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KAPL-4107

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Fuel Utilization Potential In Light Water Reactors With Once-Through Fuel Irradiation Using Various Reactivity Control Methods

(AWBA Development Program)

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Operated for the United States
Department of Energy
GENERAL ELECTRIC

1978

KAPL-4107

FUEL UTILIZATION POTENTIAL IN LIGHT WATER REACTORS WITH ONCE-THROUGH FUEL IRRADIATION USING VARIOUS REACTIVITY CONTROL METHODS

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September 1978

General Electric Company
KNOLLS ATOMIC POWER LABORATORY
Schenectady, New York
Operated for the
United States Department of Energy
Contract No. EY-76-C-12-0052

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FOREWORD

The Shippingport Atomic Power Station located in Shippingport, Pennsylvania was the first large-scale, central-station nuclear power plant in the United States and the first plant of such size in the world operated solely to produce electric power. This project was started in 1953 to confirm the practical application of nuclear power for large-scale electric power generation. It has provided much of the technology being used for design and operation of the commercial, central-station nuclear power plants now in use.

Subsequent to development and successful operation of the Pressurized Water Reactor in the DOE-owned reactor plant at the Shippingport Atomic Power Station, the Atomic Energy Commission in 1965 undertook a research and development program to design and build a Light Water Breeder Reactor core for operation in the Shippingport Station. In 1976, with fabrication of the Light Water Breeder Reactor (LWBR) nearing completion the Energy Research and Development Administration established the Advanced Water Breeder Applications program (AWBA) to develop and disseminate technical information which would assist U.S. industry in evaluating the LWBR-concept. All three of these reactor development projects have been administered by the Division of Naval Reactors with the goal of developing practical improvements in the utilization of nuclear fuel resources for generation of electrical energy using water-cooled nuclear reactors.

The objective of the Light Water Breeder Reactor project has been to develop a technology that would significantly improve the utilization of the nation's nuclear fuel resources employing the well-established water reactor technology. To achieve this objective, work has been directed toward analysis, design, component tests, and fabrication of a water-cooled, thorium oxide fuel cycle breeder reactor to install and operate at the Shippingport Station. Operation of the LWBR core in the Shippingport Station started in the Fall of 1977. After about 3 to 4 years of power operation, the LWBR core modules will be removed from the Shippingport reactor vessel and shipped to the Expended Core Facility at the Naval Reactors Facility in Idaho for a detailed core examination and determination of breeding performance.

The Advanced Water Breeder Applications (AWBA) project was initiated to develop and disseminate technical information that will assist U.S. industry in evaluating the LWBR concept for commercial-scale applications. The project will explore some of the problems that would be faced by industry in adapting technology confirmed in the LWBR program. Information to be developed includes concepts for commercial-scale prebreeder cores which will produce uranium-233 for light water breeder cores while producing electric power, improvements for breeder cores based on the technology developed to fabricate and operate the Shippingport LWBR core, and other information and technology to aid in evaluating commercial-scale application of the LWBR concept.

Technical information developed under the Shippingport, LWBR, and AWBA projects has been and will continue to be published in technical memoranda, one of which is this present report.

ABSTRACT

Most of the nuclear power plants now in operation or under construction in this country are light water reactors (LWRs) operating without fuel recycle, i.e., on a once-through basis. These LWRs are comprised of pressurized water reactors (PWRs) and boiling water reactors (BWRs). To significantly reduce the projected uranium demand, the uranium fuel utilization of the LWRs must be improved.

The purpose of this study is to determine the maximum fuel utilization, as measured by energy production per short ton of U_3O_8 (Mwy/st U_3O_8),* achievable by reactor design optimization of a practical once-through PWR. The areas examined for potential gains include the following: reactivity control systems, increased depletion, and the thorium fuel cycle. Although similar gains in fuel utilization could likely be achieved in a BWR, this report concentrates on PWRs. Although fuel utilization could be improved by reducing the diffusion plant tails assay, this report which deals with reactor design has assumed that tails assay is held at 0.2% U-235.

The following conclusions are drawn from this study:

1. The use of movable fuel for reactivity control results in gains in fuel utilization compared with present types of LWR concepts.
2. For average fuel depletions up to 50,000 Mwd/mt,** the uranium fuel cycle requires less U_3O_8 consumption than the thorium fuel cycle when the same basic reactor concept is used for both.
3. The gain in fuel utilization due to increased average depletion is small for the uranium fuel system in the range of 35,000 to 50,000 Mwd/mt; for the thorium fuel system, an increase of ~10% is calculated over the same depletion range.
4. The maximum gain in fuel utilization for a practical PWR which would incorporate the use of movable fuel without changing the design of the core reflector is about 30%. If a PWR were designed with a power producing exterior blanket that could reduce the neutron leakage from 4% (characteristic of present reactors) to 1%, a maximum overall gain of 50 to 60% in fuel utilization might be realized.

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*Mwy/st — Megawatt years per short ton.

**Mwd/mt — Megawatt days per metric ton.

ACKNOWLEDGMENT

In addition to the principal investigators, the following individuals in the KAPL AWBA Development Program contributed to this report: D. G. Blondin, W. A. Boyd, and J. R. Dwyer.

FUEL UTILIZATION POTENTIAL IN LIGHT WATER REACTORS WITH ONCE-THROUGH FUEL IRRADIATION USING VARIOUS REACTIVITY CONTROL METHODS (AWBA DEVELOPMENT PROGRAM)

F. C. Merriman, D. F. McCoy, H. J. Capossela, M. R. Mendelson

I. INTRODUCTION

A. Purpose and Scope

The purpose of this study is to determine the maximum fuel utilization achievable by reactor design optimization of a once-through converter-burner pressurized light water reactor (LWR). The features compared in this study include (1) uniform and seed-blanket cores, (2) thorium and uranium fuel cycles, (3) use of denatured (20% U-235 and 80% U-238) and highly enriched fuels in seed regions, (4) fuel management and batch loading, (5) soluble boron and movable fuel reactivity control methods, and (6) effect of extended irradiation limits of fuel elements. Although diffusion plant tails assay affects fuel utilization, this study of reactor design concepts assumes that the tails assay is fixed at 0.2% U-235.

Four sources of data were used to evaluate the effects of the above features and to support the conclusions of this study:

1. Results of previous KAPL work on highly enriched seed-blanket converter-burners. These data were extrapolated to incorporate the desired design features using the single-rod cell results described in Item 2 and the seed-blanket survey calculations described in Item 3. The results are described in Section II.
2. Single-rod cell calculations. These results, described in Section III, were used to provide the basic performance data on UO_2 and ThO_2 cores. The effects of various reactivity control methods, fuel management, and increased fuel depletion are included in these performance data.
3. Calculations based on an analytical model of an idealized seed-blanket reactor. These results, described in Section IV, were used to evaluate denatured fuel vs highly enriched fuel in both the thorium and uranium fuel cycles. The effect of fuel management was also studied with this method.
4. Numerical seed-blanket cell calculations using a diffusion theory computer code. These results, described in Sections IV and V, were used to a limited extent to verify the results of the analytical model approach above and to augment the results of the single-rod unit cell and survey calculations.

To determine the gain in fuel utilization of an optimized converter-burner compared with existing pressurized LWRs, data for the latter must first be established. The values shown in Table 1 indicate typical mining requirements in terms of a quantity E, which is the megawatt-years of thermal energy produced per short ton (Mwy/st) of U_3O_8 mined for a UO_2 core depleted to 30,400 Mwd/mt (megawatt days per metric ton) and for a ThO_2 core depleted to 33,800 Mwd/mt. These data are based on Reference 1, which compares the thorium fuel cycle with the uranium fuel cycle in the Combustion Engineering System 80 reactor. Table 1 also shows the fuel utilization for UO_2 and ThO_2 cores with an extrapolated average depletion of 50,000 Mwd/mt.

