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PERFORMANCE OF THORIUM FUELED FAST BREEDERS

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ABSTRACT

Three studies were performed to evaluate the breeding ratio of fast breeders containing thorium.

In a study of a small breeder, thorium metal and thorium oxide core designs were found to have similar breeding ratios. The slight advantage exhibited by the metal design was not considered significant since the design was based on a limited amount of thorium metal swelling data.

In a study of a 1200 MWe plant, a plutonium-uranium oxide design was compared to a uranium-thorium metal design. The uranium-thorium design had a lower breeding ratio, but also had a negative sodium void effect.

In the third study, the effect of replacing thorium oxide radial blankets with thorium metal radial blankets was evaluated. This was found to have little effect on the breeding ratio.

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INTRODUCTION

New emphasis has recently been placed on the thorium fuel cycle for fast breeder reactors. Of particular concern has been the breeding potential of such reactors.

This paper addresses that question by first comparing the performance of two designs, one designed for the use of plutonium-uranium oxide fuel and the other designed for the use of uranium-thorium metal fuel. The paper then compares the performance of two additional designs, one designed for the use of uranium-thorium oxide fuel and the other designed for the use of uranium-thorium metal fuel. Finally, the paper determines if a metal design for the radial blanket might improve the breeding ratio over that of an oxide blanket. Consequently, a metal blanket was designed and its performance was compared to that of an oxide blanket for three reactors containing different fuel systems.

The methods by which the metal fuel pins were designed and by which the performance of each reactor was calculated were identical throughout the three studies. These methods are detailed in the next two sections. A detailed description of each study then follows.

FUEL PIN DESIGN

The method used to size the metal pellet inside the cladding was the same in all three studies. Briefly, the pellet was sized such that the fuel/cladding gap would accommodate fuel swelling and the pellet with maximum burnup would just touch the cladding at end-of-life.

Fuel swelling was assumed to be isotopic with the temperature dependence shown in Figure 1.[1] The fuel pins were sodium bonded and the fuel centerline temperature was used to determine the swelling rate. This swelling rate was used up to 30% volumetric swelling. Above 30% volumetric swelling the fuel was assumed to be porous so that fission gases would be released and would not contribute to fuel swelling. The swelling rate under these conditions was taken to be 1% $\Delta V/V$ per atom percent burnup. No credit was taken for cladding diameter increases due to either swelling or creep. The plenum volume was assumed adequate to accommodate fission gas pressure so that cladding failure due to gas pressure loading was not a life limiting condition.

An example of this method is shown in Figure 2, where the fuel pellet diameter for the small breeder study is determined. The diametral measure of the initial gap and the amount of swelling at end-of-residence are plotted as a function of pellet diameter. For small diameters, the swelling is not adequate to close the gap and these designs, although viable, are not optimal. Beyond a diameter of 7.98 mm. (0.314 inches), the swelling is more than enough to close the gap, and these designs are unacceptable in accordance with the no-interaction criterion.

The pin outer diameter and cladding thickness were determined separately for each study and are discussed later in the appropriate sections.

PERFORMANCE CALCULATIONS

The procedure for calculating the performance parameters of a reactor was identical throughout the three studies.

