axial Neutron Fluxes Below the Core

Figure 22 shows the neutron fluxes extended through the concrete below the sodium tank. They were extended by removal theory and a one-group diffusion problem. As was the case for the neutron fluxes in the radial

concrete shield, this procedure was deemed adequate for this region since the material was hydrogenous and the fluxes too low to be of great importance. The fast flux through 9 ft of concrete is $5 \times 10^{-2} \text{ n}/(\text{cm}^2)(\text{sec})$.

Fig. 22. Axial Neutron Fluxes Below the Core

C. Gamma-ray Sources

The gamma rays in the EBR-II originate from the processes of fission, neutron capture, and radioactive decay, as is normally true in any reactor. Because the EBR-II is a uranium-blanketed fast reactor and because it is contained in a huge tank of sodium, one cannot assume that the relative importance of the gamma-ray sources and the degree of difficulty in calculating these sources is the same as for most other reactors.

The fission gamma-ray source follows the fission distribution. However, like the fast-neutron leakage, the gamma-ray leakage is mainly from the outer portion of the blanket. This makes the estimation of the fission gamma-ray source more difficult than for thermal reactors.

Fortunately, these gamma rays are not important far out in the shield, so that a detailed knowledge of the fission distribution is not as important for determining the effective gamma-ray source as it was for the fast-neutron source.

In EBR-II, the hard, capture gamma rays from the blanket, vessel, and neutron-shield region penetrate the entire radial shield and thus are an important consideration. Many of the neutron captures occur while the neutrons are at intermediate energy. This fact makes the capture gamma-ray source a difficult quantity to predict and is one reason that multigroup procedures are required for the calculation of neutron flux.

Radioactive decay gamma rays are usually not high enough in energy, nor are they profuse enough, to make them a serious consideration in the design of a biological shield. However, the huge mass of radioactive sodium around the EBR-II makes these gamma rays from the primary sodium a factor in the shield design. They are instrumental in setting the thickness of the biological shield as well as in other shield problems dealing with shutdown or fuel-transfer operations.

The gamma-ray fluxes from all the sources must be known in order to compute radiation heating in the vessel and neutron-shield region, and for the design of radiation-detection instruments.

D. Gamma-ray Fluxes

The calculation of the gamma-ray fluxes in EBR-II was done by the usual procedures. The formulas applied were those derived from the integration of the point kernel with linear buildup:

$$\Phi = \int dV Q \frac{\exp(-\mu R)}{4\pi R^2} B(\mu R) \quad , \quad (1)$$

where

$$B(\mu R) = 1 + k\mu R \quad .$$

The value of k in all cases was selected to be one. This is generally conservative according to the computed values of Goldstein and Wilkins.¹⁰

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¹⁰ Goldstein, H., and J. E. Wilkins, Jr., Calculations of the Penetration of Gamma Rays, NYO-3075 (June 30, 1954).

The results of integrating Eq. 1 for various source geometries and distributions are reported in the literature.^{11,12} A formula was selected for each gamma-ray problem to represent properly the source geometry and distribution. The gamma-ray source spectrum is represented by a series of appropriately weighted line spectra. The lines selected for the EBR-II problems were at 1, 2, 4, 6, and 8 Mev.

The gamma-ray fluxes for EBR-II are given in Figs. 23 and 24.

Fig. 23. Radial Gamma-ray Flux

¹¹ Rockwell, T., II, Ed. Reactor Shielding Design Manual, TID-7004 McGraw-Hill and D. Van Nostrand Book Co. (March 1956).

¹² Grotenhuis, M., Lecture Notes on Reactor Shielding, ANL-6000 (March 1959).

Fig. 24. Axial Gamma-ray Flux Above the Core

These are intended for illustrative purposes only, since neither the energy of the photons nor the source is given. A more detailed account of the gamma-ray fluxes is given in Tables XI and XII. From Table XI it can be seen that the 2-Mev gamma rays from neutron capture in the blanket dominate at the outside edge of the outer blanket but that the spectrum hardens because of captures in the neutron-shield regions to the point that 8-Mev photons are the most numerous. The gamma rays from the radioactive decay of sodium become the strongest source out through the sodium tank and as far as the blast shield. The more penetrating 8-Mev capture photons from the neutron-shield region dominate again at the outside of the concrete. The gamma-ray flux outside the radial shield through 183 cm of concrete is $3 \times 10^3 \text{ Mev}/(\text{cm}^2)(\text{sec})$, or about 5 mr/hr. The gamma-ray flux on top of the reactor through 122 cm of heavy concrete is $4 \times 10^{-5} \text{ Mev}/(\text{cm}^2)(\text{sec})$.