TABLE 1. FUEL UTILIZATION DATA FOR PRESSURIZED LIGHT WATER REACTORS
(SOLUBLE BORON CONTROL AND THREE CYCLE FUEL MANAGEMENT)

Fuel	Fuel Utilization: Average Mwd/mt	Fuel Utilization: Mwy/st U_3O_8
Based on Reference 1		
UO_2	30,400	11.7
ThO_2	33,960	9.1
ThO_2	24,124	8.1
ThO_2	10,300	4.7
Extrapolated to 50,000 Mwd/mt		
UO_2	50,000	12.4
ThO_2	50,000	10.7

B. Summary of Results

The results of the studies performed to determine the uranium fuel utilization achievable in a once-through converter-burner pressurized water reactor show that on an optimistic basis a fuel utilization of about 18.2 Mwy/st U_3O_8 appears achievable. This value is 50 to 60% greater than currently achieved in commercial pressurized water reactor core designs (e.g., see the Table 1 value of 11.7 Mwy/st U_3O_8 for a current UO_2 system).

The maximum value of 18.2 Mwy/st U_3O_8 is obtained in the following manner:

1. Optimistic scoping calculations were performed based on a single-rod cell model with no allowance for leakage. The results of these studies indicate that values of 15.8 and 19.3 Mwy/st U_3O_8 could be obtained with soluble boron control and movable fuel control, respectively. These results correspond to reasonable fuel management assumptions but do not account for neutron leakage that significantly reduces performance.

2. Allowing for leakage comparable to that in present-generation pressurized LWR's (about 4%) would result in fuel utilization values of 13.0 and 15.0 Mwy/st U_3O_8 for soluble boron control and movable fuel control, respectively.

3. If the reactor were designed with a radial blanket of natural uranium, it might be possible to reduce leakage losses to about 1%. When this assumption is made, fuel utilization values of 15.1 and 18.2 Mwy/st U_3O_8 are obtained for soluble boron control and movable fuel control, respectively.

Hence, a maximum value of about 18.2 Mwy/st U_3O_8 could be obtained for the case of movable fuel control with a radial blanket assuming that leakage is reduced to 1.0%, and assuming that diffusion plant tails assay is 0.2% U-235.

II. RESULTS DERIVED FROM PREVIOUS KAPL CONVERTER-BURNER DATA

A. Background

During 1964 and 1965, KAPL performed detailed nuclear and thermal design evaluation calculations for several converter-burner concepts utilizing movable fuel reactivity control. The results of these calculations (Section II.B) are still valid, although minor modifications may be expected because of improved knowledge of fission products and other nuclear data. Each of the KAPL burners has some of the features addressed in this work. Two of these KAPL converter-burners were selected for extrapolation to an optimized condition. The extrapolation was done using the survey seed-blanket calculations described in Section IV to assess the effects of denatured vs highly enriched seed and the single-rod cell calculations of Section III to determine the difference between fuel management and batch control.

B. Review of KAPL Converter-Burner Concepts

The KAPL Large Seed Blanket Reactor (LSBR) effort was directed toward designing a long-lived advanced converter-burner, which would have an endurance of 70,000 effective full power hours (efph) at a thermal power rating of 1790 megawatts (Mw) and which would use a fuel element that would not exceed a fission density of 20×10^{20} fissions/cc.

Of those concepts evaluated for the LSBR, 11 KAPL converter-burner concepts have been chosen to provide a representation of the performance predictions obtained in this previous effort. The 11 burners span the following range of features:

1. A seed-blanket configuration.
2. The use of movable fuel for reactivity control.
3. The use of natural urania and natural thoria in the blanket.
4. Seed fuel elements designed to have high depletion (7×10^{20} to 20×10^{20} fissions/cc).
5. Thoria blankets designed to have high power fractions.
6. Several seeds for one blanket life (a simple fuel management scheme).
7. Thoria blankets containing fissile material.
8. Highly enriched seeds with no fertile material.
9. Highly enriched seeds with thorium as a fertile material.

A summary of the results of the 11 KAPL converter-burner concepts described above is presented in Table 2. Depending on the use of fissile fuel in the stationary thoria blanket, these 11 KAPL LSBR converter-burners fall into the following four categories:

1. Natural thoria (ThO_2) blanket.
2. Natural urania (UO_2) and natural thoria blanket.
3. ThO_2 blanket containing UO_2 highly enriched in U-235.
4. ThO_2 blanket containing UO_2 enriched to 2.7% in U-235.

The first category in Table 2 is comprised of converter-burner Models 10 and 20. The second category is comprised of converter-burner Models 12B, 12F, 22, 32, and 41G. The third category is comprised of converter-burner Models 19, 55D, and 59B. The fourth category is comprised of converter-burner Model 21.

The following comments can be made about the data in Table 2:

1. The main reason for studying these data is to observe how the value of E (i.e., Mwy/st of U_3O_8) varied in the seed-blanket converter-burners studied during 1964 and 1965. The following range of E values is shown in Table 2:

a.	Natural thoria blanket	11.6 to 12.4
b.	Natural urania and natural thoria blanket	11.2 to 12.7
c.	ThO_2 blanket with highly enriched UO_2	6.3 to 13.2
d.	ThO_2 blanket with 2.7% enriched UO_2	13.2

As can be seen from these data, the use of natural urania in the blanket would increase the value of E only slightly, while the use of enriched urania in the blanket would increase the value somewhat more to 13.2 Mwy/st U_3O_8.

2. The addition of a fissile material in the blanket would increase the beginning-of-life (BOL) blanket power fraction and would thereby reduce the seed peak-to-average power density at BOL.

The following range of blanket power fractions for BOL and end of life (EOL) are shown in Table 2:

		BOL	EOL
a.	Natural thoria blanket	0	0.45
b.	Natural urania and natural thoria blanket	0.10 to 0.35	0.42 to 0.46
c.	ThO_2 blanket with highly enriched UO_2	0.17 to 0.67	0.48 to 0.65
d.	ThO_2 blanket with 2.7% enriched UO_2	0.15	0.48

As can be seen from these data, the addition of a fissile material in the blanket would have an effect on reducing the seed BOL power fraction but little or no effect on the EOL blanket power fraction with the exception of the thoria blanket containing highly enriched UO_2.

3. The effect on conversion ratio due to the addition of a fissile material in the blanket is also shown in Table 2,in which the following range was observed:

a.	Natural thoria blanket	0.58 to 0.63
b.	Natural urania and natural thoria blanket	0.53 to 0.58
c.	ThO_2 blanket with highly enriched UO_2	0.58 to 0.73
d.	ThO_2 blanket with 2.7% enriched UO_2	0.54

As can be seen from these data, only the thoria blanket converter-burners containing higly enriched UO_2 would show a conversion ratio improvement over the natural thoria blanket converter-burners.

TABLE 2. REVIEW OF KAPL CONVERTER-BURNERS

Rated Power 1790 Mw(t)

Model	Seed Average, f/cc	efph	Mwd	Fissile Fuel, kg BOL	Fissile Fuel, kg EOL	β	β′	E	CR	Pf BOL	Pf EOL	Pf Avg
				All ThO_2 Blanket								
10	20×10^{20}	66,000	49.2×10^{6}	4588	2140	0.93	0.50	11.6	0.58	0	0.45	0.43
20	15×10^{20}	66,000	49.2×10^{6}	4259	2050	0.87	0.45	12.4	0.63	0	0.45	0.43
				UO_2 + ThO_2 Blanket								
12B	20×10^{20}	70,000	52.1×10^{6}	4720	2020	0.91	0.52	11.9	0.57	0.10	0.44	0.45
12F	15×10^{20}	55,000	41.0×10^{6}	3575	1460	0.87	0.51	12.4	0.58	0.13	0.46	0.45
22	15×10^{20}	68,000	50.5×10^{6}	4422	1750	0.88	0.53	12.3	0.56	0.15	0.42	0.44
32	17.4×10^{20}	76,000	56.5×10^{6}	4787	1780	0.85	0.53	12.7	0.56	0.35	0.44	0.45
41G	26.5×10^{20}	72,000	53.7×10^{6}	5169	2051	0.96	0.58	11.2	0.53	0.28	0.45	0.43
				ThO_2 Blanket with Highly Enriched UO_2								
19	15×10^{20}	74,000	55.1×10^{6}	4537	1720	0.82	0.51	13.2	0.58	0.17	0.48	0.47
55D	$<10 \times 10^{20}$	69,000	51.4×10^{6}	4906	3250	0.95	0.32	11.4	0.71	0.67	0.65	0.70
59B	7.2×10^{20}	27,000	20.2×10^{6}	3396	2765	1.70	0.31	6.34	0.73	0.45	0.48	0.47
				ThO_2 Blanket with 2.7% Enriched UO_2								
21	15×10^{20}	74,000	55.1×10^{6}	4532	1460	0.82	0.55	13.2	0.54	0.15	0.48	0.47

