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EBWR CORE DESIGN STUDIES

by

H. P. Iskenderian
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I. INTRODUCTION

A study was made of EBWR core designs in which uniformly enriched UO_2 fuel rods clad with zirconium, capable of a maximum exposure of about 10,000 Mwd/tonne, were to be used.

The core of these designs was to have a length of 5 ft (152.4 cm) and a diameter of about 5 ft, filling the present EBWR with its total of 147 fuel box elements. The longer core of 5 ft was considered, at the time, to yield greater stability of operation. A burnable poison was to be used to allow for k_{ex} required for a maximum fuel burnup of 10,000 Mwd/tonne.

A first core design, A, with H_2O/UO_2 volume ratio of 1.63 had some desirable nuclear characteristics, requiring a reasonably low initial enrichment for criticality and a good conversion ratio. For a closer view of the power distribution characteristics of this core, some educated guesses were made of the operating void distribution in the core, and the corresponding neutron flux and source distributions obtained, with the aid of PDQ-2 two-dimensional calculations.(1) These data indicated a high maximum-to-average power ratio of 4.1.

Hydrodynamic calculations made, subsequently, for this core indicated that the assumed void distribution had been optimistic, and that it would not be possible to obtain 100 Mw with this core. The difficulty was due to excessive (calculated) pressure losses in the system, resulting in reduced velocities of inflow fluid in the core, which occurred when a core of closely packed design was used in EBWR.

In a second core design B, the number of fuel pins per element used in the upper half of the core was reduced from 7 x 7 to 6 x 6. This resulted in a reduction in the value of maximum-to-average power ratio of 4.1 for design A to about 2.8.

It appeared from these results that a further improvement of core performance should result by using a tight core lattice everywhere but in the upper central quarter of the core, where the void concentration was the highest.

In the design C of the core embodying these improvements, there is a more uniform fluid flow distribution, radially, as well as greater plutonium formation, than in design B. The calculated maximum-to-average power ratio in design C was 2.1, which is nearly half of that of design A, and three-fourths of that of design B.

Details of core designs A, B, and C, nuclear constants, and burnup characteristics are given in this report.

II. INITIAL STUDY OF CORE DESIGNS

From nuclear considerations for an efficient core, the ratio of H_2O/UO_2 should be small in order to obtain a high conversion ratio and a long core life.

Some preliminary calculations were made to determine the enrichments required for cold clean cores, with varying H_2O/UO_2 ratios (see Figure 1 and Table I). It will be noted from these data that the enrichments for criticality of these core designs with H_2O/UO_2 values varying from 1.276 to 2.775 do not vary greatly. Furthermore, if consideration be given to fuel required for burnup (~10,000 Mwd/tonne), one will find that the cores with tighter lattice require lower initial enrichments by virtue of their high ICR, or ability to produce plutonium. The ICR value of the core with H_2O/UO_2 = 1.632 will be greater than that for the core with H_2O/UO_2 = 2.775 by a factor greater than 1.27, allowing for excess enrichment required for burnup of fuel in these cores.

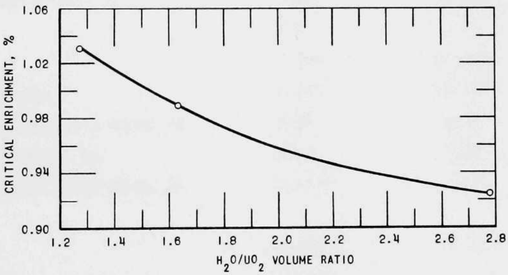

Fig. 1. Critical Enrichment vs H_2O/UO_2 Volume Ratio for Cold, Clean Core (EBWR Alternative Core Design)

III. DETAILED STUDY OF CORE DESIGNS

A. Core Design A

The core with H_2O/UO_2 = 1.632 (design A) was studied in greater detail to determine the advantages and limitations of this design with a tight lattice. Volume fractions and nuclear constants of design A are given in Tables II and III.

Table I

EBWR ALTERNATIVE CORE DESIGN STUDIES: UO_2 FUEL MATERIAL

Core Diameter = 147.4 cm   Core Height = 152.4 cm (5 ft)

Cold Clean Critical Core Constants

$$\left[\Lambda \equiv (1 + L^2B^2)(1 + \tau B^2) \simeq 1.052\right]$$

H_2O/UO_2	Pellet Diam (in.)	kT_n (ev)	f_{th}	ϵ	p	ϵp	$(\eta f)_{crit}$*	η^U	Enrichment for Criticality (%)	ICR**
1.276	0.420	0.028	0.8650	1.0568	0.7727	0.8166	1.2884	1.4894	1.03	0.389 + 0.474 = 0.863
1.632	0.392	0.028	0.8415	1.047	0.8094	0.8474	1.2412	1.4746	0.99	0.402 + 0.394 = 0.796
2.023	0.367	0.027	0.8276	1.0411	0.8375	0.8719	1.2065	1.4578	0.92	0.413 + 0.296 = 0.709
2.775	0.330	0.027	0.8080	1.0333	0.8722	0.9012	1.1673	1.4446	0.92	0.432 + 0.260 = 0.692

* $(\eta f)_{crit} = \dfrac{\Lambda = (1 + L^2B^2)(1 + \tau B^2)}{\epsilon p} \simeq \dfrac{1.052}{\epsilon p}$; $\tau = 38\ cm^2$; $B^2 = 0.00123\ cm^{-2}$; $\eta^{U^{235}} = 2.07$

** $ICR = \dfrac{\Sigma_a^{28}}{\Sigma_a^{25}} + \eta^{25}\,\epsilon(1 - p)\,e^{-\tau B^2}$ (neglecting contribution from fast capture in U^{238})

Table II

EBWR CORE 2 ALTERNATIVE DESIGN STUDIES, DIMENSIONS, AREAS, AND VOLUME FRACTIONS OF MATERIALS IN A 12.75-in. x 12.75-in. CELL CONSISTING OF NINE SIMILAR SUBASSEMBLIES