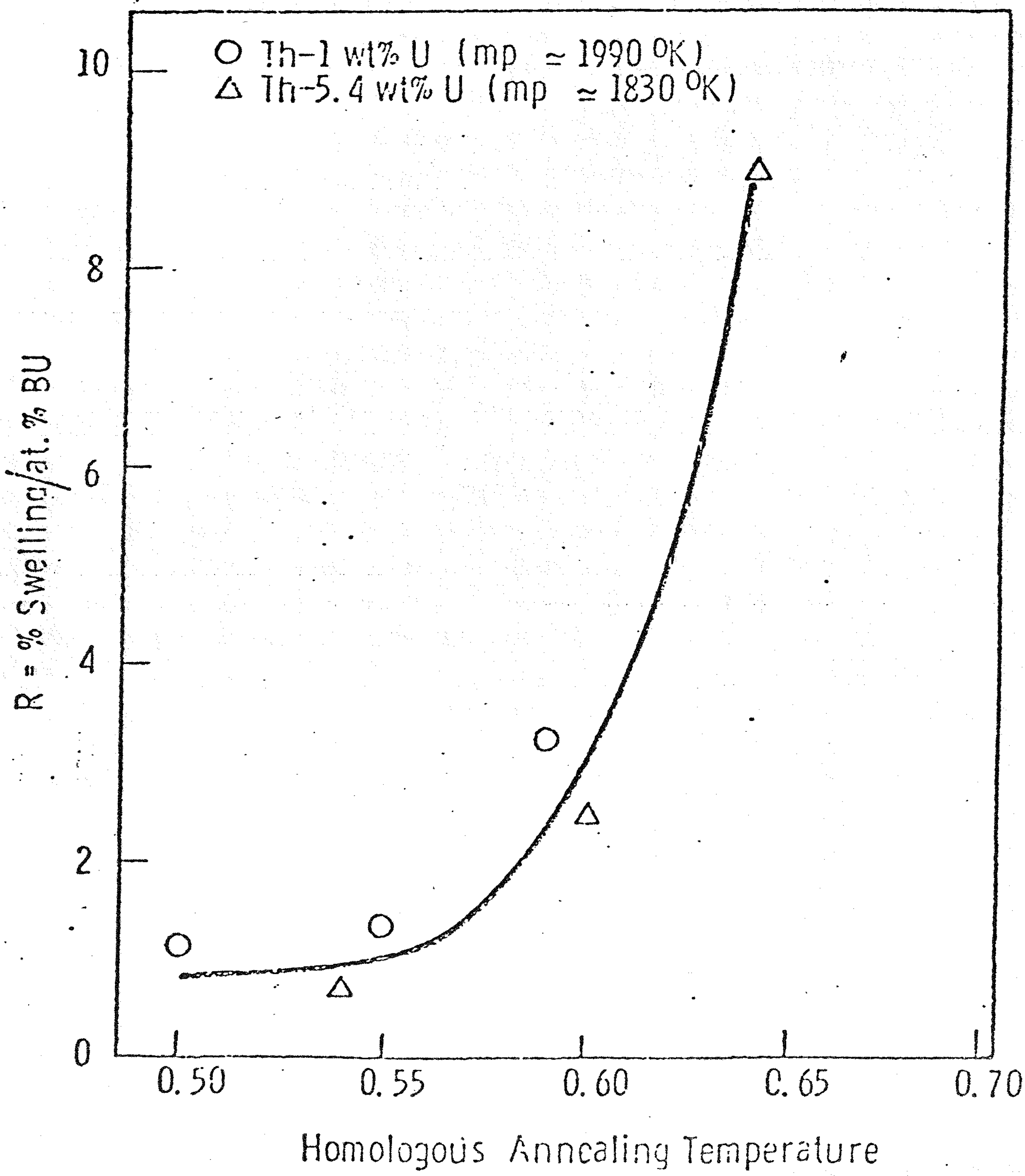


Figure 1. Thorium Metal Swelling Rate

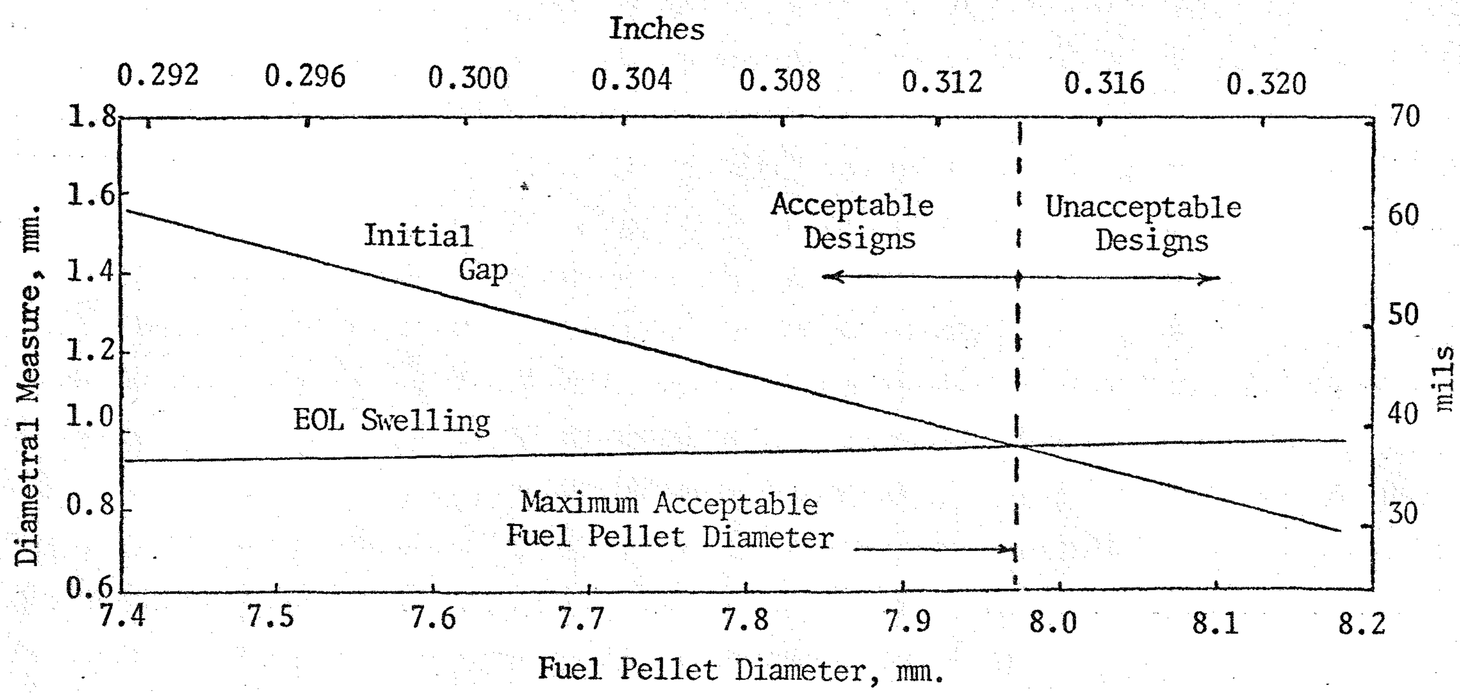


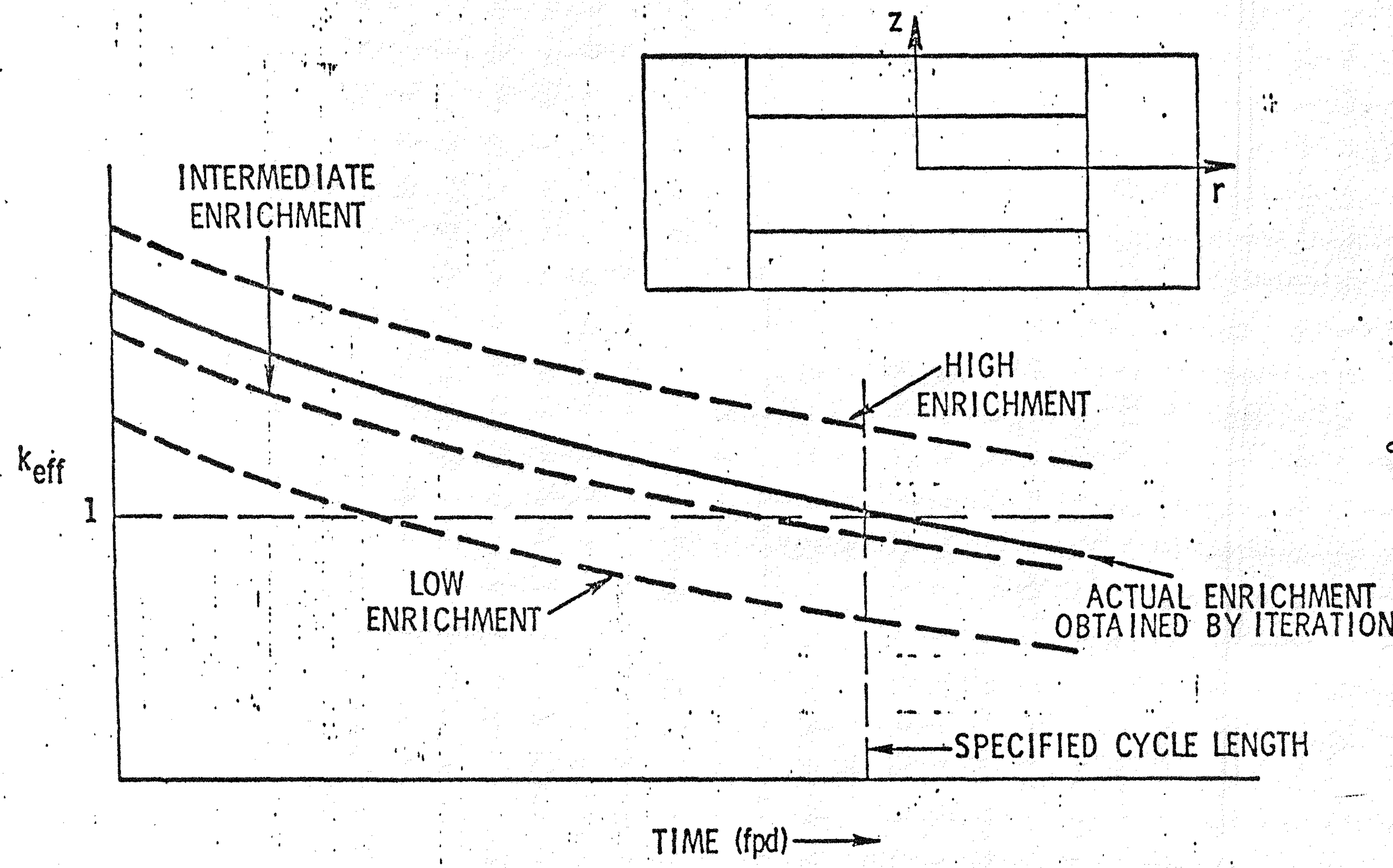
Figure 2. Fuel Pellet Diameter Determination

Two dimensional diffusion theory calculations using the computer code 2DB[2] with four energy groups were used to determine flux and power levels. The core was then burned in a depletion calculation in order to determine the local flux and power density as a function of time. The equilibrium feed enrichment was determined so as to provide no excess reactivity at the end of the equilibrium cycle with all the control rods withdrawn. As shown in Figure 3, several enrichments were tried and the final enrichment--that which would provide an end-of-cycle eigenvalue of unity--was determined by an iterative process. In the case of the two batch designs, one half of the core was scatter reloaded after the first cycle and an additional irradiation cycle was performed. This second cycle was then taken to be the equilibrium cycle unless additional cycles were required to establish radial blanket equilibrium.