β BOL gm U-235/Mwd
β′ gm U-235 burned/Mwd
E Mwy/short ton U_3O_8
CR Conversion ratio = rate of creation of fissile fuel/rate of destruction of fissile fuel
Pf Blanket power fraction
f/cc Fissions per cc of fuel compartment

4. The fuel load per Mwd has very little spread in most cases. The value of β (i.e., grams of U-235 at BOL per Mwd) varies from 0.82 to 0.96 except in Case 10, in which ThO_2 seed was enriched with U-235.

5. All 11 converter-burners require fuel loads in the range of 1.9 to 2.9 kg U-235/Mw(t).

C. Extrapolation of KAPL Converter-Burner Data

Two KAPL converter-burner concepts were selected for extrapolation to include the effects of fuel management and denatured fuel. KAPL Converter-Burner 20 was used as a base case for a natural thoria blanket, and KAPL Converter-Burner 32 was used as a base case for a natural urania blanket. These converter-burners were chosen because they have the highest E values in their catagories. In both cases, the blankets were assumed to be depleted to an average of about 35,000 Mwd/mt.

The extrapolated data obtained from the single-rod cell and survey seed-blanket analyses are described in Table 10.* The comparison of these extrapolated converter-burner data with the corresponding pressurized LWR data is summarized in Table 3. The data in Table 3 indicate that the use of denatured fuel would reduce fuel utilization, whereas fuel management would greatly increase it. For blanket depletions averaging about 35,000 Mwd/mt, a fuel utilization of 16.1 Mwy/st U_3O_8 would be achieved in the case of a urania blanket, while a lower value of 15.0 Mwy/st U_3O_8 would be obtained with a thoria blanket. Increasing the average blanket depletion to 50,000 Mwd/mt would have no effect in the case of a urania blanket, but would improve E to 16.2 Mwy/st U_3O_8 for a thoria blanket. Based on these data, the maximum gain in E compared with a current pressurized LWR would be a factor of 1.38.

TABLE 3. EXTRAPOLATED CONVERTER-BURNER DATA

Condition	E, Mwy/st U_3O_8 UO_2 Blanket	E, Mwy/st U_3O_8 ThO_2 Blanket
Base case KAPL converter-burners	12.7	12.4
Base case corrected for denatured fuel	9.9	9.2
Base case corrected for denatured fuel and fuel management	16.1	15.0
Base case corrected for denatured fuel, fuel management, and 50,000 Mwd/mt	16.1	16.2

*P. 50.

These extrapolated LSBR results, although obtained by an indirect method, fall within the range of E values calculated by the methods of Sections III-V and hence provide further confidence in the conclusions of this report.

III. SINGLE-ROD CELL ANALYSIS

This section describes an evaluation of E (Mwy/st U_3O_8) using a single-rod cell analysis of a UO_2 and a ThO_2 fuel system. In these analyses, the single-rod cell assumes an infinite array of fuel rods in water. Each rod is assumed to consist of a cylindrical fuel region surrounded by an annular cladding region. The purposes of the evaluation are to compare:

1. UO_2 with ThO_2.
2. Soluble boron control with spectral shift and movable fuel control. (The latter control system simulates a seed-blanket core.)
3. The effect of increased fuel depletion over that of typical pressurized LWR designs.
4. Fuel management with batch loading.

When the effects of neutron leakage and other losses are included, the single-rod cell approach also provides data for E that include the effects of engineering constraints. The methods used to analyze the spectral shift control concept have not been qualified with experiment and may be less reliable than the methods used to analyze the other control concepts.

A. Model Description

The single-rod cell analysis for concepts which are not fuel-managed is based on a homogenized diffusion theory model of a Combustion Engineering (CE) System 80 reactor module. This CE module has an assembly pitch of 8.18 in. and has 236 fuel rods and 5 guide tubes for instrumentation and control. The computer model is a homogenized representation of this module and is equivalent to an infinite medium (i.e., uniform composition in axial and radial directions, zero buckling, and reflecting boundary conditions). The water-to-metal ratio of the unit cell is 1.43, which allows for the guide tube metal and associated guide tube water.

The model does not include any grid material or correction for axial and radial leakage. On a relative basis, however, the model is sufficient to obtain the comparisons intended.

The BOL model, for both UO_2 and ThO_2, is normalized based upon BOL RCP Monte Carlo calculations. The average linear heat generation rate for each fuel rod is set at 5.34 kw(t)/ft-rod, which is identical to the

CE System 80 value. The UO_2 fuel pellet is assumed to be comprised of a low enriched uranium oxide. The ThO_2 fuel pellet is assumed to be comprised of a homogenized solid-state solution of fully enriched uranium and thorium oxides. The analysis examines fissile fuel loads ranging from 2 to 5 wt %.

The control systems assumed are soluble boron, spectral shift, or movable fuel. The soluble boron is permitted to vary from timestep to timestep, with the study being terminated when the boron is no longer needed as the reactivity level becomes subcritical. The spectral shift control is assumed to consist of a variable mixture of light and heavy water. The H_2O content at BOL would be very low and would keep increasing over the fuel cycle until the mixture would be 100% H_2O as the reactivity level becomes subcritical. Although spectral shift control would cause no parasitic absorption, the D_2O in the moderator would harden the neutron energy spectrum and reduce the value of η-235 (fission neutrons per U-235 absorption) compared with light water. A movable fuel control system has the advantages of both no parasitic absorption and an optimized value of η-235. To obtain results for the movable fuel control scheme, the spectral shift calculations are modified by changing the value of η-235 from a mixed D_2O/H_2O system to a pure H_2O system. The resulting increase in η-235 leads to an increase in endurance for the same fuel load and hence to a gain in E. This approach to modeling the movable fuel control system based on spectral shift calculations is judged to be accurate to first order.

The single-rod cell analysis of fuel-managed conceptual designs is based on the same model described above for nonfuel-managed concepts except that the cell is divided into three regions. Each region has the identical load at BOL, with one third being discharged at the end of each cycle and with the remaining two regions being moved to accommodate a fresh third whose load is the same as the BOL value. The following schematic shows the location of the fuel management regions and their cycling pattern:

Cycle 1	1	2	3	
Cycle 2	2	3	1′	(Fresh fuel in 1′)
Cycle 3	3	1′	2′	(Fresh fuel in 2′)

This fuel management process is continued until a fuel cycle that is representative of an equilibrium fuel cycle is achieved. The BOL fuel load, or fissile weight percent chosen for each fuel management model, is a variable determined by the length of the fuel cycle and the requirement that the system use as little reactivity control as needed to maintain criticality.