Table XI
Part I
RADIAL GAMMA-RAY FLUXES

Region	Source	Outer Radius (cm)	Flux, Mev/(cm ²)(sec)				
			1 Mev	2 Mev	4 Mev	6 Mev	8 Mev
Core	Fission and fission products	18.9	2.17×10^{14}	2.00×10^{14}	1.72×10^{13}	3.44×10^{13}	
	Total		4.69×10^{14}				
Rod	Core fission and fission products	24.3	1.32×10^{13}	3.04×10^{13}	3.23×10^{12}	6.16×10^{12}	
	Rod region fission and fission products		1.65×10^{14}	1.32×10^{14}	1.46×10^{13}	2.97×10^{13}	
	Inner blanket capture gamma rays		1.10×10^{13}	6.70×10^{13}	2.10×10^{13}	2.55×10^{11}	5.56×10^{12}
			1.89×10^{14}	2.29×10^{14}	3.88×10^{13}	3.61×10^{13}	5.56×10^{12}
	Total		4.98×10^{14}				
Inner Blanket	Core and rod region fission and fission products	34.9	9.62×10^9	3.04×10^{11}	6.11×10^{10}	9.20×10^{10}	
	Inner blanket fission and fission products		3.54×10^{12}	4.07×10^{12}	3.83×10^{11}	7.39×10^{11}	
	Inner blanket capture gamma rays		1.10×10^{13}	6.70×10^{13}	2.10×10^{13}	2.55×10^{11}	5.56×10^{12}
	Outer blanket capture gamma rays		6.92×10^{12}	4.23×10^{13}	1.26×10^{13}	9.80×10^{10}	3.25×10^{12}
			2.14×10^{13}	1.14×10^{14}	3.40×10^{13}	1.18×10^{12}	8.81×10^{12}
	Total		1.79×10^{14}				
Outer Blanket	Outer blanket fission and fission products	78.1	4.20×10^{10}	4.28×10^{10}	3.99×10^9	7.55×10^9	
	Outer blanket capture gamma rays		1.10×10^{10}	7.31×10^{11}	2.32×10^{11}	1.71×10^9	5.72×10^{10}
	Inside neutron shield capture gamma rays		4.72×10^9	1.42×10^{10}	6.92×10^{10}	9.18×10^{10}	2.41×10^{11}
			5.78×10^{10}	7.88×10^{11}	3.05×10^{11}	1.01×10^{11}	2.98×10^{11}
	Total		1.55×10^{12}				
Inside Neutron Shield	Attenuated blanket gamma rays	111.8	6.30×10^8	1.84×10^{10}	2.07×10^{10}	1.09×10^9	7.63×10^9
	Neutron shield capture gamma rays		4.72×10^9	1.42×10^{10}	6.92×10^{10}	9.18×10^{10}	2.41×10^{11}
	Vessel capture gamma rays		1.98×10^9	9.70×10^9	4.80×10^{10}	6.48×10^{10}	1.83×10^{11}
			7.33×10^9	4.23×10^{10}	1.38×10^{11}	1.58×10^{11}	4.32×10^{11}
	Total		7.78×10^{11}				