	Design A, 49 Pins Per Element	Designs B and C: 49 Pins Per Element	Designs B and C: 36 Pins Per Element
Diameter of Fuel Pins, in.	0.445	0.395	0.395
Diameter Pellets, in.	0.392	0.350	0.350
Thickness of Zircaloy-2 Clad, in.	0.025	0.020	0.020
Enrichment,* %	2.4		
UO_2 Area, sq in.	53.223	42.428	31.171
Zircaloy-2 Clad Area, in.	14.547	10.390	7.633
Zircaloy-2 Guides and Spacers Area, in.	4.58	4.58	4.58
Zircaloy-2 Followers Area, in.	2.50	2.50	2.50
Helium (between UO_2 and Clad) Area, in.	0.8177	1.221	0.897
Volume Fraction			
UO_2	0.3274	0.2610	0.1918
H_2O	0.5345	0.6240	0.7122
H_2O/UO_2	1.63	2.39	3.71
Zircaloy-2	0.1331	0.1075	0.0905
Helium	0.0050	0.0075	0.0055

Density of UO_2 = 10.2 gm/cc

*Enrichment in Designs B and C: 2.4% in lower central quarter of core
2.8% in remainder of core

Table III

BURNUP CALCULATIONS ON EBWR 5-ft CORE DESIGN A

(Enrichment = 2.4%; kT_n = 0.047 ev)

Operating Core

$\hat{\sigma}_a^{25}$ = 655 b $\hat{\sigma}_a^{49}$ = 1769 b $\hat{\sigma}_a^{41}$ = 1760 b $\hat{\sigma}_a^{40}$ = 1300 b σ_a^{26} = 7 b

η^{25} = 2.043 η^{49} = 1.6708 η^{41} = 2.223 σ_a^{42} = 206 b

ϵp = 0.7856

p = 0.740

$(1 + L^2B^2)(1 + \tau B^2) = 1.1132 = \Lambda$

	$t \times 10^{-7}$ sec	0	2	4	8
	Mwd/tonne	0	2660	5320	10,640
	Σ_a^{25}, cm^{-1}	0.119538	0.102835	0.08770	0.063535
	Σ_a^{28}, cm^{-1}	0.0201624	0.0201353	0.0201082	0.020054
	DisF $\times\ \Sigma_a^{p}$, cm^{-1}	0.0098436			
(1)	DisF $\Sigma_a^p + \Sigma_a^{28}$, cm^{-1}	0.03006	0.0299789	0.0299518	0.029898
	Σ_a^{49}, cm^{-1}	0	0.023351	0.038918	0.05440
	Σ_a^{41}, cm^{-1}	0	0.000176	0.00088	0.004752
	Σ_a^{40}, cm^{-1}	0	0.001137	0.00390	0.010270
	Σ_a^{42}, cm^{-1}	0	0	0	0.0000927
	Σ_a^{26}, cm^{-1}	0	0.000030	0.000058	0.000104
(2)	Σ_a^{FP} (σ_a^{FP} = 50 b), cm^{-1}	0	0.000125	0.000250	0.000500
	$\eta^{25}\Sigma_a^{25}$, cm^{-1}	0.244216	0.210092	0.179314	0.129802
	$\eta^{49}\Sigma_a^{49}$, cm^{-1}	0	0.0390148	0.065024	0.090891
	$\eta^{41}\Sigma_a^{41}$, cm^{-1}	0	0.000391	0.0019562	0.0105637
	$\Sigma(\eta\Sigma_a)$, cm^{-1}	0.244216	0.249498	0.246294	0.231257
(3)	$\Sigma_a^{25} + \Sigma_a^{49} + \Sigma_a^{41}$, cm^{-1}	0.119538	0.126362	0.127568	0.122687
(4)	1.035 times values of (3)	0.123721	0.130784	0.132032	0.126980
(5)	$\Sigma_a^{26} + \Sigma_a^{40} + \Sigma_a^{42}$, cm^{-1}	0	0.0012925	0.004208	0.0109667
	Σ_a = (1) + (2) + (4) + (5), cm^{-1}	0.153727	0.162180	0.166442	0.168345
	$(\eta f) = \Sigma\eta\Sigma_a/\Sigma_a$	1.5886	1.5384	1.4798	1.3737
	$k_{eff} = \left(\frac{pe}{\Lambda}\right)(\eta f) = 0.7057\,(\eta f)$	1.1210	1.0856	1.0443	0.9694

Burnup calculations were made with the aid of the Analog Computer (see Appendix A). Isotopic concentrations of fuel elements are given in Figures 2 and 3.

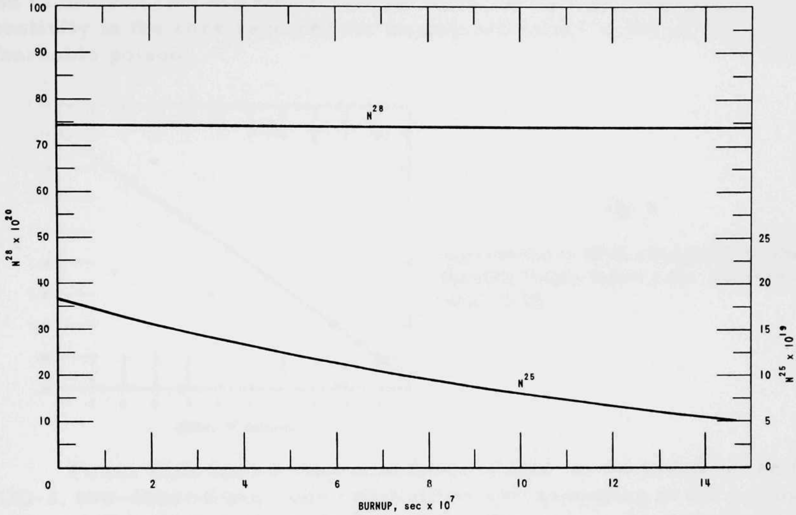

Fig. 2. Isotopic Concentration of U^{235} and U^{238} vs Burnup for EBWR Alternative Core Design A. Initial Enrichment = 2.4%. Average Thermal Neutron Flux = 1.18×10^{13} n/cm^2-sec.