TABLE 4. SINGLE-ROD CELL MODEL CHARACTERISTICS

Geometry	1 region, homogenized module, infinite medium
Computer code	NOVA/SPRITE, diffusion theory
Normalization	RCP model at BOL
Average linear heat generation rate, kw(t)/ft rod	5.34
Water/metal ratio	
Including guide tube water	1.43
Not including guide tube water	1.24
Fuel rod dimensions, in.	
Pitch	0.5063
Rod OD	0.382
Pellet OD	0.325
Cladding thickness	0.025
Gap thickness	0.0035
Fuel pellet composition	
Low enriched U	UO_2 (U-235, U-238)
Highly enriched U/Th	Homogenized mixture of ThO_2 and UO_2 (U-235)
Control systems	Spectral shift, soluble boron, simulated movable fuel
Region temperature, °F	550

B. Sensitivity of BOL Reactivity to Water-to-Metal Ratio

An analysis that determines the sensitivity of the BOL reactivity level of a single-rod cell to its water-to-metal ratio (W/M) has been performed. The purpose of this analysis was to ensure that the W/M used in the cell model and in the seed-blanket model (Section V) yields the highest BOL reactivity level. The assumption is made that maximizing the BOL reactivity level also maximizes the E value. Depletion effects could modify this optimization slightly, but have not been included in selecting W/M.

The study was based on a homogenized diffusion theory model similar to the model described in Section III.A. The only differences were the rod size, the pellet compositions, and the guide tube metal. The rod size for the sensitivity study was assumed to be 0.350 in. OD, compared with 0.382 in. OD for the homogenized model of the CE lattice. No guide tube metal was included in the analysis. The fuel pellets consisted of the following assumed isotopic mixtures:

1. UO_2 with 5.1 wt % U-235
2. ThO_2/UO_2 with 2 wt % U-235 and 8 wt % U-238
3. Natural urania (~0.7 wt % U-235)
4. Fully enriched UO_2/ZrO_2
5. Denatured UO_2 (20 wt % U-235, 80 wt % U-238)/ZrO_2.

Variation of the BOL reactivity level as a function of W/M values ranging from ~0.3 to 3.0 is shown in Figure 1 for the five pellet compositions. As can be seen from these data, the BOL k_∞ of the various conceptual designs would peak at different W/M values, with four of the five conceptual designs being very flat and close to the maximum reactivity value in the W/M range from 1.0 to 1.5. The UO_2 curve with a U-235 wt % of 5.1 peaks at a W/M of ~2.5. At a W/M of 1.4, the reactivity level is 3.3% lower than the maximum. At a W/M of 1.0, the reactivity level is 6.7% lower than the maximum. Therefore, with the exception of the UO_2 case, any analysis with a W/M of 1.0 to 1.5 can be assumed to maximize E from a W/M standpoint. For UO_2, a small increase in E may be realized if the W/M is increased.

C. Batch Loading — Soluble Boron and Spectral Shift Reactivity Control

The batch loading analysis consisted of evaluating and comparing the values of E for both UO_2 and ThO_2 for the various control systems and fissile fuel weight percents described in Section III.A. In each comparison, the values of E are plotted as a function of Mwd/mt.

The results of the ThO_2 spectral shift and soluble boron analyses are presented in Figure 2. U-235 weight percents of 3, 4, and 5 were examined, resulting in a range of average depletion from 18,000 to 52,000 Mwd/mt. As can be seen from Figure 2, E would be maximized by depleting each case beyond the point reached in a typical pressurized LWR. The soluble boron control curve would yield the lowest E values and would peak at ~43,000 Mwd/mt, with an E of 9.3 Mwy/st U_3O_8. The spectral shift control curve would peak at ~45,000 Mwd/mt with an E of 11.3 Mwy/st U_3O_8.

The results of the UO_2 spectral shift and soluble boron analyses are also presented in Figure 2. U-235 weight percents of 2, 3, 4, and 5 were

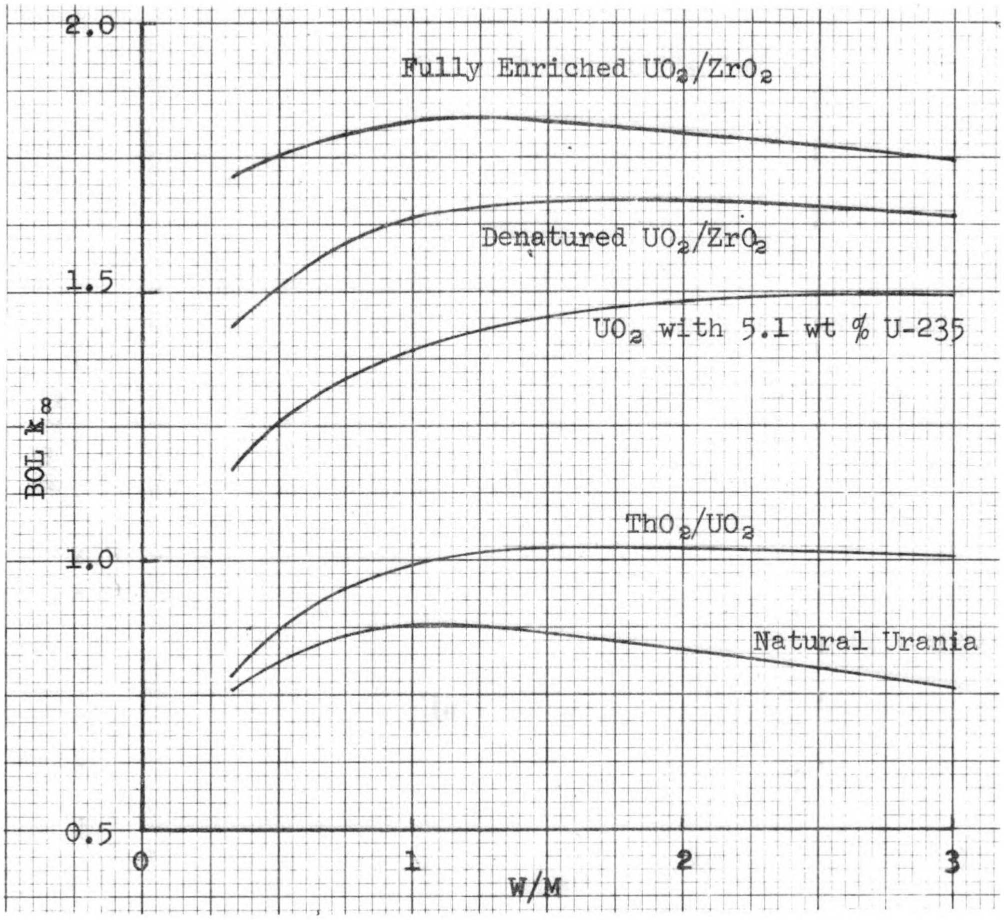

FIGURE 1. BOL Reactivity as a Function of Water-to-Metal Volume Ratio for Various Fuel Rod Types.