Part II

Region	Source	Outer Radius (cm)	Flux, Mev/(cm ²)(sec)					
			1 Mev	2 Mev	3 Mev	4 Mev	6 Mev	8 Mev
Vessel	Attenuated blanket gamma rays	121.0	7.09 × 10 ⁷	3.81 × 10 ⁹		6.10 × 10 ⁹	3.50 × 10 ⁸	2.58 × 10 ⁹
	Inside neutron shield capture gamma rays		5.72 × 10 ⁸	3.21 × 10 ⁹		2.19 × 10 ¹⁰	3.21 × 10 ¹⁰	8.75 × 10 ¹⁰
	Vessel capture gamma rays		1.98 × 10 ⁹	9.7 × 10 ⁹		4.80 × 10 ¹⁰	6.48 × 10 ¹⁰	1.83 × 10 ¹¹
	Outside neutron shield capture gamma rays		6.27 × 10 ⁹	2.89 × 10 ¹⁰		6.12 × 10 ¹⁰	8.30 × 10 ¹⁰	2.40 × 10 ¹¹
			8.89 × 10 ⁹	4.56 × 10 ¹⁰		1.37 × 10 ¹¹	1.80 × 10 ¹¹	5.13 × 10 ¹¹
	Total		8.84 × 10 ¹¹					
Outside Neutron Shield	Attenuated neutron shield and vessel gamma rays	178.9	1.06 × 10 ⁶	5.64 × 10 ⁷		1.21 × 10 ⁹	2.62 × 10 ⁹	9.61 × 10 ⁹
	Outside neutron shield capture gamma rays		8.90 × 10 ⁷	1.24 × 10 ⁹		5.68 × 10 ⁹	1.04 × 10 ¹⁰	3.54 × 10 ¹⁰
	Primary sodium activity		1.47 × 10 ⁹		4.80 × 10 ⁹			
			1.56 × 10 ⁹	1.29 × 10 ⁹	4.80 × 10 ⁹	6.89 × 10 ⁹	1.30 × 10 ¹⁰	4.50 × 10 ¹⁰
		Total		7.25 × 10 ¹⁰				
Sodium Tank	Attenuated vessel and neutron shield gamma rays	396.2	1.13 × 10 ³	4.91 × 10 ⁵		2.54 × 10 ⁷	1.14 × 10 ⁸	6.15 × 10 ⁸
	Primary sodium activity		1.47 × 10 ⁹		4.80 × 10 ⁹			
			1.47 × 10 ⁹	4.91 × 10 ⁵	4.80 × 10 ⁹	2.54 × 10 ⁷	1.14 × 10 ⁸	6.15 × 10 ⁸
	Total			7.03 × 10 ⁹				
	Structural	Attenuated neutron shield and sodium gamma rays	455.3	3.94 × 10 ⁸	1.71 × 10 ⁵	2.02 × 10 ⁹	9.7 × 10 ⁶	5.36 × 10 ⁷
		Total		2.77 × 10 ⁹				
Blast Shield	Attenuated neutron shield and sodium gamma rays	523.2	4.36 × 10 ⁷	2.56 × 10 ⁴	4.14 × 10 ⁸	2.46 × 10 ⁶	1.52 × 10 ⁷	8.85 × 10 ⁷
	Concrete capture gamma rays			7.19 × 10 ⁴		2.29 × 10 ⁵	3.21 × 10 ⁵	4.23 × 10 ⁴
	Blast shield capture gamma rays			1.11 × 10 ⁵		2.12 × 10 ⁵	2.35 × 10 ⁵	1.11 × 10 ⁶
			4.36 × 10 ⁷	2.09 × 10 ⁵	4.14 × 10 ⁸	2.90 × 10 ⁶	1.58 × 10 ⁷	8.96 × 10 ⁷
		Total		5.66 × 10 ⁸				
Concrete	Attenuated neutron shield, sodium & blast shield gamma rays	706.1	6.13 × 10 ⁻⁵	6.70 × 10 ⁻⁴	7.04 × 10 ¹	3.25	1.53 × 10 ²	2.60 × 10 ³
	Concrete capture gamma rays			0.99		8.12	2.60 × 10 ¹	6.09
			6.13 × 10 ⁻⁵	0.99	7.04 × 10 ¹	1.14 × 10 ¹	1.79 × 10 ²	2.61 × 10 ³
	Total			2.87 × 10 ³				