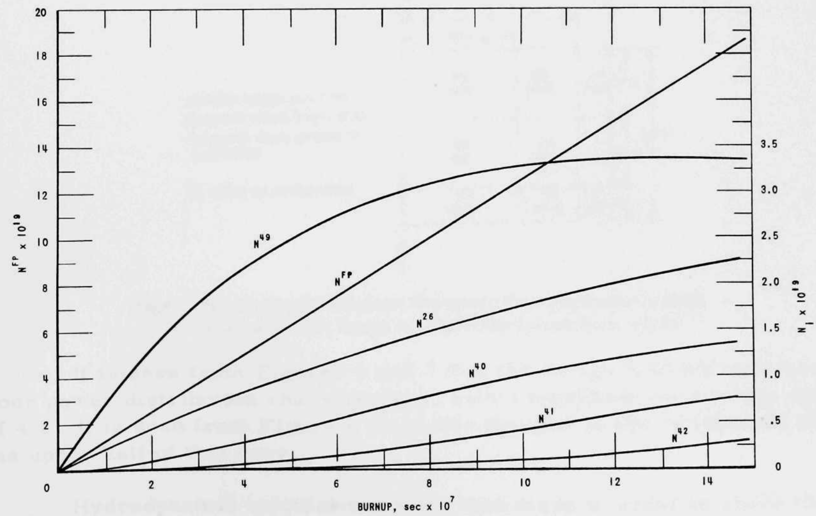

Fig. 3. Isotopic Concentrations (N_i) of U^{236}, Pu^{239}, Pu^{240}, Pu^{241}, Pu^{242}, and of Fission Product (N^{FP}) vs Burnup for EBWR Alternative Core Design A. Initial Enrichment = 2.4%. Average Thermal Neutron Flux = 1.18×10^{13} n/cm^2-sec. $\sigma_a^{U^{240}} = 1300$ b.

Burnup calculations were made with the aid of the Analog Computer (see Appendix A). Isotopic concentrations of fuel elements are given in Figures 2 and 3.

Fig. 2. Isotopic Concentration [illegible] vs Burnup for SGVR Alternative Core Design [illegible] Initial Enrichment [illegible] Average Thermal Neutron Flux = 1.19 x 10[illegible] n/cm²-sec.

Fig. 3. Isotopic Concentration [illegible] and of [illegible] Product of [illegible] vs Burnup for SGVR Alternative Core [illegible] Average Thermal Neutron Flux = 1.14 x 10[illegible] [illegible]

Reactivity calculations are given in Table III, and k_{eff} vs Mwd/tonne exposures are shown in Figure 4. These burnup calculations referred to a core with a uniform void distribution (20%) and with no allowance for variation in neutron flux distribution in the core. It was assumed that the excess reactivity in the core required for burnup allowance would be controlled by a burnable poison.

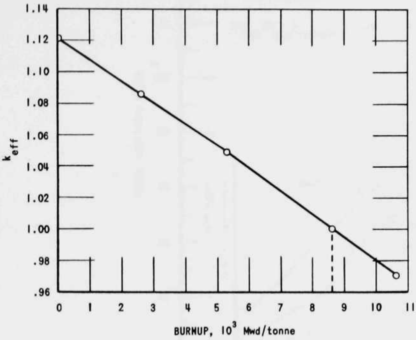

Fig. 4

k_{eff} vs Burnup for EBWR Alternative Core Design A. H_2O/UO_2 Volume Ratio = 1.63. Initial Enrichment = 2.4%.

Power distribution characteristics of this core were obtained from PDQ-2, two-dimensional code calculations corresponding to the estimated core void distribution shown in Figure 5. These characteristics are shown in Figures 6 and 7.

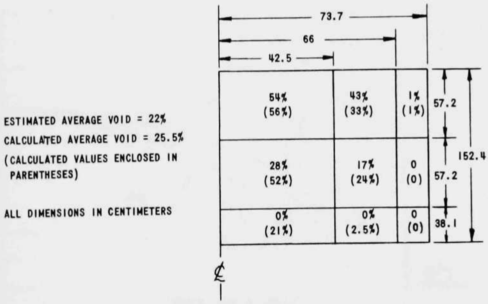

Fig. 5. Estimated and Calculated Percentage Void Distribution in EBWR Alternative Coil Design A. H_2O/UO_2 Volume Ratio = 1.63.

It is seen from Figures 6 and 7 that the design A of the core has a poor power distribution characteristic, with a maximum-to-average ratio of 4.1. It is seen from Figure 6 that little thermal power is obtained from the upper half of this core.

Hydrodynamic calculations were then made in order to check the accuracy of the assumed void distribution shown in Figure 5. These

Reactivity calculations are given in Table [illegible], and k_{eff} vs [illegible] are shown in Figure 4. The [illegible] calculations referred to a core with a uniform void distribution [illegible] (25%) and with no allowance for variation in neutron flux distribution in the core. It was assumed that the excess reactivity in the core required for burnup allowance would be controlled by a burnable poison.

Power distribution [illegible] characteristics of this core were [illegible] from PDQ-4, two-dimensional [illegible] corresponding to the [illegible] void distribution shown in Figure 5. These characteristics are shown in Figures 6 and 7.

Fig. 5. Subassembly and Coolant-Mixing Percentage Void Distribution [illegible] Alternate [illegible] Design A, H_2O/UO_2 [illegible]

It is seen from Figures 6 and 7 that [illegible] power distribution characteristics, with [illegible] of [illegible]. It is seen from Figure [illegible] that little [illegible] the upper half of the core.

Hydrodynamic calculations were [illegible] adequacy of the assumed void distribution [illegible] calculations indicated that the assumed void distribution had been somewhat optimistic, and that it would not be possible to obtain 100 Mw from design A of the core.