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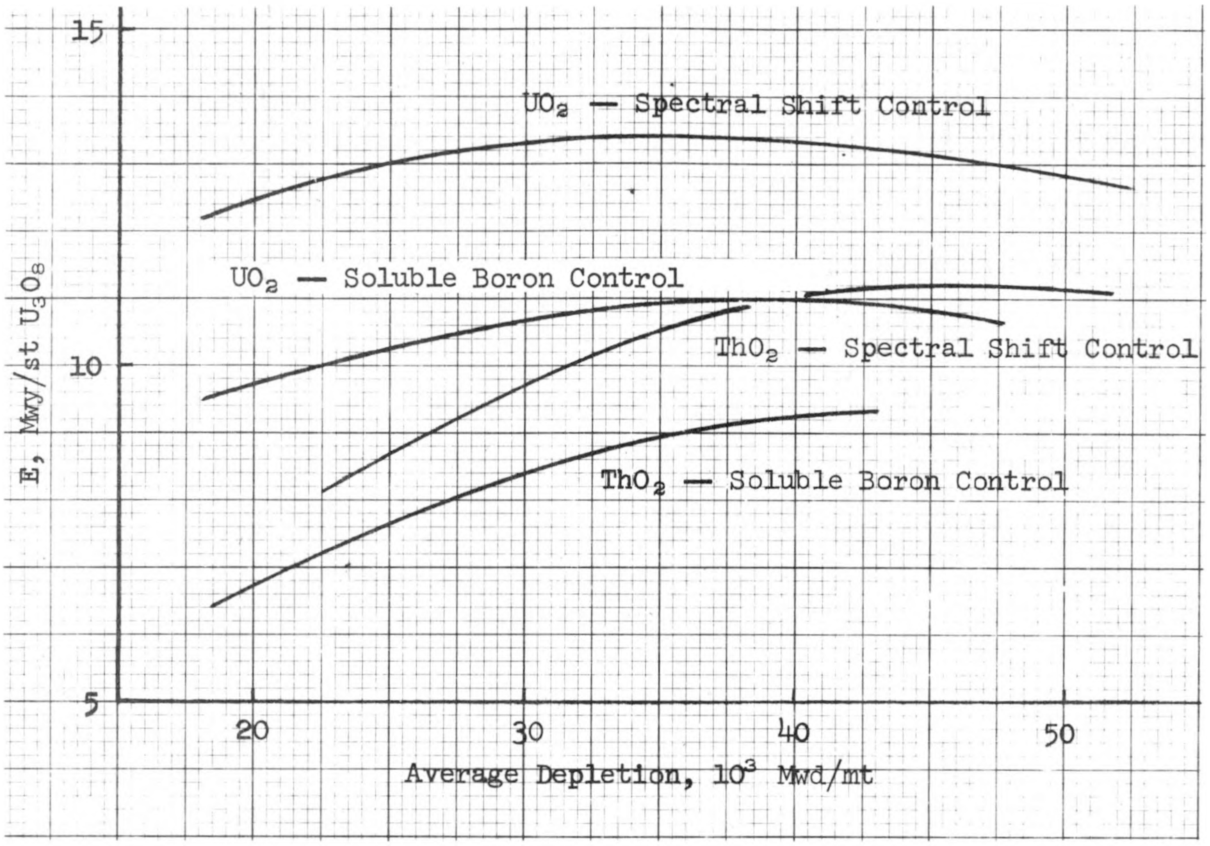

FIGURE 2. Batch Loading E Comparison for ThO_2 and UO_2 Single-Rod Cell Analysis.
KS-75978

examined, resulting in a range of average depletion from 18,000 to 52,000 Mwd/mt. Both control systems would result in E values that peak at ~35,000 Mwd/mt, with a maximum soluble boron value of 11.0 and a maximum spectral shift value of 13.4.

A comparison of these ThO_2 and UO_2 data shows that the UO_2 conceptual designs have consistently higher values of E than the corresponding ThO_2 conceptual designs.

A sensitivity of E to power density has been evaluated for the ThO_2 and UO_2 spectral shift conceptual designs at a U-235 weight percent of 4.0. A factor of 3 reduction in the average linear heat generation rate (5.34 kw/ft to 1.78 kw/ft) results in an increase in E of 9.1% for UO_2 and of 30.9% for ThO_2. Therefore, as the power density is decreased, a point at which ThO_2 yields higher E values than UO_2 is reached. It should be noted that an average linear heat generation rate of 1.78 kw/ft is too low to be practicable. It was considered in this study only for the purpose of obtaining sensitivity information.

The reason why the thorium cycle is more sensitive to power density than the uranium cycle is that the half-lives of Pa-233 and Np-239 are different. In the thorium chain,

$$\text{Th-232} \xrightarrow{(n,\gamma)} \text{Th-233} \xrightarrow[22.2m]{\beta^-} \text{Pa-233} \xrightarrow[27.0d]{\beta^-} \text{U-233} \ ,$$

while in the uranium chain,

$$\text{U-238} \xrightarrow{(n,\gamma)} \text{U-239} \xrightarrow[23.5m]{\beta^-} \text{Np-239} \xrightarrow[2.35d]{\beta^-} \text{Pu-239} \ .$$

Both the half-life and the resonance cross-section values of Pa-233 are much larger than the values for Np-239. Therefore, more neutrons are absorbed in Pa-233 than in Np-239. As a consequence, a reduction in power level increases the Pa-233 decay to U-233 more than it increases the Np-239 decay to Pu-239.

D. Fuel Management — Soluble Boron and Spectral Shift Reactivity Control

The fuel management analysis, as described in Section III.A, involves a scheme whereby the homogenized single-rod cell is divided into thirds and each third is fuel-managed until it is discharged at the end of three cycles. This results in an equilibrium cycle with three distinct fuel depletions and a reduction in the fissile fuel load compared with batch loading with the same fuel endurance. For each fuel system and control scheme, fuel cycle lengths of 6,000, 10,000, and 13,000 efph are examined. The BOL fissile fuel load of each case is iterated upon until the system just maintains criticality throughout the equilibrium fuel cycle.

The results of the ThO_2 fuel management analysis are presented in Figure 3. The three cycle lengths would require U-235 weight percents of 3.1, 4.4, and 5.6 for soluble boron control and of 2.9, 3.6, and 4.5 for spectral shift control. Both control systems would result in a range of average depletion from 28,000 to 62,000 Mwd/mt. E would be maximized by depleting each case beyond 50,000 Mwd/mt. At 50,000 Mwd/mt, the value of E would be equal to 14.3 Mwy/st U_3O_8 for spectral shift control and to 12.6 Mwy/st U_3O_8 for soluble boron control

The results of the UO_2 fuel management analysis are also presented in Figure 3. The three cycle lengths would require U-235 weight percents of 2.1, 3.2, and 4.2 for soluble boron control and of 1.9, 2.8, and 3.7 for spectral shift control. This would result in a range of average depletion from 26,000 to 57,000 Mwd/mt for both control systems and in maximum E values of 18.1 Mwy/st U_3O_8 for spectral shift and of 15.8 Mwy/st U_3O_8 for soluble boron. Both of these maximum values would occur at $\sim$45,000 Mwd/mt.

Figure 3 provides a comparison of the ThO_2 and UO_2 data. For both control systems, UO_2 would yield a higher value of E through 50,000 Mwd/mt.

The gain in E due to fuel management is shown in Figure 4. The factor by which E would increase is plotted as a function of Mwd/mt and ranges from 1.15 to 1.46. UO_2 would have consistently higher gains than ThO_2. In all cases, the gains would increase with depletion.

Although the effect of power density has not been calculated for fuel management, it is expected that the results are similar to the batch loading case, i.e., that the E value of ThO_2 would increase more than for UO_2 as the power density is decreased.

E. Simulated Movable Fuel Reactivity Control

As described in Section III.A, the movable fuel reactivity control scheme is inferred from the spectral shift calculations by making an η correction to E that accounts for the loss in reactivity due to the D_2O present in the moderator. This correction was made to the batch loading and fuel management calculations just described; the results are presented Figure 5. Movable fuel reactivity control would increase the maximum batch loading E values by 11.2% for UO_2 and by 8.0% for ThO_2 compared with spectral shift. For fuel management, movable fuel control would increase the maximum E values by 6.6% for UO_2 and by 5.6% for ThO_2 compared with spectral shift.