Table XII

Part I

AXIAL GAMMA-RAY FLUXES ABOVE THE CORE

Region	Source	Outer Radius (cm)	Flux, Mev/(cm ²)(sec)				
			1 Mev	2 Mev	4 Mev	6 Mev	8 Mev
Core	Fission and fission products Blanket capture gamma rays	18.1	2.08×10^{14} 1.16×10^{13}	2.10×10^{14} 9.20×10^{13}	1.82×10^{13} 2.25×10^{13}	4.03×10^{13} 7.20×10^{11}	9.55×10^{12}
			2.20×10^{14}	3.02×10^{14}	4.07×10^{13}	4.10×10^{13}	9.55×10^{12}
		Total	6.13×10^{14}				
Blanket	Core fission and fission products Blanket fission and fission products Blanket capture gamma rays Plenum capture gamma rays	74.9	1.88×10^2 1.52×10^{11} 1.06×10^{12} 8.79×10^{10}	2.98×10^6 1.71×10^{11} 9.04×10^{12} 2.27×10^{11}	2.13×10^6 1.61×10^{10} 2.28×10^{12} 9.99×10^{11}	2.50×10^6 3.35×10^{10} 7.23×10^{10} 1.24×10^{12}	9.56×10^{11} 3.01×10^{12}
			1.30×10^{12}	9.44×10^{12}	3.30×10^{12}	1.35×10^{12}	3.97×10^{12}
		Total	1.94×10^{13}				
Plenum	Attenuated core and blanket gamma rays Plenum capture gamma rays Neutron shield capture gamma rays	147.4	7.80×10^6 5.96×10^9	1.61×10^9 1.86×10^{10} 4.66×10^8	2.68×10^9 1.02×10^{11} 9.48×10^8	2.26×10^8 1.40×10^{11} 9.17×10^8	2.60×10^9 3.49×10^{11} 4.31×10^9
			5.97×10^9	2.07×10^{10}	1.06×10^{11}	1.41×10^{11}	3.56×10^{11}
		Total	6.30×10^{11}				

Part II

Region	Source	Outer Radius (cm)	Flux, Mev/(cm ²)(sec)					
			1 Mev	2 Mev	3 Mev	4 Mev	6 Mev	8 Mev
Neutron Shield	Attenuated blanket and plenum gamma rays Neutron shield capture gamma rays Sodium activity	228.1	4.28 × 10 ⁶	1.04 × 10 ⁸ 2.84 × 10 ⁶ 1.47 × 10 ⁹		2.42 × 10 ⁹ 1.90 × 10 ⁷	5.33 × 10 ⁹ 2.98 × 10 ⁷	1.76 × 10 ¹⁰ 1.80 × 10 ⁸
			4.28 × 10 ⁶	1.58 × 10 ⁹	4.8 × 10 ⁹	2.44 × 10 ⁹	5.36 × 10 ⁹	1.78 × 10 ¹⁰
		Total		3.20 × 10 ¹⁰				
Sodium	Attenuated plenum and neutron shield gamma rays Steel balls capture gamma rays Sodium activity	518.9	0.47	1.09 × 10 ³ 1.01 × 10 ³ 1.47 × 10 ⁹		5.26 × 10 ⁵ 1.79 × 10 ³	4.02 × 10 ⁶ 1.94 × 10 ³	2.41 × 10 ⁷ 8.87 × 10 ³
			0.47	1.47 × 10 ⁹	4.8 × 10 ⁹	5.28 × 10 ⁵	4.02 × 10 ⁶	2.41 × 10 ⁷
		Total		6.30 × 10 ⁹				
Steel Balls	Attenuated plenum and neutron shield gamma rays Steel balls capture gamma rays	641.		1.84 × 10 ²	6.27 × 10 ³	2.36 0.21	4.44 × 10 ¹ 0.27	3.77 × 10 ² 1.32
				1.84 × 10 ²	6.27 × 10 ³	2.57	4.47 × 10 ¹	3.78 × 10 ²
		Total		6.88 × 10 ³				
Heavy Concrete	(ρ = 280 lb/ft ³) Attenuated gamma rays	803.					2.2 × 10 ⁻⁶	4.2 × 10 ⁻⁵
		Total		4.4 × 10 ⁻⁵				

IV. APPLICATION OF RADIATION CALCULATIONS TO THE SHIELD DESIGN

A. Neutron Shield

The primary function of the neutron shield is to reduce the neutron fluxes escaping from the blanket to levels which will not cause undue activation of the equipment located in the sodium tank. This includes pumps, heat exchanger, etc., which may have to be removed for service or replacement, and also includes the secondary sodium which passes from the reactor building to the sodium plant through large exposed pipes. These pipes would be expensive to shield; therefore, it is desirable to protect the secondary sodium from activating neutrons while it is in the heat exchanger. It is quite difficult to compute the neutron fluxes in the sodium with enough accuracy to assure minimum activation without excessive over shielding.