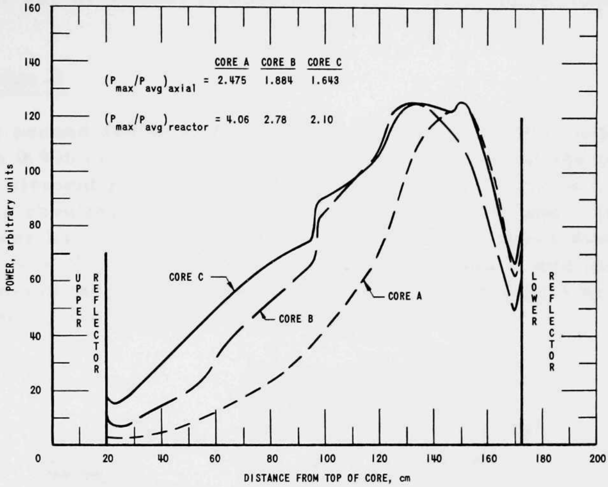

Fig. 6. Calculated Axial Power Distribution in EBWR Alternative Core Designs A, B, and C

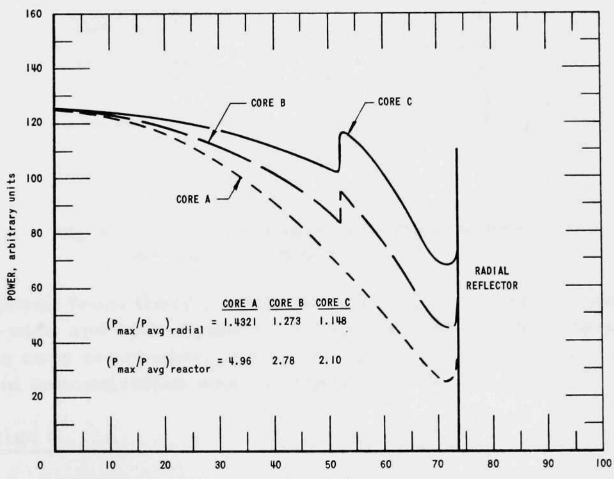

Fig. 7. Calculated Radial Power Distribution in EBWR Alternative Core Designs A, B, and C

calculations indicated that the assumed axial distribution had been somewhat optimistic, and that it would not be possible to obtain 100 Mw from design A of the core.

Fig. 5. Calculated Axial Power Distribution in ENVB Alternative Core Designs A, B, and C

Fig. 6. Calculated Radial Power Distribution [illegible] Core Designs A, B, and C

The above unsatisfactory characteristics were due to the great reduction of water content in the upper half of the core with this tight lattice (H_2O/UO_2 = 1.63). To improve the axial neutron flux, or power, distribution in the core, it was necessary to open the lattice design in the upper half of the core.

B. Core Design B

In the second design of the core B, the size of the fuel pellets was reduced from 0.996 cm (0.392 in.) to 0.89 cm (0.35 in.), and the number of fuel pins per element in the upper half of the core was reduced from 7 x 7 to 6 x 6. This resulted in H_2O/UO_2 volume ratios of 3.71 and 2.39 in the upper and lower halves of the core, respectively. The power distribution characteristics of this core are shown in Figures 6 and 7, and indicate a reduction in maximum-to-average ratio from 4.1 for design A to about 2.8. The calculated void distribution is given in Figure 8.

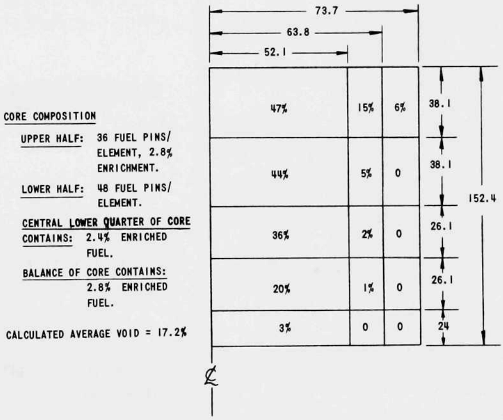

Fig. 8. Calculated Percentage Void Distribution in EBWR Alternative Core Design B.

It followed from these results that an improvement in core design, both nuclear-wise and hydrodynamic-wise, could be obtained by a closer packing of the core everywhere but at the upper central quarter of the core, where the void concentration was the highest.

C. Core Design C

Core design C has 7 x 7 pins per fuel element everywhere in the core except at the central upper quarter, where 6 x 6 pins per element are used. The void and power distribution characteristics of this core are shown in Figures 5, 6, and 9. The void distribution was obtained again by

The above unsatisfactory characteristics were due to the great reduction of water content in the upper half of the core with this tight lattice ($H_2O:UO_2 = 1.65$). To improve the nuclear characteristics, or power, distribution in the core, it was necessary to loosen the lattice design in the upper half of the core.

B. Core Design B

In the second design of the core, B, the size of the fuel pellets was reduced from 0.996 cm (0.392 in.) to 0.889 cm (0.35 in.), and the number of fuel pins per element in the upper half of the core was reduced from 7 x 7 to 6 x 6. This resulted in $H_2O:UO_2$ volume ratios of 3.71 and 2.39 in the upper and lower halves of the core, respectively. The power distribution characteristics of this core are shown in Figures 6 and 7, and indicate a reduction in maximum-to-average ratio from 4.4 for design A to about 2.8. The calculated void distribution is given in Figure 8.

Fig. 8. Calculated Percentage Void Distribution in BWR Superheat Core, Design B.

It followed from these results that an improvement in core design, both nuclear-wise and hydrodynamic-wise, could be obtained by a looser packing of the core everywhere and at the upper central quarter of the core, where the void concentration was the highest.

C. Core Design C

Core design C has 7 x 7 pins per fuel element [illegible] core except at the central upper quarter, where [illegible] used. The void and power distribution characteristics [illegible] shown in Figures 9, 10, and 11. The void distribution [illegible] physics-hydrodynamics iterations. As indicated in Figure 9, two enrichments, 2.4% and 2.8%, were used in the initial core. The lower enrichment used in the lower quarter core was for obtaining flatter power characteristics. Use of a uniform initial enrichment of 2.8% throughout the core would increase somewhat the maximum-to-average power ratio from its value of 2.1; there should, however, be also an increase in the average core life of the first core. With burnup, the power characteristics should approach that of Figures 6 and 7.