FTR Set 300 cross sections[3] were used in all three studies. This 30 group cross section set in the Bondarenko form has proven equivalent to ENDF/B-III in the analyses of many fast critical assemblies[4], and was used in the Fast Test Reactor (FTR) design process. No ^{233}U lumped fission product cross section was available in this set. Based on an examination of the Bondarenko data[5], the ^{233}U lumped fission product was modeled with the ^{239}Pu fission product from Set 300 with the capture cross section reduced by 20%.

The isotope composition of the enriching material for the reactors fueled with uranium was assumed to be pure ^{233}U. The isotopic composition of the fissile material for those reactors fueled with plutonium was assumed to be LWR discharge.

Sodium void calculations were performed using the end-of-equilibrium cycle composition taken from the four group depletion calculation. However,



INPUT: FUEL PIN GEOMETRY, DUCT GEOMETRY, AND CROSS SECTIONS

CALCULATE: POWER DENSITY, NEUTRON FLUX, BREEDING RATIO, AND DOUBLING TIME

Figure 3. Reactor Physics Model

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because of the sensitivity of the sodium void calculation to spectrum effects, a twelve group cross section set was employed for the sodium void calculation. The four and twelve group energy structures are shown in Table 1. Separate twelve group sets were generated for the sodium containing and sodium voided compositions. The voiding pattern used to determine these compositions was the following. All flowing sodium in the active fuel region was voided; however, interduct gaps, control positions, and blanket assemblies were not voided. The end-of-equilibrium cycle reactivity was then calculated for each case, and the sodium void effect was obtained from the difference in k-effective.

Axial growth of the fuel column due to swelling during irradiation was not modeled. Slight modeling variations which do exist in the different studies are discussed later in their respective sections.

SMALL BREEDER STUDY

Introduction

This study sought to take advantage of the higher density and thermal conductivity of metal in order to improve breeding performance.

Other things being equal, a higher fuel density generally provides an increased fertile to fissile ratio, thus increasing the breeding ratio. However, the swelling of metal fuel is larger than that of the oxide fuel, and thus a redesign of the fuel pin is required to assure adequate lifetime.

In spite of a lower melting point for metal fuel, the higher thermal conductivity allows one to design a larger diameter fuel pin. Because of the thermal hydraulic advantage enjoyed by large pins, the pitch-to-diameter ratio may be decreased. This results in a higher fuel packing density which again improves the breeding ratio.

ENERGY LIMIT (EV)	30 GROUP	12 GROUP	4 GROUP
1.000 + 07			
6.065 + 06	1	1	
3.679 + 06	2		
2.231 + 06	3	2	1
1.353 + 06	4		
8.209 + 05	5	3	
4.979 + 05	6		
3.877 + 05	7	4	
3.020 + 05	8		
1.832 + 05	9	5	2
1.111 + 05	10		
6.738 + 04	11	6	
4.087 + 04	12		
2.554 + 04	13	7	
1.989 + 04	14		
1.503 + 04	15		3
9.119 + 03	16	8	
5.531 + 03	17		
3.355 + 03	18		
2.840 + 03	19	9	
2.404 + 03	20		
2.035 + 03	21		
1.234 + 03	22	10	
7.485 + 02	23		4
4.540 + 02	24	11	
2.754 + 02	25		
1.301 + 02	26		
6.144 + 01	27		
1.371 + 01	28	12	
6.826 - 01	29		
THERMAL	30		

Table 1. Energy Boundaries Of The Cross Section Sets

This study compares the breeding performance of a given reactor core for each of the two fuel forms, allowing only pin and bundle design differences for the respective fuel forms.

Core and Assembly Design

The core design used in this study was a small heterogeneous core composed of 156 fuel assemblies with 73 internal blanket assemblies. The core configuration, which possesses third core symmetry is shown in Figure 4. In addition to the configuration, the subassembly pitch, and duct design were fixed for the study, and only the pin and bundle designs were changed.

The oxide and metal fuel assembly designs are summarized in Table 2. The metal design contained a 91 pin bundle composed of fuel pins with an outer diameter of 9.70 mm. (0.382 inches). The oxide design contained 169 pins with a pin diameter of 6.99 mm. (0.275 inches). With the average subassembly power fixed, the difference in the number of pins per assembly reflects the difference in thermal conductivities of the two fuel forms.

The radial blanket assembly design was structurally the same for the two designs, containing 37 pins with a pin outer diameter of 16.4 mm. (0.647 inches). The fuel pellet diameter and diametral gap do differ, however, and the metal design was sodium bonded. Details of the two designs are given in Table 3.

Physics Models

The reactor was modeled in R-Z geometry with midplane symmetry. Internal blankets were homogenized into equivalent volume annular rings. Control positions were modeled as sodium filled ducts in both the core and axial blankets. These control positions were homogenized into the