Boron carbide was selected to be the main constituent of the neutron shield. It is stable and can withstand the high system temperatures. Since hydrogenous materials are ruled out because of safety considerations, carbon fulfills the function of light atoms required for slowing down the neutrons. Boron absorbs neutrons but does not emit hard capture gamma rays. However, boron carbide is expensive and the boron content is far in excess of the amount required by the shield design. The neutron-shield material was therefore changed to graphite with 3% (by weight) boron carbide added.

The thickness of the neutron shield was initially set at 2 ft on the basis of calculations that assumed two neutron energies, fast and thermal. These calculations were crude as well as pessimistic, but were adequate as a preliminary design procedure. Since the fission rate in the blanket was not very well known, more accurate shield calculations were not yet appropriate. In any case, established methods and constants were not available at that time.

The actual design of the neutron shield was made up of 4-in. square stainless steel boxes filled with the neutron-shield material and cooled by sodium circulating between the boxes. The first design called for six rows of 4-in. square boxes. Counting from the inside, or blanket side, rows one, two, three, and five were filled with graphite and rows four and six were filled with 3% boron carbide in graphite. Rough calculations showed that the three rows of boxes nearest to the blanket could be graphite with no boron without any sacrifice in shield performance. A few neutrons, which otherwise would have been lost to boron, could thus be reflected back to the blanket. This was the arrangement of the neutron shield for which the ten-group STAR code neutron calculations were performed (see Tables IV, V, and VI). Considerations of mechanical design finally led to the specification of seven rows of boxes for the neutron shield as shown in Fig. 8 and Tables I, II, and III.

Results of the STAR problems did not indicate that the original estimates of the thickness of the neutron shield should be changed. However, other considerations in the reactor design did require changes in the neutron shield. Since large deviations were caused by the location of the instrument thimbles (see Section IV-C-1), significant portions of the neutron shield had to be completely unborated. In general, these regions did not coincide with locations that required low activation outside of them. The geometries were much too complex to calculate such effects explicitly; however, problems comparing a nonborated neutron shield with a borated shield (see Figs. 14 ff) give a rough idea of the changes. Omission of the boron in the neutron shield increases the thermal-neutron flux at the instrument location by nearly two orders of magnitude.

B. Sodium Activity

There are two separate sodium systems in the EBR-II. The primary sodium is pumped from the reactor to the heat exchanger and cannot leave the sodium tank that encloses the reactor. Most of its activity is caused by neutron capture as it flows through the reactor and neutron shield. The secondary sodium circulates between the heat exchanger in the reactor tank and the sodium boiler plant. The secondary sodium is activated during its passage through the heat exchanger. The pipe carrying this sodium passes through regions to which people may gain access readily.

1. Primary Sodium Activation

The tremendous amount of active primary sodium around the reactor represents a highly intense source of gamma rays. The activity of this sodium is due primarily to its exposure to neutrons as it circulates through the reactor and neutron shield. The activity due to the neutron absorptions in this region was first determined by the early twenty-group diffusion problems which were done primarily for purposes of reactor core design. Neutron leakage into the sodium tank from the neutron shield contributed a relatively small amount of additional activity. This was based on early two-group calculations and substantiated by the ten-group STAR problems.

The importance of the primary sodium activity lies chiefly in the high level of activity that is distributed uniformly throughout the sodium tank. It is a factor to consider in the total shield thickness, concrete heating, and instrument design. Access to components above the reactor after shutdown may be limited by the primary sodium activity. The primary sodium activity must be known rather well to analyze any of these situations.