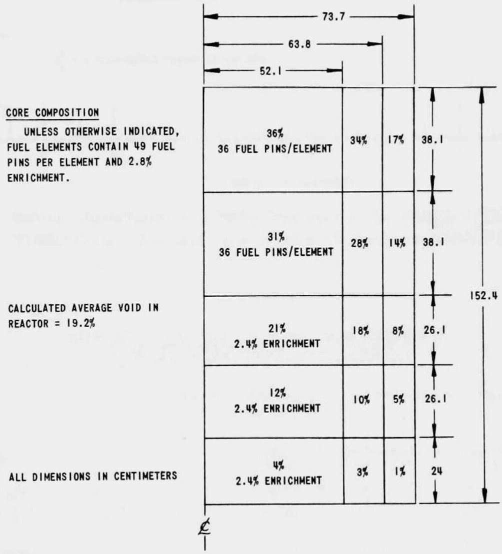

Fig. 9. Calculated Void Distribution in EBWR Alternative Core Design C

Burnup characteristics (see Figure 10) of this core (Figure 9) were obtained by TURBO[2] code calculations. Isotopic concentrations of the initial core are given in Table IV. As already stated, initial shim control of the reactor was obtained by means of a uniformly distributed poison (such as B^{10}-steel rods) in the core. Subsequent control was achieved by an additional poison, with macroscopic cross section Σ_a^p, which was varied in the core, as indicated in Table V. The value of Σ_a^p may be considered to be the equivalent of a uniform poison, such as H_3BO_3, dissolved in reactor water, allowance being made for its effects on reflector savings. It may be seen from Figure 10 that the core with an equivalent average enrichment of 2.7% would remain in operation for about 7500 Mwd/tonne without reshuffling of fuel in the core.

physics-hydrodynamics iterations. As indicated in Figure 9, two enrichments, 2.4% and 2.8%, were used in the initial core. The lower enrichment used in the inner quadrants was for obtaining flatter power characteristics. Use of a uniform initial enrichment of 2.8% throughout the core would increase somewhat the maximum-to-average power ratio from the value of 2.1; there should, however, be also an increase in the average core life of the base core. With burnup, the power characteristics should approach that of Figures 6 and 7.

Fig. 9. Calculated Fuel Distribution in BWR Alternate Core Design C

Burnup characteristics (see Figure 10) [illegible] were obtained by [illegible] calculations. [illegible] the initial core are given in Table IV. As [illegible] of the reactor was obtained by means of a [illegible] such as B^{10} [illegible] in the core. [illegible] an additional [illegible] macroscopic [illegible] in the core [illegible] Table V. [illegible] for the equivalent [illegible] portion [illegible] water allowance [illegible] for the [illegible] be seen from Figure [illegible] core [illegible] ment of [illegible] operation [illegible] probability [illegible] in the core.

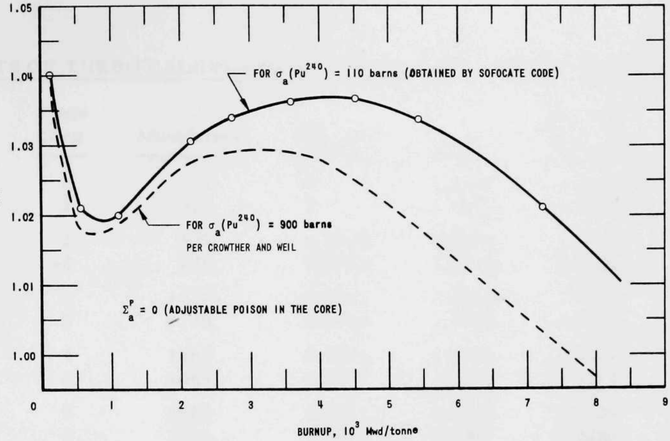

Fig. 10. Burnup Characteristics of EBWR Alternative Core Design C Calculated by TURBO Code. Thermal Cross Sections Obtained by SOFOCATE Code.

Table IV

BURNUP IN AN EBWR 5-ft CORE DESIGN C, TURBO 2-GROUP CALCULATIONS

Isotopic concentrations initially and after 7500 Mwd/tonne exposure (in units of 10^{24})

	Initially		After 7500-Mwd/tonne exposure						
Composition	N^{25} x 10^4	N^{28} x 10^2	N^{25} x 10^4	N^{26} x 10^5	N^{28} x 10^2	N^{49} x 10^4	N^{40} x 10^5	N^{41} x 10^6	$N^{Fiss\ Prod}$ x 10^4
2	1.260	0.4391	0.9403	0.5535	0.4369	0.1310	0.1936	0.4301	0.3268
3	1.720	0.5976	1.436	0.4972	0.5956	0.1435	0.1341	0.1837	0.2801
4	1.720	0.5976	1.450	0.4551	0.5961	0.1048	0.09947	0.1398	0.2561
5	1.260	0.4391	0.7499	0.8851	0.4351	0.1910	0.4361	1.498	0.5671
6	1.720	0.5976	1.231	0.8551	0.5938	0.2292	0.3522	0.7775	0.5140
7	1.720	0.5976	1.251	0.7911	0.5948	0.1661	0.2598	0.5824	0.4682
8	1.480	0.6000	0.9197	0.9890	0.5944	0.2942	0.5933	1.773	0.6531
9	1.720	0.5976	1.156	0.9777	0.5932	0.2439	0.4403	1.149	0.6036
10	1.720	0.5976	1.201	0.8742	0.5945	0.1745	0.3047	0.7635	0.5240
11	1.480	0.6000	0.8779	1.040	0.5944	0.2734	0.6214	2.123	0.7002
12	1.720	0.5976	1.144	0.9860	0.5935	0.2234	0.4250	1.178	0.6095
13	1.720	0.5976	1.241	0.8053	0.5949	0.1608	0.2609	0.6048	0.4773
14	1.480	0.6000	0.9774	0.8493	0.5963	0.1975	0.3970	1.197	0.5413
15	1.720	0.5976	1.310	0.6953	0.5951	0.1573	0.2198	0.4466	0.4077
16	1.720	0.5976	1.405	0.5629	0.5960	0.1102	0.1203	0.1877	0.2991

Notes: Compositions 2 and 5 contain elements with 36 pins; all others contain 49 pins per element; compositions 8, 11, and 14 have fuel with 2.4% enrichments; all others have fuel with 2.8% enrichment.