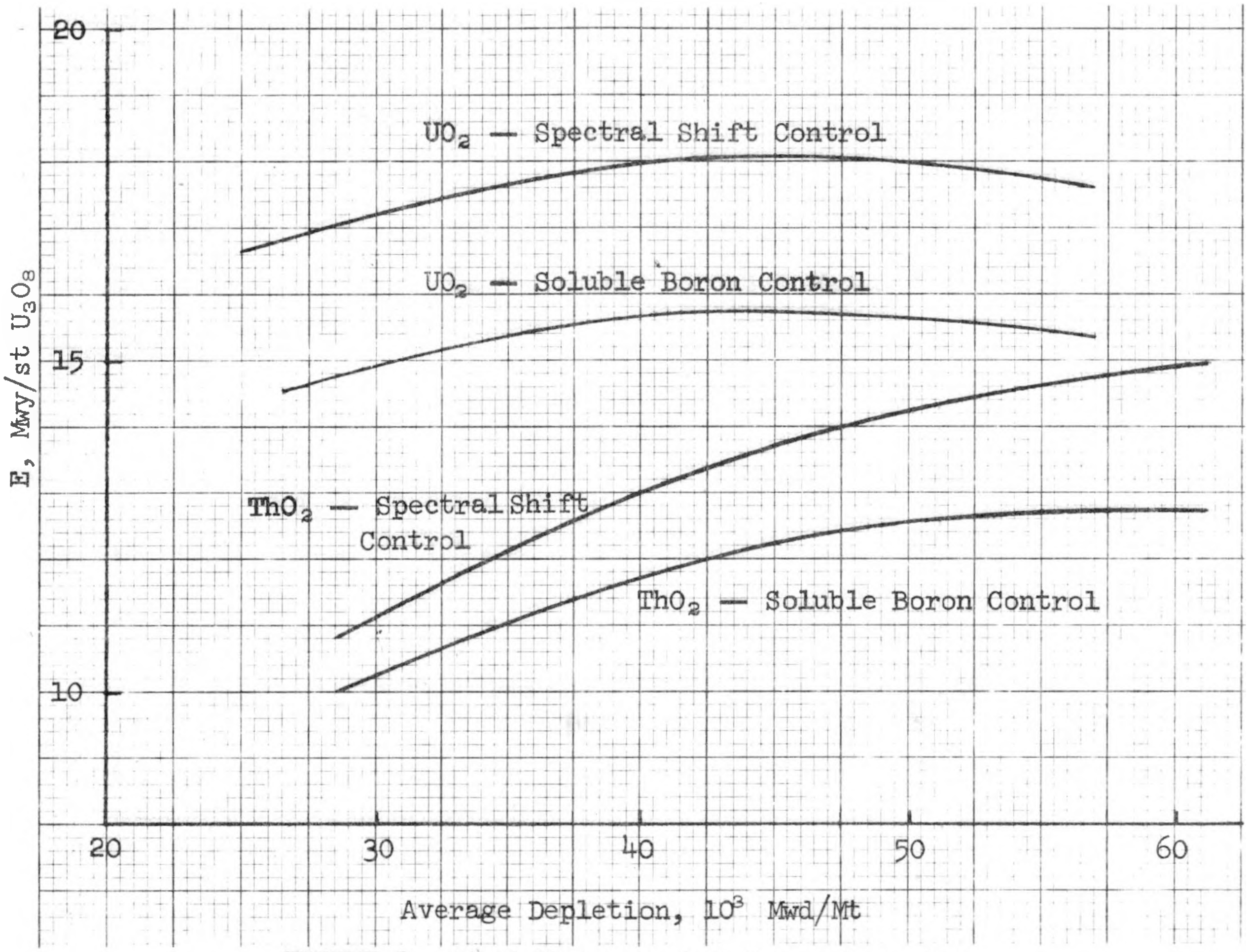

FIGURE 3. Fuel Management E Comparison for ThO_2 and UO_2 Single-Rod Cell Analysis.
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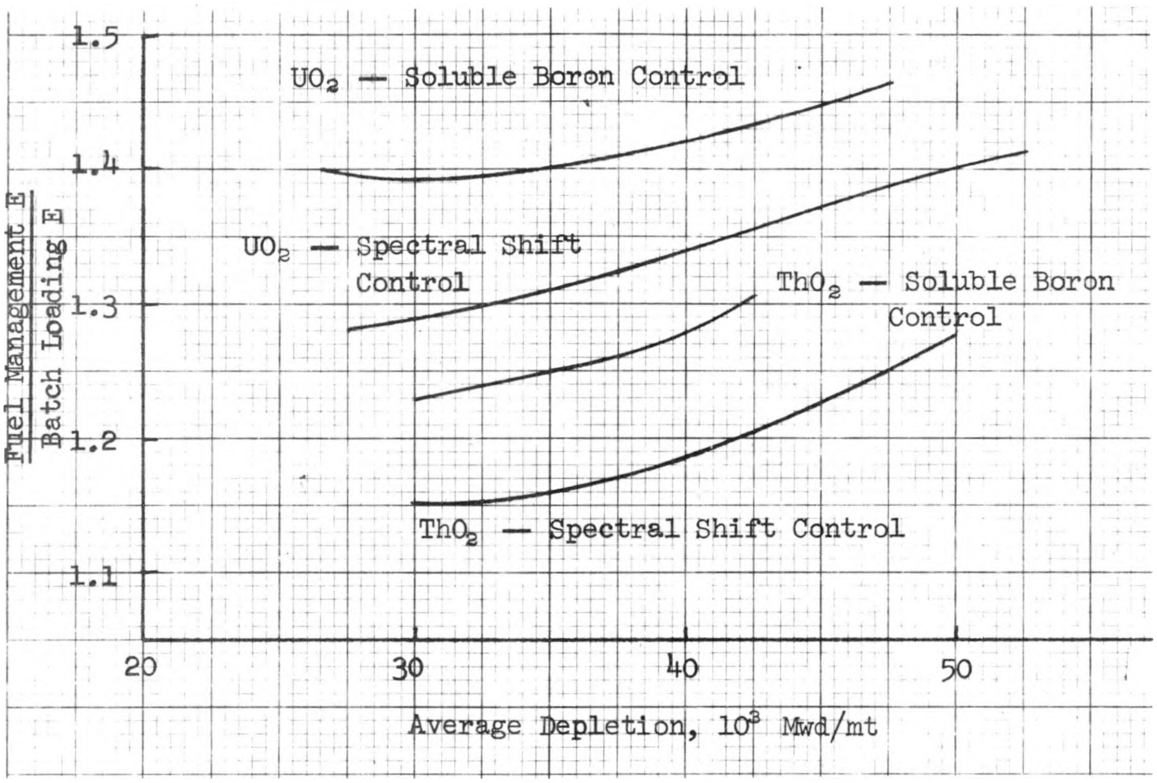

FIGURE 4. Comparison of Gains Resulting from Fuel Management for ThO_2 and UO_2 Single-Rod Cell Analysis.

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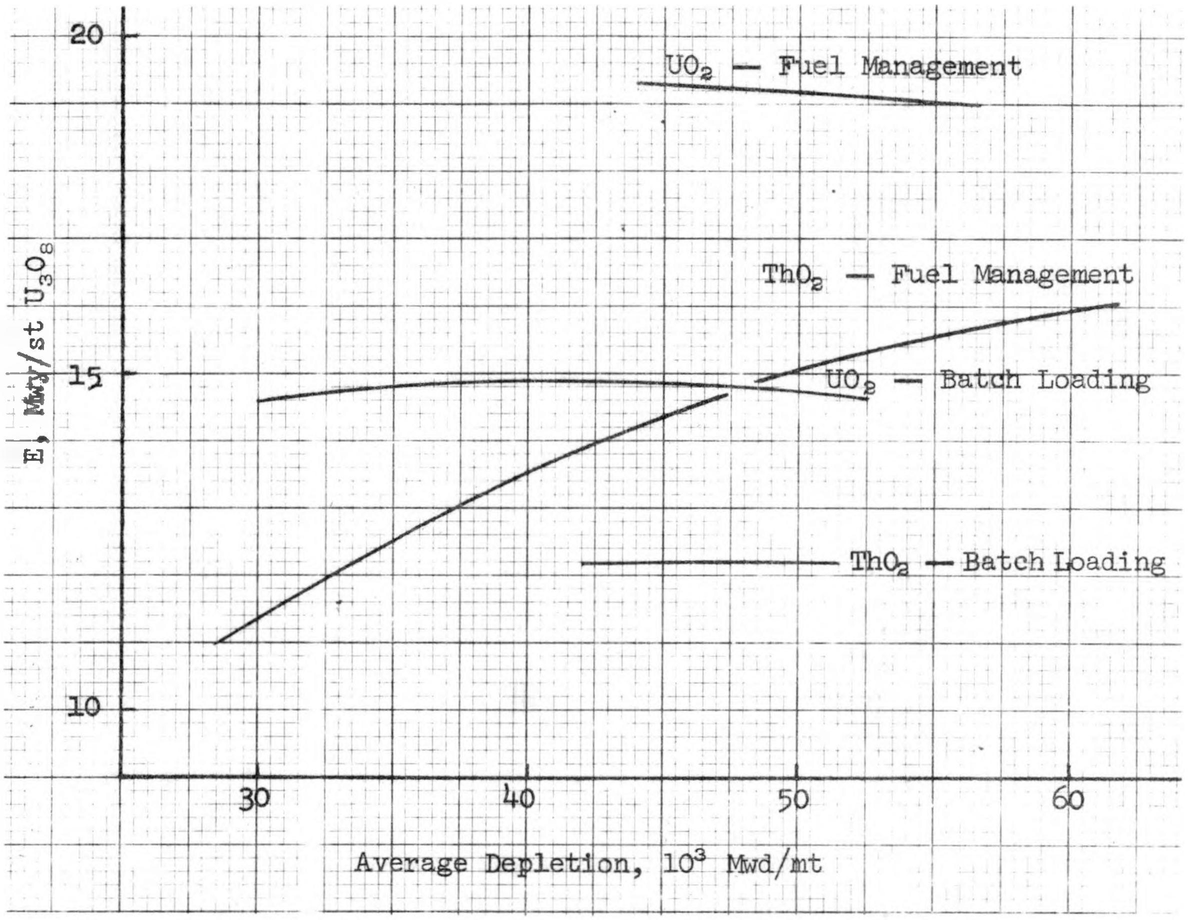