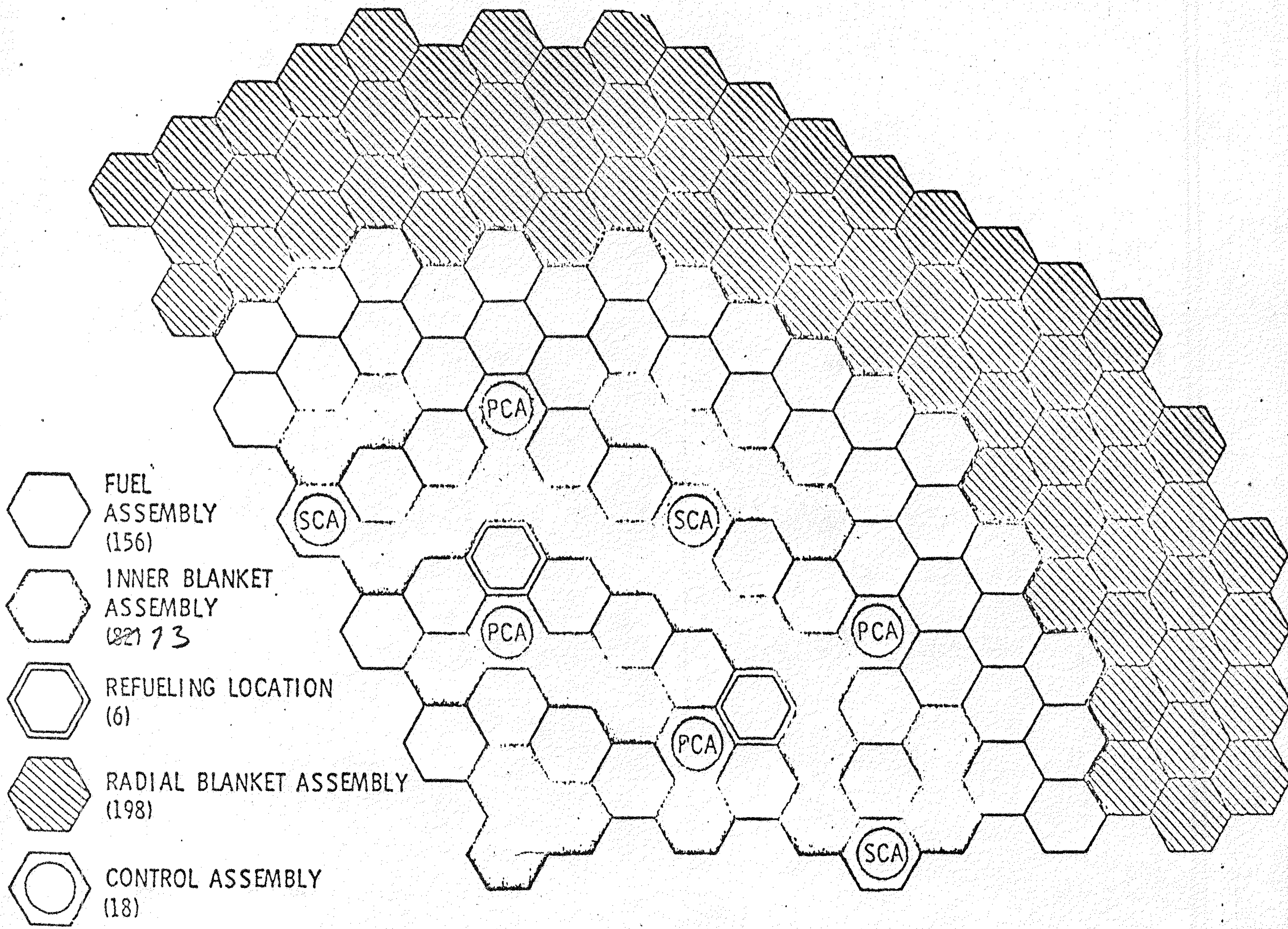

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Figure 4. Small Breeder Core Layout

Table 2

FUEL ASSEMBLY DESIGN

Fuel Material	U3,Th	$(U3,Th)O_2$
Theoretical Density - g/cc	12.8	10.0
Pins/SA	91	169
Blanket Material	Th	ThO_2
Theoretical Density - g/cc	11.7	10.0
Fuel Rod O.D. - mm. (in.)	9.70(0.382)	6.99(0.275)
Cladding Wall - mm. (mils)	0.381(15.0)	0.394(15.5)
Fuel-To-Cladding Gap - mm.(mils)	0.965(38)	0.203(8)
Gap Bond Material	Sodium	Helium
Pellet O.D. - mm. (in.)	7.98 (0.314)	6.01 (0.2365)
Smear Density - %TD	79.6	91.6
Fuel Pin P/D	1.154	1.189
Figure of Merit	4.60	3.24

Table 3

RADIAL BLANKET ASSEMBLY DESIGN

Blanket Material	Th	ThO_2
Theoretical Density-g/cc	11.7	10.0
Pins/SA	37	37
Pin O.D.-mm. (in.)	16.4(0.647)	16.4(0.647)
Cladding Wall-mm.(mils)	0.381(15)	0.381(15)
Diametral Gap-mm.(mils)	0.864(34)	0.178(7)
Gap Bond Material	Sodium	Helium
Pellet O.D.-mm.(in.)	14.8(0.583)	15.5(0.610)
Smear Density-%TD	89.3	95.3
Pin P/D	1.07	1.07
Figure of Merit	5.89	4.72

respective fuel and axial blanket annuli. The control rods were assumed to be fully withdrawn, and consequently, no boron was present in the calculation. The row outside the radial blanket was assumed to be an Inconel reflector with the composition shown in Table 4. One foot of plenum was included in the calculation to provide axial reflection. The plenum was modeled as empty fuel pins. Axial blankets were taken to be 43.2 cm. (17 inches) long.

The isotopic transmutation model is shown in Figure 5. ^{233}Pa is modeled explicitly so the end of cycle reactivity includes the effect of the ^{233}Pa holdup.

The fuel management scheme was based on the AFMS strategy for CRBR. There was a complete reload of the fuel and internal blankets shown in Figure 5 every two years. Radial blanket residence was assumed to be six years with one third replaced every two years. A 75% full power capacity factor was used.

Results

The performance of the metal and oxide designs is shown in Table 5. The metal design shows a better breeding performance with a breeding ratio of 1.14, slightly higher than the 1.10 exhibited by the oxide design. There is, however, a large uncertainty in the performance of the metal design due to uncertainty in the metal swelling data.

The sodium void at end-of-equilibrium cycle is negative for both designs with a value of approximately three dollars.

The annual pin fabrication and heavy metal reprocessing requirements are also shown in Table 5. With the smaller pin size, the oxide design incurs a fabrication cost penalty due to the large number of pins that must be fabricated annually. The metal design, on the other hand, incurs