Table V

RESULTS OF TURBO CALCULATIONS FOR BURNUP IN CORE OF DESIGN C

Problem Series*	Time Step	Mwd/tonne	Σ_a^P, cm^{-1}	k_{eff}	k_{eff} with $\Sigma_a = 0$	σ_a^{FP}, b
100	1	314	0	1.0400	1.0400	65
200	1	314	0	1.0400	1.0400	65
100	2	575	0.0025	0.9853	1.0210	65
200	2	575	0.0015	0.9968	1.0182	65
100	3	1120	0.0015	0.9986	1.0200	65
200	3	1120	0.0010	1.0040	1.0182	65
100	4	1665	0.0012	1.0065	1.0236	65
200	4	1665	0.0012	1.0071	1.0242	65
100	5	2205	0.0013	1.0117	1.0310	65
200	5	2205	0.0015	1.0067	1.0280	65
100	6	2750	0.0015	1.0128	1.0342	65
200	6	2750	0.0015	1.0077	1.0290	65
100	7	3625	0.0018	1.0107	1.0364	65
200	7	3625	0.0018	1.0035	1.0292	65
100	8	4535	0.0020	1.0081	1.0366	65
200	8	4535	0.0020	1.0191	1.0476	40
100	9	5440	0.0020	1.0052	1.0337	65
200	9	5440	0.0020	0.9896	1.0182	40
100	10	6350	0.0020	1.0381		40
200	10	6350	0.0010	0.9964	1.0178	40
100	11	7250	0.0020	0.9924	1.0210	40
200	11	7250	0	1.0027	1.0027	40
100	12	8150	0.0015	0.9965	1.0179	40

*For Series 100 Problems, $\sigma_a(Pu^{240}) \simeq 110$ b obtained by SOFOCATE; for Series 200, $\sigma_a(Pu^{240}) = 900$ b, Σ_a^P = adjustable poison in core used to retain near criticality.

Notes: In Series 100 Problems, for Step 10, all self-shielding factors of fuel and B^{10} rods were brought up to date; for Steps 11 and 12, changes were made in self-shielding factors.

In Series 200 Problems, changes were made in self-shielding factors of B^{10} at Steps 8, 9, 10, and 11.

An objectionable feature of core designs B or C is an unavoidable water gap between the upper and lower sections of the core with different lattices. This could result in flux peaking and objectionable hot spots, unless the gap was partially filled with an absorbing material. DSN calculations were made to determine such flux peaking at the fuel of the lower

edge of gap for the three cases shown in Table VI. It is seen from the data of this table that such peaking flux may be suppressed with adequate poisoning of the gap section of the core.

Table VI

THERMAL FLUX PEAKING DUE TO GAP BETWEEN SECTIONS OF CORE

(Gap = $1\frac{1}{4}$ in.)

Material (%) in Gap Region	Flux at Lower Edge of Gap / Flux at Axial Peak
(1) H_2O (Zero Void)	1.56
(2) 0.597 H_2O + 0.315 Zr-2 + 0.088 He (45% void)	1.13
(3) 0.597 H_2O + 0.134 Zr-2 + 0.181 SS + 0.088 He (45% void)	0.92

An objection to the configuration of design C may be that not all of its fuel elements are interchangeable, in a scheme of reshuffling of fuel, for maximum fuel exposure. This difficulty is not, however, real, since the initial average rate of exposure of fuel elements with 6 x 6 pins per element is considerably lower (about 40%) than that directly below it. These fuel elements with 6 x 6 pins per box, may, therefore, be retained in position except for reversals, during their lifetime in the core. A programming scheme shown in Figure 11 illustrates this point.

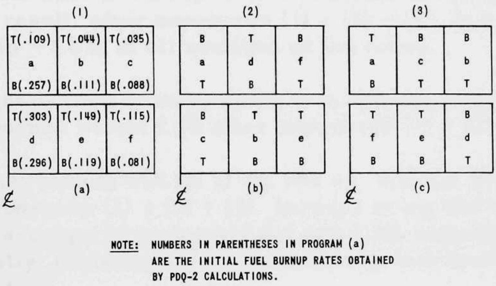

Fig. 11. Possible Programming of Fuel in EBWR Core II, Design C

IV. CONCLUDING REMARKS

It has been shown in the present study that requirements of a satisfactory power distribution, coupled with hydrodynamic considerations in the core, limit the use of tight lattices in the EBWR.

It was indicated that a reduction in maximum-to-average power ratio of the core from 4.1 to 2.1 could be achieved by increasing the volume edge of gap for the three cases shown in Table [illegible]. It is seen from the data of this table that such peaking flux may be suppressed with adequate poison-ing of the gap section of the core.

Table [illegible]

[illegible]AL FLUX PEAKING DUE TO GAP BETWEEN SECTIONS OF CORE

[illegible]

Material [illegible] Region	Flux at Further Edge [illegible] / Flux at Solid [illegible]
(1) [illegible] (Zero Void)	1.55
(2) [illegible] + 0.1[illegible] [illegible] voids	1.11
(3) [illegible] + 0.1[illegible] [illegible]	1.04

An objection to [illegible] configuration of design C may be that not all its fuel elements are [illegible] in a scheme of [illegible] of fuel for maximum fuel exposure. This difficulty is not, however, real, since the [illegible] average rate of [illegible] of fuel elements with [illegible] per [illegible] is considerably [illegible] (about 40%) than that directly below it. These fuel elements with [illegible] per box, may, therefore, be retained in posi-tion except for [illegible] during their lifetime in the core. A programming scheme shown in Figure [illegible] illustrates this point.