FIGURE 5. Simulated Movable Fuel E Comparison for ThO_2 and UO_2 Single-Rod Cell Analysis.
KS-75981

The gains in E due to movable fuel reactivity control with fuel management would be still higher if an idealized continuous fuel management scheme were employed. To evaluate the effect of increasing the frequency of refueling, the change in E due to the change in average depletion level has been determined from the batch and fuel management analyses. The average depletion level due to continuous refueling is one half of the average depletion due to batch loading. Using the E vs depletion correlation and the known change in average depletion level, the maximum E value obtainable with continuous fuel management and movable fuel reactivity control would be 21.5 Mwy/st U_3O_8 for the UO_2 conceptual design and would be 16.6 Mwy/st U_3O_8 for the ThO_2 conceptual design through 50,000 Mwd/mt. These results are an upper bound on E but do not represent a practical design because they assume zero neutron leakage, which is obviously not attainable.

F. Fuel Management with Leakage Losses

The results presented for both batch and fuel management analyses are based on infinite-medium calculations (i.e., uniform composition in the axial and radial directions, zero buckling, and reflecting boundary conditions) and do not account for neutron leakage that significantly reduces fuel utilization. An efficient present-generation PWR has about 4% leakage (Reference 1). To account for such leakage, infinite-medium fuel management calculations were performed with a neutron multiplication factor, k_∞, of 1.04 rather than 1.00, thus simulating a finite core with 4% leakage. In addition, the spectral shift calculations were further adjusted to account for the increased leakage effect due to the heavy water present in the moderator.

A comparison of the ThO_2 and UO_2 fuel management data adjusted for leakage is shown in Figure 6. For a UO_2 core, maximum E values of 13.0, 13.6, and 15.0 Mwy/st U_3O_8 would be obtained for soluble boron, spectral shift, and movable fuel reactivity control, respectively. For a ThO_2 core, maximum E values of 11.1, 11.7, and 12.8 Mwy/st U_3O_8 would be obtained for the same reactivity control systems, respectively. These results are judged to represent the maximum values obtainable in an optimized practical burner conceptual design without changing the design of the core reflector.

G. Fuel Management with Reduced Leakage Losses

The majority of the leakage in a present-generation PWR is in the radial direction. A burner that is designed to make optimum use of this leakage would result in an increased value of E. This could be done, for example, by using a radial blanket of fertile material driven by leakage neutrons to create and burn additional fissile material.

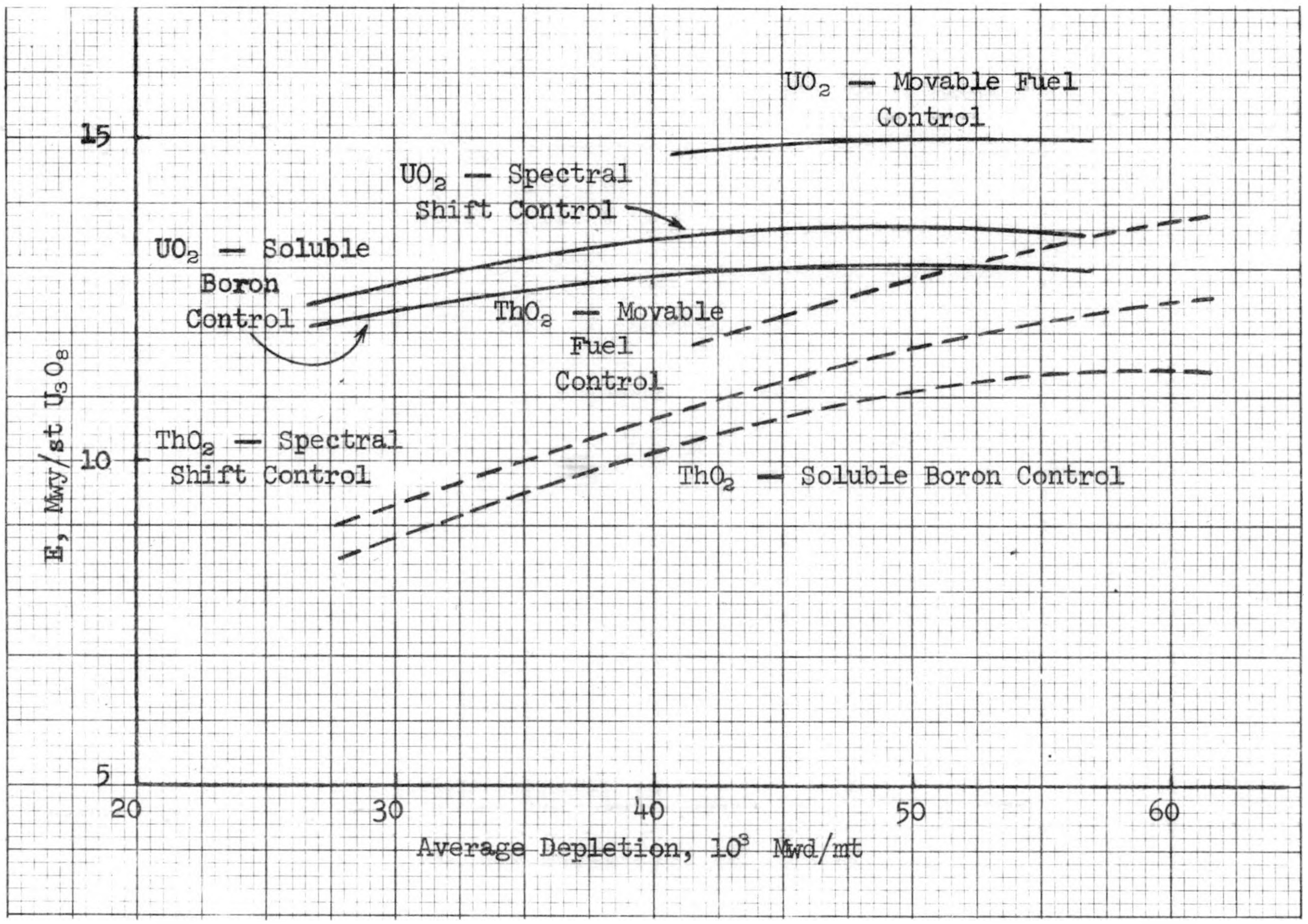


FIGURE 6. Fuel Management E Comparison for ThO_2 and UO_2
Single-Rod Cell Analysis Adjusted for Leakage
Typical of Present Generation PWRs.