Table 4

RADIAL REFLECTOR COMPOSITION

Material	Atoms/barn-cm.
Na	0.00204
Fe	0.02163
Cr	0.01394
Ni	0.04164

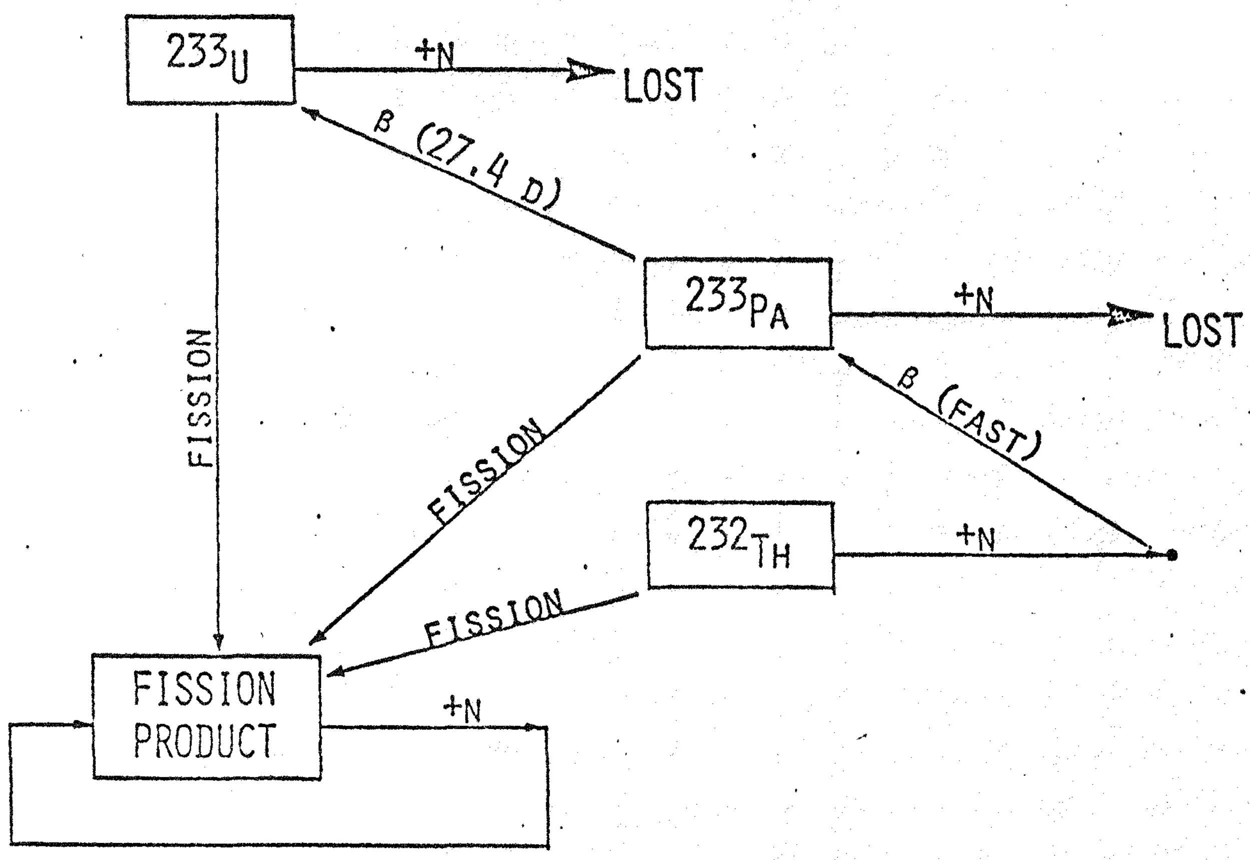

Figure 5. Depletion Model for Small Breeder Study

Table 5

PERFORMANCE CHARACTERISTICS

Fuel Material	U3,Th	$(U3,Th)O_2$
Peak Heat Rate-Kw/m(Kw/ft)	104(31.6)	54.1(16.5)
Residence Time-yrs	2	2
Average Discharge Exposure-MWD/Kg	60	76
Peak Fluence (E>0.1 MeV)	1.56×10^{23}	1.23×10^{23}
Fuel Enrichment (U/U+Th) (wt%)	23.2	25.3
Core Enrichment (U/U+Th) (wt%)	14.7	15.9
Core Conversion Ratio	0.61	0.63
Breeding Ratio	1.14	1.10
Doubling Time-yrs	95	140
EOEC Sodium Void-$	-3	-3
Fuel Pins/Year	7098	13182
Kgs Heavy Metal/Year	13098	10323

a reprocessing cost penalty due to the higher heavy metal flow.

LARGE BREEDER STUDY

Introduction

This study evaluated the breeding performance and sodium void worth of a large breeder fueled with thorium metal by comparing it to a reference plutonium-uranium oxide design.

Core and Assembly Design

The HEDL Large Heterogeneous Reference Fuel Design Study (LHRFDS) Level I Homogeneous design[6] was chosen for the reference plutonium-uranium oxide design. This is a conservative design utilizing the reference CRBR pin and assembly and has a compound system doubling time of 36 years. It is composed of a two zone core of 678 fuel subassemblies each containing 217 pins 5.84 mm. (0.230 inches) in diameter in a 11.5 cm. (4.535 inch) duct. The core layout is shown in Figure 6. For the thorium metal design, these 678 assemblies were replaced with the thorium metal assembly design described in the previous study of a small breeder. Details of the core and fuel assembly designs are given in Tables 6 and 7. The radial blanket designs are shown in Table 8.

Physics Models

The core was modeled in R-Z geometry with two homogeneous core zones, a radial blanket zone, a radial reflector zone, an axial blanket zone, and an axial reflector zone. Midplane symmetry was used to cut calculational expense. The 35.6 cm. (14 inch) lower axial blanket and the 53.3 cm. (21 inch) upper axial blanket were modeled by reflection as a single 44.5 cm. (17.5 inch) thick region. All the control positions were homogenized into the first core zone with each control position taken to have the composition

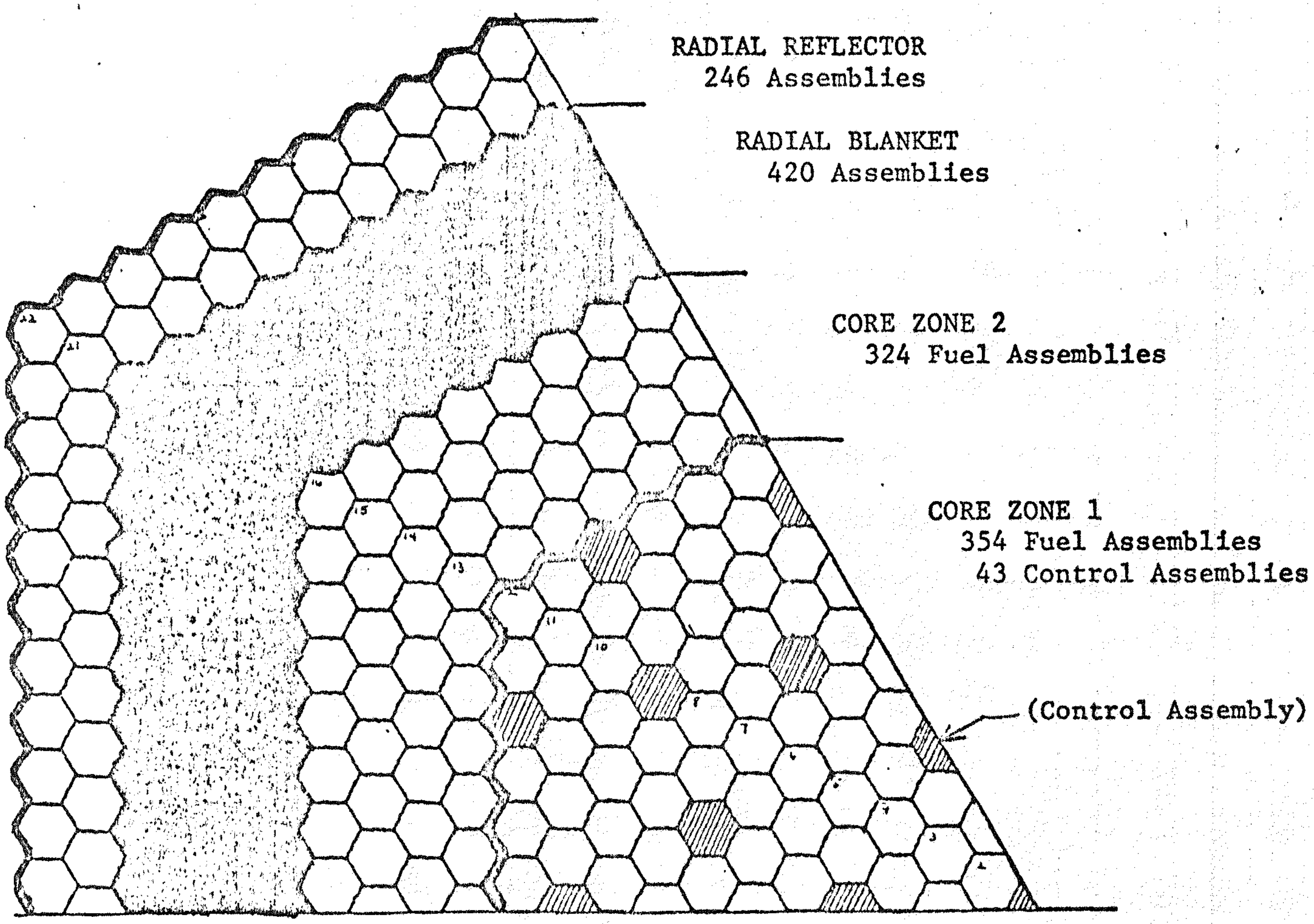

FIGURE 6. LHRFDS Level I Homogeneous Core Map.