Fig. 11. [illegible] Programming of [illegible]

[illegible]. CONCLU[illegible]

[illegible] has been shown in the present [illegible] satisfactory [illegible] conf[illegible] in the core [illegible] data [illegible]

[illegible] was [illegible] reduct[illegible] range of the core from [illegible] could [illegible] ratio of H_2O to UO_2 from 1.63 to about 2.72 in a two-lattice system, with an initial average enrichment of 2.7% as compared with 2.4% for the first design. With these enrichments, the two cores were capable of average burnups of about 8000 Mwd/tonne.

In boiling reactors with a long "riser" and adequate lattice designs, smaller H_2O/UO_2 ratios than would be tolerable in the EBWR could be used. There is, however, a limit to this tightening of the lattice from considerations of stability in the reactor. These considerations are against obtaining a core with a high conversion ratio, and hence of long core life, unless a high initial enrichment be used. High enrichment means poor neutron economy associated with the control problems.

With the use of UO_2, with its ability to sustain long irradiation exposures, as practical fuel material in boiling reactors, it is very desirable to have cores with long burnup capacities. Use of tight lattices, which would yield long core lives, is limited in boiling reactors, as indicated in this report.

(A core with a nonuniformly distributed fuel, which would have more uniform reactivities to control over its lifetime than the usual designs, may give a longer core life with high neutron economy.)

If the initial rate of burnup were sustained, the programming of Figure 11 would result, after exposures (1) + (2) + (3), in a uniform average burnup, within $\sim \pm 8\%$, in all sections of the cores.

The top and bottom of sections b, c, d, and top of e of (1) will all have a uniform burnup within $\pm 5\%$ after exposures (1) + (2) + (3).

The highest burnup will be at a_B and a_T, and the lowest at e_B, f_T, and f_B. After exposures (1) + (2) + (3), burnups at a_B and a_T will be greater than the average by factors of 1.4 and 1.09, assuming initial rates of burnup. Actually, allowing for some flattening with burnup, these factors should be reduced.

V. METHOD OF CALCULATION

Two-group theory was used to calculate reactivity and power distributions. The equations are:

$$- D_1 \nabla^2 \phi_1 + \left(\Sigma_{a_1} + \Sigma_{s1_1} + \Sigma_{res}^{U^{238}} \right) \phi_1 = \epsilon \nu \Sigma_{f_2} \phi_2$$

$$- D_2 \nabla^2 \phi_2 + \Sigma_{a_2} \phi_2 = \Sigma_{s1_1} \phi_1 \quad ,$$

ratio of H_2O to UO_2 from 1.63 to about 2.12 in a two-lattice system, with an initial average enrichment of 2.7% as compared with 2.1% for the first design. With these enrichments, the two cores were capable of average burnups of about 8000 Mwd/tonne.

In boiling reactors with a long "riser" and adequate lattice designs, smaller H_2O/UO_2 ratios than would be tolerable in the EBWR could be used. There is, however, a limit to this tightening of the lattice from considerations of stability in the reactor. These considerations are against obtaining a core with a high conversion ratio, and hence of long core life, unless a high initial enrichment be used. High enrichment means poor neutron economy associated with the control problems.

With the use of UO_2, with its ability to sustain long irradiation exposures, as prospective fuel material in boiling reactors, it is very desirable to have cores with long burnup capacities. Use of tight lattices, which would yield long core lives, is limited in boiling reactors, as indicated in this report.

(A core with a nonuniformly distributed fuel, which would have more uniform reactivities to control over its lifetime than the usual designs, may give a longer core life with high neutron economy.)

If the initial rate of burnup were sustained, the programming of Figure 11 would result, after exposures (1) + (2) + (3), in a uniform average burnup, within ~1.5%, in all sections of the cores.

The top and bottom of sections b, c, d, and top of a of (1) will all have a uniform burnup within 2.5% after exposures (1) + (2) + (3).

The highest burnup will be at a_B and a_T, and the lowest at e_B, f_T, and f_B. After exposures (1) + (2) + (3), burnups at a_B and a_T will be greater than the average by factors of 1.4 and 1.09, assuming initial rates of burnup. Actually, allowing for some flattening with burnup, these factors should be reduced.

V. METHOD OF CALCULATION

Two-group theory was used to calculate reactivity and power distributions. The equations are:

$$-D_1 \nabla^2 \phi_1 + \left(\Sigma_{a_1} + \Sigma_{sl_1} + \Sigma^{U^{238}}_{res} \right) \phi_1 = \epsilon k \Sigma_{f} \phi_2$$

$$-D_2 \nabla^2 \phi_2 + \Sigma_{a_2} \phi_2 = \Sigma_{sl_1} \phi_1 .$$

where D is the diffusion coefficient, Σ_{sl_1} the slowing-down or removal cross section, $\Sigma_{res}^{U^{238}}$ = (N^{28} Res. Int.)/(Δu = lethargy interval for resonance neutrons), $\Sigma_f = \frac{1-p}{p} \Sigma_{sl_1}$ is the resonance absorption cross section in U^{238}; p, ϵ, and ν have their usual meaning, and Σ_{a_2} is the equivalent thermal cross section and includes epithermal absorption, as defined by Westcott,(3) (i.e., $\Sigma_{a_1} = 0$, when Westcott's cross section is used).

In the study of burnup in design C by the TURBO code, Σ_{a_2} was the thermal cross section obtained by the SOFOCATE(4) code. In this burnup study, two series of problems were run; in the first series, 100, the value of $\sigma_a^{U^{240}} \simeq 110$ b, obtained by the SOFOCATE code, was used. In the second series, 200, the value of $\sigma_a^{U^{240}} \simeq 900$ b was used to allow for the large resonance-capture line at 1 ev in Pu^{240}, in accordance with the work of Crowther and Weil.(5) The built-in library of fast cross sections Σ_{a_1} in the TURBO code does not include the plutonium series.