In computing the sodium activity, neutron absorptions must be considered to occur at essentially all neutron energies from thermal to over

1 Mev. The characteristic flux peak at thermal energies, at which most activation takes place in thermal reactors, is not always present. Therefore, multigroup techniques were required for calculation of the activation. The volume of the sodium in the reactor core at any given time is about one-tenth of one per cent of the total sodium in the primary system. It has been assumed for these computations that mixing is uniform between sodium that has passed through the core and the rest of the primary sodium. The 15-hr half-life is long enough so that this should be a safe assumption. Even if the most active sodium were to collect preferentially, it is not expected to cause problems because of the conservatism in the design of the biological shield.

An example of the significance of each of the neutron-energy ranges is given for the control-rod region and the neutron-shield region in Table XIII. In this particular case the calculations were not made on the basis of the presence of a borated neutron shield, although calculations show that borating the neutron shield does not affect the number of sodium absorptions in the core. It is clear that the high-energy neutrons make a major contribution to the activation of sodium. It is also clear that the effective neutron energy for absorption changes quite radically between the rod region and the neutron shield. Comparisons of the contribution to the total sodium activity from each of the reactor regions are given in Table XIV. It should be noted here that all of the sources of activation listed in Table XIV are considered to be in the "vessel" as differentiated from activations that occur while the sodium is in the rest of the primary tank. The large amount of activation in the vessel and neutron-shield region is due in part to the effective moderation of neutrons by graphite and the subsequent greater amount of absorption. It is also due to the increased sodium volume in those areas, which leads to a greater number of absorptions.

Table XIII

PRIMARY SODIUM ACTIVATION BY NEUTRON ENERGY

Groups	Energy Range (Mev)	a	b
1 - 2	0.4979 - ∞	-	6.78
3 - 4	0.1832 - 0.4979	-	6.44
5 - 6	0.0674 - 0.1832	0.0018	4.93
7 - 8	0.00912 - 0.0674	0.0068	1.84
9 - 10	300. - 9112×10^{-6}	0.0099	0.056
11 - 12	43. - 300×10^{-6}	1.216	0.0004
13 - 14	22. - 43×10^{-6}	0.810	-
15 - 16	9. - 22×10^{-6}	1.500	-
17 - 18	0.7 - 9×10^{-6}	7.835	-
19 - 20	0.0 - 0.7×10^{-6}	105.7	-

a) relative neutron absorptions in the neutron shield.

b) relative neutron absorptions in the control-rod region.

Table XIV

PRIMARY SODIUM ACTIVATION BY REGION

Region	Sodium Activations (%)	Absorptions/sec
Core	7.1	9.3×10^{14}
Rod	6.6	8.6×10^{14}
Inner Blanket	7.5	9.8×10^{14}
Outer Blanket	6.5	8.5×10^{14}
Vessel	37.4	4.9×10^{15}
Neutron Shield	34.9	4.6×10^{15}
Total		1.31×10^{16}

If $3.4 \times 10^8 \text{ cm}^3$ is used as the volume of the primary sodium, the total absorption rate of 1.3×10^{16} absorptions/sec will give an equilibrium activity of $1.6 \times 10^8 \text{ Mev}/(\text{cm}^3)(\text{sec})$. The validity of this type of calculation has been tested against data on the EBR-I NaK coolant and gave results which checked measurements to within 10%. The activity of a 1-liter sample of NaK from the primary system of EBR-I was measured and found to give a dose of 10 mr/hr at a distance of 6 ft immediately after shutdown. Fluxes from a six-group diffusion theory problem gave a dose of 9 mr/hr. A similar determination of the activity based on fluxes from a three-group, two-dimensional problem gave an activity that produced a dose of 11 mr/hr.

2. Secondary Sodium Activation

The EBR-II sodium system does not permit any sodium from the primary system to leave the reactor building. The heat exchanger, where the transfer between the radioactive primary sodium and the secondary sodium takes place, must be located in an area of minimum neutron flux to keep the activity of the secondary sodium, which does leave the reactor building, to a minimum. Since the heat exchanger is located in the primary sodium tank there will be some neutrons reaching it. The first estimate of this neutron flux by simple one- and two-group diffusion calculations predicted an activity of the secondary sodium that would require about 3 in. of lead shielding on the pipes to reduce the radiation to 7.5 mr/hr at the surface. The ten-group STAR calculations performed later indicated that the radiation from the secondary sodium activity would not exceed 25 mr/hr at the surface of the unshielded 1-ft-diameter secondary sodium pipe.