Table 6

LARGE BREEDER CORE DESIGN

Fuel Type	U3,Th	$(Pu,U8)O_2$
Electric Power-MWe	1200	1200
Reactor Thermal Power-MWt	3318	3318
Coolant Inlet Temperature-°C(°F)	380(716)	380(716)
Coolant Outlet Temperature-°C(°F)	518(965)	518(965)
Core Height-cm. (in.)	91.4(36)	91.4(36)
Axial Blanket-cm. (in.)	88.9(35)	71.1(28)
Lattice Pitch-cm. (in.)	12.09(4.76)	12.14(4.78)
Duct O.D.-cm.(in.)	11.52(4.535)	11.52(4.535)
Duct Wall-mm.(mils)	3.05(120)	3.05(120)
No. Pins/SA	91	217
No. Fuel Assemblies	678	678
No. Radial Blanket Assemblies	426	420

Table 7

LARGE BREEDER FUEL ASSEMBLY DESIGN

Fuel Material	U3,Th	$(Pu,U8)O_2$
Theoretical Density-g/cc	12.2	11.0
Pins/SA	91	217
Blanket Material	Th	UO_2
Theoretical Density-g/cc	11.7	11.0
Fuel Rod O.D.-mm.(in.)	9.70(0.382)	5.84(0.230)
Cladding Wall-mm.(mils)	0.381(15)	0.381(15)
Fuel-To-Cladding Gap-mm.(mils)	0.965(38)	0.165(6.5)
Gap Bond	Na	He
Pellet O.D.-mm.(in.)	7.98(0.314)	4.92(0.1935)
Smear Density-%TD	79.6	85.5
Fuel Pin P/D	1.154	1.244
Figure of Merit	4.38	2.85

Table 8

LARGE BREEDER RADIAL BLANKET ASSEMBLY DESIGN

Blanket Material	Th	UO_2
Theoretical Density-g/cc	11.7	11.0
Pins/SA	37	61
Pin O.D.-mm.(in.)	16.4(0.647)	12.9(0.506)
Cladding Wall-mm.(mils)	0.381(15)	0.381(15)
Diametral Gap-mm.(mils)	0.864(34)	0.178(7)
Gap Bond Material	Na	He
Pellet O.D.-mm.(in.)	14.8(0.583)	11.9(0.469)
Smear Density-%TD	89.3	93.7
Pin P/D	1.07	1.07
Figure of Merit	5.89	5.19

of a fuel assembly voided of fuel. The control rods were assumed to be fully withdrawn, and consequently no boron present in the calculation. One foot of plenum--i.e., voided fuel pins, was modeled as a homogeneous axial reflector. Two rows of radial reflector with the composition shown in Table 4 were used.

The fuel depletion models are shown in Figures 7 and 8. Note that the ^{233}Pa was not explicitly modeled, so that a capture in thorium immediately produced ^{233}U. Note, however, that the higher uranium isotopes were modeled in detail. Explicitly including ^{233}Pa would have decreased the breeding ratio slightly.

Fuel residence time was two years with annual refueling of the two batch core. A 70% full power capacity factor was used. Radial blanket residence time was set at 5 years.

Results

The performance of the uranium-thorium metal design is compared in Table 9 to the performance of the plutonium-uranium oxide design. The breeding ratio (1.11) of the metal design is significantly lower than that of the oxide design (1.17). Since doubling time is an inverse function of breeding gain--that is, the breeding ratio minus one--the doubling time is disproportionately larger for the metal design. As Table 9 shows, the doubling time for the uranium-thorium metal design is 62 years versus 37 years for the oxide design.

The sodium void effect of the uranium-thorium metal design was considerably lower than that of the plutonium-uranium oxide design--i.e., minus two dollars at end-of-cycle versus plus four dollars for the oxide design.