A. Neutron Temperature

The temperature of the neutron moderator for use in Westcott's cross-section data was obtained from Brown's formula(6):

$$kT_n = kT_{mod}\left[1 + A\,\frac{\Sigma_a(kT_{mod})}{\Sigma_s}\right] .$$

B. Disadvantage Factor DisF

Thermal disadvantage factors were evaluated by P_3 spherical harmonic approximations.

C. Fast Fission Factor ϵ

The fast fission factor ϵ was obtained from the formula(7)

$$\epsilon - 1 = \frac{[\nu^{28} - (1 + \alpha^{28})]\,\chi\Sigma_f^{28}}{\Sigma_c + \Sigma_{in} + \Sigma_f^{28} - \nu^{28}\,\chi\Sigma_f^{28}} ,$$

where all the cross sections are the values above the threshold, and χ is the fraction of the fission spectrum above the threshold. Also,

$$\alpha^{28} = \Sigma_c^{U^{238}} / \Sigma_f^{U^{238}} .$$

D. Neutron Age τ and Fast Diffusion Constant D_1

τ and D_1 were calculated by means of Deutsch's equivalence factors.(8)

Where D is the diffusion coefficient, Σ_a [illegible] the slow neutron [illegible] absorption cross section, Σ_{res}^{moo} = [illegible] Kev. [illegible] resonance neutrons. $\Sigma_r = \frac{1-p}{p}$ [illegible] is the resonance absorption cross section in U^{238}, [illegible] and [illegible] have their usual meaning, and Σ_{ag} is the [illegible] thermal cross section and includes epithermal [illegible] Westcott's [illegible] Σ_{ag} = 0, when Westcott's [illegible] is used.

In the study of [illegible] design [illegible] the thermal [illegible] spectrum obtained by the [illegible] study, two [illegible] of problems were [illegible] in the first series, [illegible] the value of [illegible], obtained by the SOFOCATE [illegible] was used. In the second series, [illegible] the value of [illegible] 900 [illegible] resonance [illegible] line of 1 ev in Pu^{240} [illegible] with the work of Crowther and Weil.[5] The [illegible] in the TURBO code does not [illegible] the [illegible]

A. Neutron Temperature

The temperature [illegible] the neutron [illegible] for the [illegible] Westcott's cross-section data was obtained from [illegible] formula:[6]

$$KT_n = KT_{mod}\left[1 + [illegible]\right]$$

B. Disadvantage Factor [illegible]

Thermal disadvantage factors were evaluated by P[illegible] spherical harmonic approximations.

C. Fast Fission Factor ε

The fast fission factor ε was obtained from the formula[7]

$$\varepsilon - 1 = [illegible]$$

where [illegible] cross sections are the values above the threshold and [illegible] is the fraction of the fission spectrum above the threshold. Also,

$$[illegible]$$

D. Neutron Age τ and Thermal Diffusion Constant D

τ and D were calculated by means of [illegible] factors.[8]

E. Resonance Escape Probability p

In calculating p, Hellstrand's experimental value[9] for the resonance integral of UO_2 was used, with Dancoff's self shielding corrections.

F. Void Dependence on Power

Void distributions were determined by physics-hydrodynamics-physics iterations.

APPENDIX

The differential equations used to obtain the isotopic concentrations of fuel material are:

$$\frac{dN^{25}}{d\tau} = -(N\hat{\sigma}_a)^{25} \quad ;$$

$$\frac{dN^{28}}{d\tau} = -(N\hat{\sigma}_a)^{28} \quad ;$$

$$\frac{dN^{26}}{d\tau} = (N\hat{\sigma}_a)^{25}\left(\frac{\alpha^{25}}{1+\alpha^{25}}\right) - (N\hat{\sigma}_a)^{26} \quad ;$$

$$\frac{dN^{49}}{d\tau} = (N\hat{\sigma}_a)^{28} + \sum_i (N\sigma_a)^i \eta^i \epsilon^i (1-p)e^{-\tau B^2} - (N\hat{\sigma}_a)^{49}$$

or

$$\frac{dN^{49}}{d\tau} = (N\hat{\sigma}_a)^{28} + \epsilon(1-p)\, e^{-\tau B^2} \sum_i (N\sigma_a)^i \eta^i - (N\hat{\sigma}_a)^{49} \quad ;$$

$$\frac{dN^{40}}{d\tau} = (N\hat{\sigma}_a)^{49} \frac{\alpha^{49}}{1+\alpha^{49}} - (N\hat{\sigma}_a)^{40} \quad ;$$

$$\frac{dN^{41}}{d\tau} = (N\hat{\sigma}_a)^{40} - (N\hat{\sigma}_a)^{41} \quad ;$$

$$\frac{dN^{42}}{d\tau} = (N\hat{\sigma}_a)^{41} \frac{\alpha^{41}}{1+\alpha^{41}} - (N\hat{\sigma}a)^{42} \quad ;$$

$$\frac{dN^{FP}}{d\tau} = \left[(N\hat{\sigma}_a)^{25}/(1+\alpha^{25}) + (N\hat{\sigma}a)^{49}/(1+\alpha^{49}) + (N\hat{\sigma}_a)^{41}/(1+\alpha)^{41}\right] \Big/ \left(1 + \frac{F.F.}{T.F.}\right) \quad ;$$

$$\phi(t) = \frac{K}{\Sigma_f^{25}(t) + \Sigma_f^{49}(t) + \Sigma_f^{41}(t)} \quad ,$$

where

$$\tau = \int_0^t \phi dt = \phi t$$

Numerics 25, 28, 49, 40, 41 and 42 refer to U^{235}, U^{238}, Pu^{239}, Pu^{240}, Pu^{241} and Pu^{242}, respectively. The index i under summation refers to U^{235}, Pu^{239}, and Pu^{241}.

$$K = 1.2 \times 10^{12} \quad .$$

The isotopic concentrations of the various elements defined by the above differential equations have been evaluated by the Analog Electric Computer.

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