In order to reduce the activation in the sodium, a boron-steel shield was located around the heat exchanger (see Fig. 10). The multigroup STAR problems that were used to determine the activity of the secondary

sodium system include some that were designed to indicate the value of boron in the neutron shield and in the heat exchanger. As might be expected in a region of materials that are not efficient neutron moderators, such as iron and sodium, and in which, therefore, a large thermal neutron flux is not present, boron has a limited value as a neutron absorber. The effect of boron on the neutron absorptions may be estimated from Table XV. The boron concentration in the heat exchanger wall was taken to be 1.25% by weight for these problems. The calculations indicate that the secondary sodium activity is decreased by a factor of two by putting the boron in the neutron shield and by another factor of three by putting the boron steel around the heat exchanger. In the final design a 1-in.-thick stainless steel plate with a boron concentration of 1.5% was used on the side facing the reactor core, and a $\frac{1}{2}$ -in. plate with 1.1% boron was used around the other side of the heat exchanger.

Table XV

AVERAGE NEUTRON-ABSORPTION RATE IN
THE SECONDARY SODIUM

Case	% Boration in Borated Regions of Neutron Shield	Absorption Rate	
		Absorptions/(cm ³)(sec)	Relative
Unborated Wall			
1	0	1.9 x 10 ⁴	1
2	0.6	1.2 x 10 ⁴	0.60
3	3.0	6.7 x 10 ³	0.35
4	30	1.3 x 10 ³	0.067
Borated HX Wall			
5	0	4.6 x 10 ³	0.24
6	0.6	3.7 x 10 ³	0.19

The use of the ten-group neutron fluxes to determine secondary sodium activation was considered to be quite conservative. These fluxes were computed as though the centerline of the heat exchanger was on the radial centerline of the reactor. The activation of the sodium was computed on the basis that all of the heat exchanger was exposed to these centerline neutron fluxes. Measurements in the dry critical experiments (discussed below) indicated that the activation rates might be higher than predicted by theory. In order to check the prediction that no shielding would be required on the secondary sodium pipes, an additional determination of neutron flux was made.

The new flux determination was done with a 20-group diffusion theory calculation, using the RE-122 program⁷ for the IBM-704. The material thicknesses for the new problems are given in Tables I, II, and III.

The energy groups are those in Table VII. In this set of problems advantage was taken of the fact that the heat exchanger is not on the reactor centerline but above it, and also that the neutron fluxes are appreciably attenuated along the centerline of the heat exchanger from bottom to top. In addition, the final material arrangements in the neutron shield were used. Several problems were run, each having the sodium thickness between the neutron shield and the heat exchanger specified by the proper angular path to the heat exchanger (see Fig. 25).

Fig. 25

Diagram for Off-Centerline
Heat Exchanger Study

The RE-122 problems were used to correct the older STAR results for the geometrical effects, which were neglected originally, and for changes in the final arrangements of materials. The results indicate that the radiation level at the surface of the 1-ft-diameter secondary sodium pipe will not exceed 5 mr/hr. However, flux measurements made during the dry critical experiments gave fluxes higher than those predicted. Allowance of a factor of ten for this uncertainty yields a prediction of 50 mr/hr for the activity at the sodium pipe wall. Even if this figure is approached in practice,

it will not pose any serious problem for the operation of EBR-II.

As a part of the dry critical experiment program, fluxes in the neutron-shield region were measured with a BF_3 counter, both with and without a cadmium sleeve.¹³ A total flux of about $2 \times 10^{10} \text{ n}/(\text{cm}^2)(\text{sec})$ was inferred from these measurements at a point near the outside edge of the neutron shield. Calculations were made at this location with the sodium in the system replaced by void and the flux level obtained was $2 \times 10^9 \text{ n}/(\text{cm}^2)(\text{sec})$, about an order of magnitude too low. It is believed that part of this discrepancy lies in the relationship of counter response to total flux. Cadmium ratio measurements gave a ratio of 2.3 compared with a

¹³McVean, R. L., et al., EBR-II Dry Critical Experiments - Experimental Results, ANL-6462 (Feb 1962).

calculated value of 1.1. This softer spectrum would yield an appreciably higher counter response and result in a measured flux value that is too high. The factor of ten allowed for these uncertainties should be quite conservative.