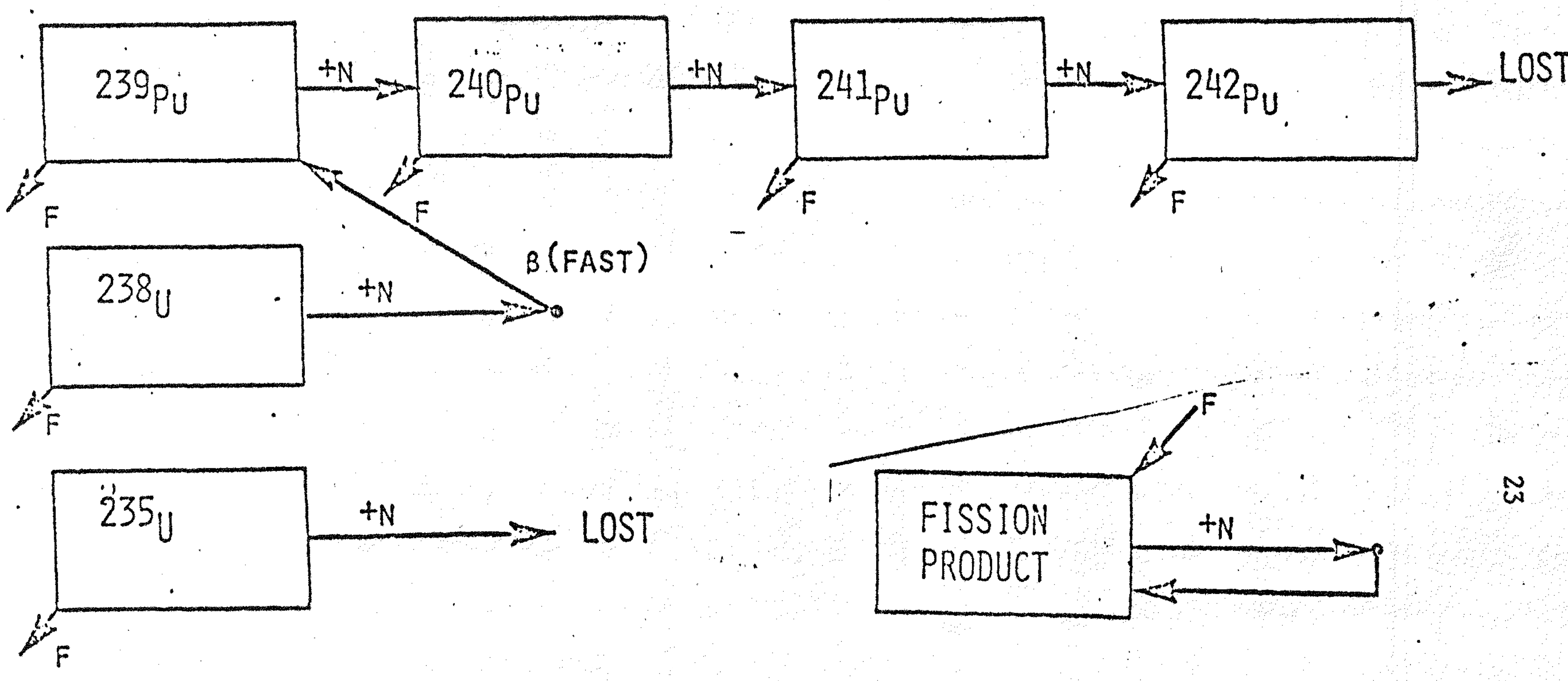

Figure 7. Large Breeder Depletion Model - Plutonium Cycle

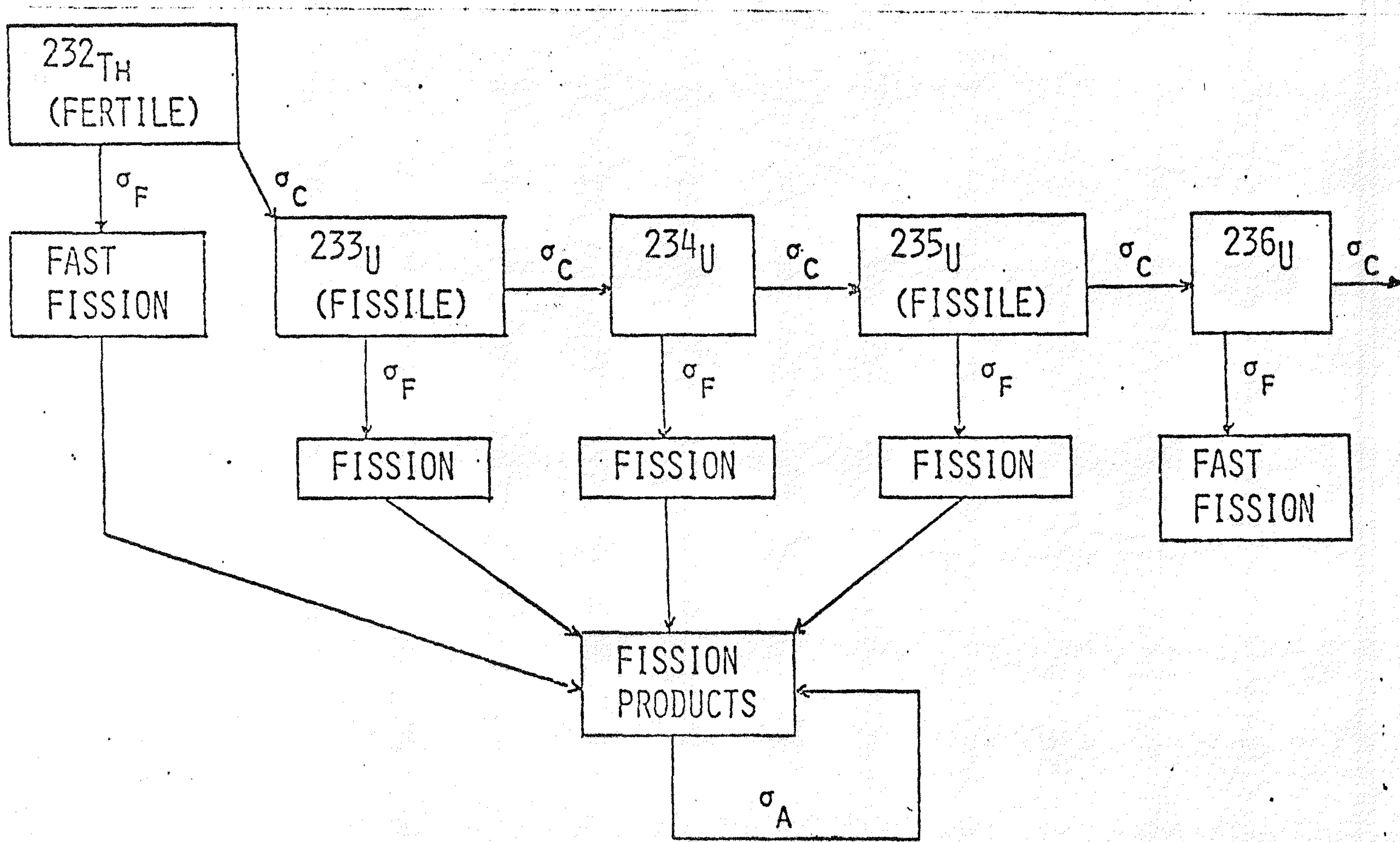

Figure 8. Large Breeder Depletion Model - Thorium Cycle

Table 9

METAL CORE PERFORMANCE

Fuel Material	U3,Th	$(Pu,U8)O_2$
Peak Heat Rate-Kw/m(Kw/ft)	90.2(27.5)	35.4(10.8)
Residence Time-yrs	2	2
Ave. Discharge Exposure -MWD/Kg	40.4	69
Peak Fluence (E>0.1 MeV)	$2.65x10^{23}$	$1.87x10^{23}$
Fuel Enrichment (%)	11.4/14.2	18.1/21.8
Core Conversion Ratio	0.70	0.74
Breeding Ratio	1.11	1.17
Doubling Time-yrs	61.7	36.5
EOEC Sodium Void-$	-2.11	+4.28
Fuel Pins/Yr (Core Only)	29939	73563
Kgs Heavy Metal/Yr (Core & Blkts)	39105	27849

As in the small breeder study, the small pin oxide design has a higher fabrication cost, while the large pin metal design has a higher reprocessing cost.

RADIAL BLANKET STUDY

Introduction

This study investigated the possibility that a metal design for the radial blanket might show an advantage over an oxide design, again because of increased fuel density. The effect of changing the blanket was investigated for three different reactors. These reactors had identical core designs, but employed different fuel systems. The fuel systems were: (1) a U3/U8 oxide core with a uranium oxide axial blanket, (2) a U3/U8 oxide core with a thorium oxide axial blanket, and (3) a U3/Th oxide core with a thorium oxide axial blanket.

Core Design

The core layout and fuel assembly designs used throughout this study are those of a low temperature early PLBR design[7]. This was a two zone homogeneous design with a doubling time of 15 years when fueled with plutonium-uranium oxide. Details of this design are given in Table 10.

The fuel assembly designs for each of the three fuel systems used in this study are shown in Table 11. Differences are only in the fuel and blanket loadings. The first two designs use ^{233}U fuel denatured with ^{238}U, one with a uranium oxide axial blanket and the other with a thorium oxide axial blanket. The third fuel system is ^{233}U in thorium oxide fuel with a thorium oxide axial blanket.

The radial blanket design for the oxide system is shown in Table 12. The redesign for the metal blanket proceeded as follows. A 61 pin

Table 10

BLANKET STUDY-CORE DESIGN

Core Layout	Low Temperature Early PLBR
Electric Power-MWe	1200
Thermal Power-MWt	3736
Coolant Inlet-°C (°F)	327 (620)
Coolant Outlet-°C(°F)	482 (900)
Core Height-cm. (in.)	119 (46.8)
Axial Blanket-cm. (in.)	66.0 (26.0)
Lattice Pitch-cm. (in.)	15.8 (6.22)
Duct OD-cm. (in.)	15.0 (5.917)
Duct Wall (Mils)-mm. (mils)	2.13 (84)
# Fuel Assemblies	380
# Blanket Assemblies	184

Table 11

BLANKET STUDY - FUEL ASSEMBLY DESIGNS

Fuel Material	$(U3,U8)O_2$	$(U3,U8)O_2$	$(U3,Th)O_2$
Theoretical Density-g/cc	11.0	11.0	10.1
Pins/SA	271	271	271
Blanket Material	UO_2	ThO_2	ThO_2
Theoretical Density-g/cc	11.0	10.0	10.0
Fuel Rod O.D.-mm. (in.)	7.26 (0.286)	7.26 (0.286)	726 (0.286)
Cladding Wall-mm. (mils)	0.305(12)	0.305(12)	0.305(12)
Diametral Gap-mm. (mils)	0.140(5.5)	0.140(5.5)	0.140(5.5)
Gap Bond Material	He	He	He
Pellet O.D.-mm. (in.)	6.52(0.2565)	6.52(0.2565)	6.52(0.2565)
Smear Density - %TD	88	88	88
Fuel Pin P/D	1.199	1.199	1.199
Figure of Merit	3.72	3.72	3.42

Table 12

RADIAL BLANKET ASSEMBLY DESIGNS

Blanket Material	ThO_2	Th
Theoretical Density-g/cc	10.0	11.7
Pins/SA	127	61
Pin O.D.-mm. (in.)	11.8(0.4657)	16.9(0.667)
Cladding Wall-mm. (mils)	0.381(15)	0.381(15)
Diametral Gap-mm. (mils)	0.178 (7)	0.770 (30.3)
Gap Bond Material	He	Na
Pellet O.D.-mm. (mils)	10.9(0.4287)	15.4(0.6067)
Smear Density - %TD	93.7	-
Pin P/D	1.07	1.07
Figure of Merit	4.65	6.16

assembly was selected. Then the pin diameter was determined such that the coolant pressure drop was identical to that of the reference oxide design. With the cladding set at 0.381 mm. (15 mils), the blanket pellet size was then determined as previously described,--i.e., such that no fuel-cladding interaction would occur at end-of-life. This metal blanket pin design is also shown in Table 12.