C. Other Problems

1. Instrument Location

In order to know the neutron-flux levels for control of the reactor power, neutron-detecting devices must be located near the core (see Fig. 9). The boron carbide had to be omitted from those portions of the neutron shield which were adjacent to the neutron detectors in order to retain sufficient neutrons to produce an adequate signal.

The size of the regions from which boron carbide was to be omitted involved two considerations: first, that the function of the neutron shield would still be assured in the locations where it was essential, and, second, that the neutron fluxes would be sufficient at the detectors to provide an adequate response at source power levels to permit a safe startup procedure. Neutron-flux problems were specified with an unborated neutron shield (see Table IX) in order to gain some knowledge about the effect of removing boron from certain portions of the neutron shield. Neutron fluxes when a borated neutron shield is present and neutron fluxes in the presence of an unborated neutron shield can be considered as limiting cases of the situation in which a partly borated neutron shield is present. Additional problems were done by the twenty-group code for the purpose of estimating neutron fluxes in the neutron shield in the vicinity of the instrument locations. The determination of response at source power was not possible in a precise analytical manner and was therefore estimated as well as possible and then checked experimentally with a mockup assembly on ZPR-III. The results of the calculations and experiments are given by Loewenstein.¹⁴ The comparison indicates that the calculations could predict the qualitative features of the fission distribution but that the prediction was probably not precise enough for reactor design. This means that experimental results may play a significant role in this problem in the future.

2. Fuel Transfer

The transfer of fuel from the reactor to the process cell is accomplished in steps. The first step is to remove the fuel subassembly and place it in the storage rack, which is located inside the primary tank (see Fig. 6). The subassembly remains in the storage rack for a minimum of 14 days. While in the storage rack, no special shielding is needed, since it is still in the primary tank and, hence, within the biological shield. Dissipation of the decay heat is accomplished by natural circulation of the primary sodium through the storage rack.

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¹⁴Loewenstein, W. B., The Physics Design of the EBR-II ANL-6383 (July 1961), p. 44.

When the subassembly is transferred from the primary tank, it is drawn through the top shield of the reactor into a refueling machine (see Fig. 26) located directly above the storage basket. The refueling machine has a $13\frac{1}{2}$ -in. lead shield, which reduces the radiation level from the most radioactive subassembly to less than 5 mr/hr at 20 in. from the outside of the machine. Additional shielding in the form of an adjustable lead ring attached to the refueling machine serves to reduce the radiation streaming during passage of the subassembly into the machine. The $3\frac{1}{2}$ -in. fuel-removal duct is closed by a rotating plug which consists of 12 in. of lead.

Fig. 26. Refueling Machine

3. Top Shield Radiation Leakage

The top shield of the EBR-II contains a large number of voids and penetrations (see Figs. 5, 6, and 12). The rotating plugs are stepped to prevent streaming at the circumference. Some of the remaining penetrations are not directly stepped. The control rod drives are examples. This is necessary because the available space is too limited to permit stepping. Streaming is reduced by keeping clearances as small as possible and by making the top shield thicker than would otherwise be necessary. If the top shield were solid, the thickness could be reduced by nearly 3 ft.

The secondary sodium pipes are a potential streaming problem (see Fig. 10) which can be relieved by offset and patch-shielding techniques. Since this shield must attenuate gamma rays, patching is a straightforward problem of keeping a constant mass per unit area between the primary sodium and the shield exterior.

A particularly difficult problem to assess is that of streaming along the annuli in the twelve control rod drives. These rods consist of steel and void, and extend from the reactor core through the primary sodium and the top shield. The rods are in much too small an area to permit step-shielding techniques. As a result, the voids are kept as small as possible and staggered within the rod whenever possible. Streaming was estimated by the standard annular gap formulas¹¹ to be within acceptable limits. Because these formulas are approximations and because the actual configurations are complex, these values will be determined experimentally during the initial operation of the reactor.

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