Physics Models

The reactor model used in 2DB was an R-Z model with two homogeneous core zones. Midplane symmetry was used to reduce calculational expense. Control positions were equally spread throughout the core and homogenized with the fuel assemblies.

Control assemblies were taken to be sodium filled ducts, both in the core and in the axial blankets. The control rods were assumed to be fully withdrawn, and consequently no boron was present in the calculation. One row of radial reflector with the composition shown in Table 4 was used outside the radial blanket. In contrast to the other studies, axial reflection from areas outside the axial blankets was ignored. Axial growth of the fuel column was not modeled during the depletion calculation.

The fuel depletion model is shown in Figure 9. Note that ^{233}Pa is not explicitly modeled, so that a capture in thorium immediately produced ^{233}U. This would cause a slight overestimation of the breeding ratio since the feed enrichment would be slightly underestimated.

The fuel management scheme was a two batch core with a residence time of 900 days at a 70% capacity factor. The blanket residence time was 4500 days. The uranium in the core was assumed to be obtained from the tails stockpile, was enriched with pure ^{233}U.

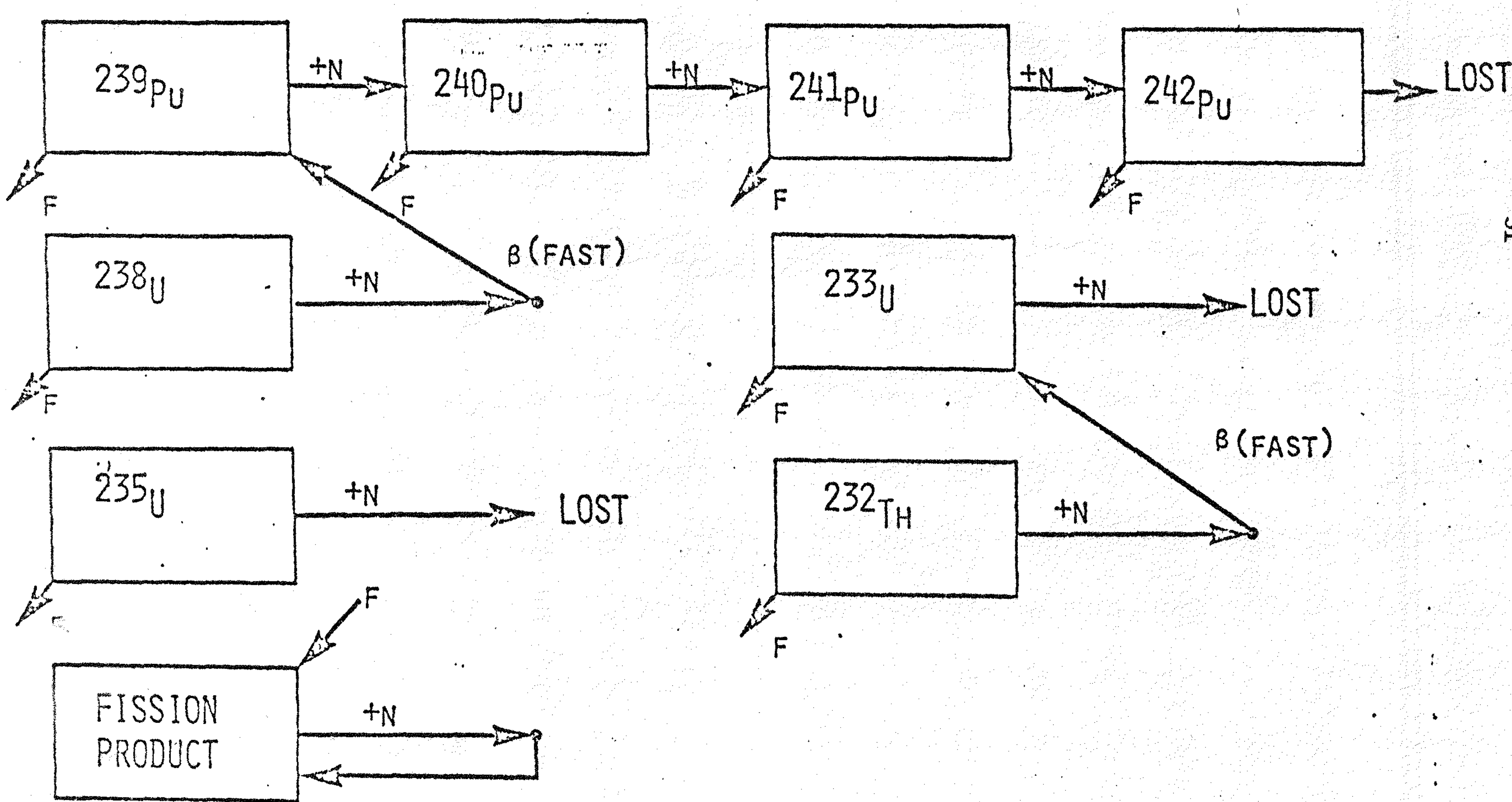

Figure 9. Depletion Model for Blanket Study

Results

The performance characteristics of each blanket for the three fuel systems are shown in Tables 13 through 15. There was no notable improvement in the breeding characteristics of any of these fuel systems by changing to a metal blanket.

CONCLUSIONS

A uranium-thorium metal design shows a higher breeding ratio and lower doubling time than that shown by a uranium-thorium oxide design. However, the difference is small and since the metal design is based on a limited amount of swelling data, the conclusion may be subject to revision.

A uranium-thorium metal design has a lower breeding ratio and longer doubling time than a conservative plutonium-uranium oxide design.

There seems to be no breeding ratio advantage associated with the use of thorium metal blankets in place of thorium oxide radial blankets.

Table 13

U3-U8/U8/Th PERFORMANCE CHARACTERISTICS

	Oxide Blanket	Metal Blanket
K_{eff} - BEC	1.053	1.052
- EEC	1.001	1.000
Breeding Ratio-MEC	1.187	1.190
Conversion Ratio-MEC	0.895	0.893
Fissile Load-Kg	3644	3622
Enrichment (U/U+Th)-w/o		
Core 1	8.71	8.75
Core 2	11.32	11.36
Doubling Time - Yrs	22.8	22.5

Table 14

U3-U8/Th/Th PERFORMANCE CHARACTERISTICS

	Oxide Blanket	Metal Blanket
K_{eff} - BEC	1.052	1.053
- EEC	1.000	1.000
Breeding Ratio-MEC	1.180	1.180
Conversion Ratio-MEC	0.890	0.885
Fissile Load-Kg	3666	3696
Enrichment (U/U+Th)-w/o		
Core 1	8.76	8.83
Core 2	11.39	11.47
Doubling Time - Yrs	24.8	24.8

Table 15

U3-Th/Th/Th PERFORMANCE CHARACTERISTICS

	Oxide Blanket	Metal Blanket
K_{eff} - BEC	1.035	1.038
- EEC	0.999	1.000
Breeding Ratio-MEC	1.063	1.058
Conversion Ratio-MEC	0.789	0.781
Fissile Load-Kg	4142	4194
Enrichment (U/U+Th)-w/o		
Core 1	10.86	10.99
Core 2	14.17	14.35
Doubling Time - Yrs	185	222

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