KS-75982

An estimate of the additional increase in E that could be obtained in a practical system as a result of the reduction of leakage has been made based on the following assumptions:

1. Leakage and other neutron losses can be limited to about 1% by efficient design. One percent leakage is assumed to be a lower limit achievable in a pressurized LWR.

2. Neutrons absorbed in a radial blanket have the same efficiency for the production and utilization of fuel as in-core neutrons absorbed in the blanket material.

3. For a given reactivity control system, the percent change in E due to leakage is independent of the fuel cycle employed.

The effect of 1% leakage losses on the E value is determined by linearly interpolating between the infinite-medium calculations with no leakage correction and the corresponding calculations with a 4% leakage correction. A comparison of the ThO_2 and UO_2 fuel management data adjusted for about 1% leakage is shown in Figure 7. For a UO_2 core, maximum E values of 15.1, 17.0, and 18.2 Mwy/st U_3O_8 would be obtained for soluble boron, spectral shift, and movable fuel reactivity control, respectively. For a ThO_2 core, maximum E values of 12.2, 13.6, and 14.5 would be obtained for the same reactivity control systems maintaining average fuel depletion below 60,000 Mwd/mt. Increasing average depletion might increase the E values for the ThO_2 core, but these values have not been calculated. These results are judged to represent the maximum values obtainable in an optimized practical burner conceptual design.

When the infinite medium fuel management calculations for soluble boron reactivity control are corrected for about 4% leakage, E values of 9.4 Mwy/st U_3O_8 at 34,000 Mwd/mt and of 12.3 Mwy/st U_3O_8 at 30,400 Mwd/mt are obtained for ThO_2 and UO_2, respectively (Figure 6). These values compare with CE data (Reference 1) of 9.1 and 11.7 Mwy/st U_3O_8 for the corresponding fuel systems and depletion times. If the soluble boron curves in Figure 6 are normalized to the CE values at these depletion times, the effect of increased depletion would result in maximum E values of 10.7 Mwy/st U_3O_8 for ThO_2 and of 12.4 Mwy/st U_3O_8 for UO_2 for average fuel depletions not exceeding 50,000 Mwd/mt.

This comparison shows very good agreement between infinite-medium calculations, which have been corrected for leakage, and the more detailed CE System 80 results, thus qualifying the infinite-medium fuel management data.

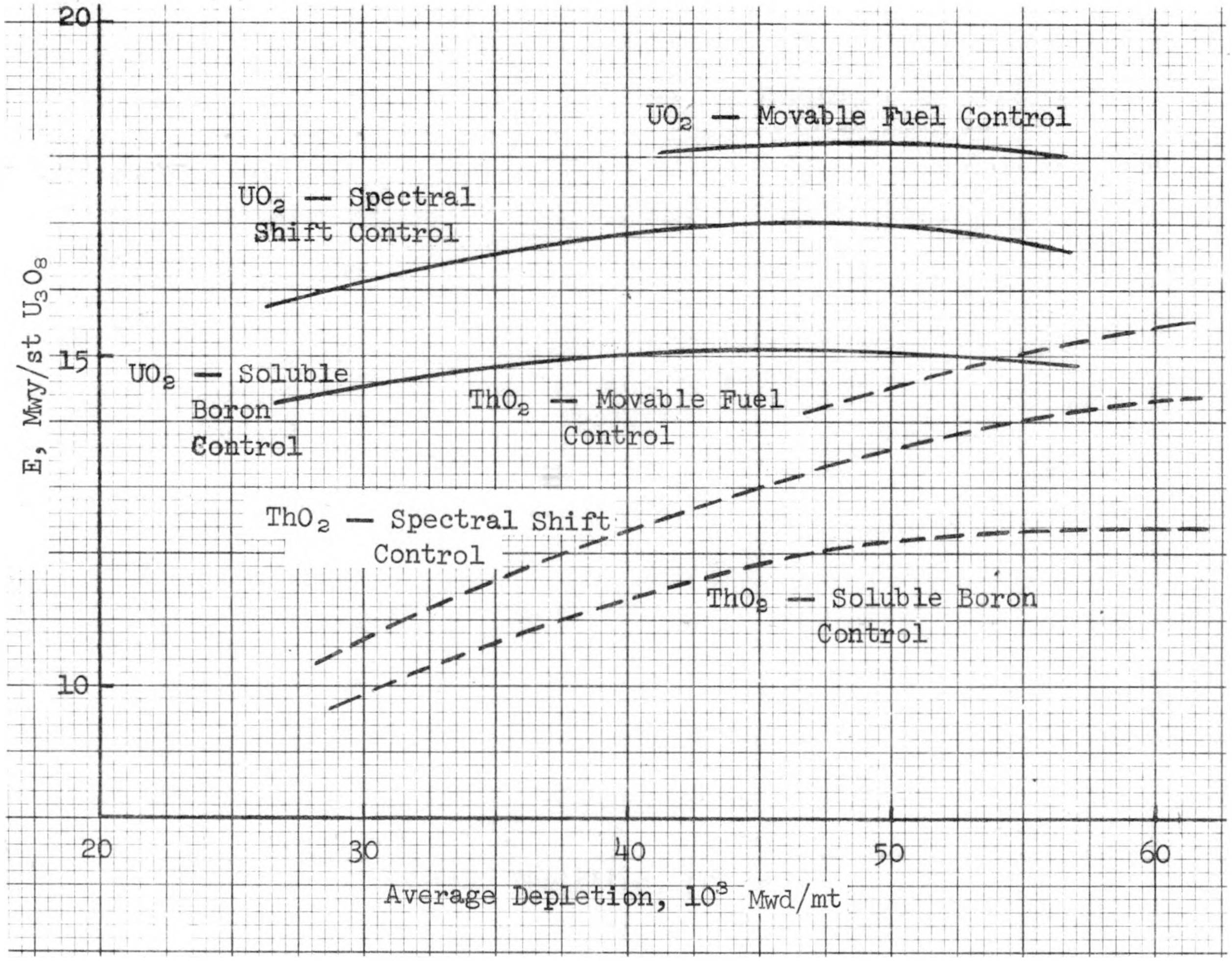


FIGURE 7. Fuel Management E Comparison for ThO_2 and UO_2 Single-Rod Cell Analysis Adjusted for Leakage with an Optimized Core Reflector.

KS-75983

I. Summary of Single-Rod Cell Analysis

The analysis results for the UO_2 and ThO_2 fuel systems can be summarized by the following:

1. For average fuel depletions not exceeding 50,000 Mwd/mt, UO_2 consistently yields a higher value of E than ThO_2 yields for the same control system at a power density consistent with pressurized LWRs.

2. Based on fuel management calculations, the simulated movable fuel control (simulated seed-blanket core) produces the highest E values with a maximum increase of 10.3% compared with spectral shift control. The E values for soluble boron control are always lower than for spectral shift.

3. A practical fuel management scheme (3 cycles) has the potential of increasing the value of E for both the ThO_2 and the UO_2 conceptual designs. For average fuel depletions not exceeding 50,000 Mwd/mt, the increase in the maximum E value is 44% for UO_2 and 35% for ThO_2. In the limit of continuous fuel management, an upper bound on the increase in E is 65% for UO_2 and 53% for ThO_2.

4. The gains in E for both batch loading and fuel management due to increased average depletion are small between 35,000 and 50,000 Mwd/mt for UO_2 and increase through 50,000 Mwd/mt for ThO_2.

5. When the fuel management E values are adjusted for leakage typical of present-generation PWRs (~4% leakage), the single-rod cell results are reduced by a maximum of 25% for UO_2 and 18% for ThO_2.

6. If optimum use of the leakage is made by the use of a fertile radial blanket, which would use leakage neutrons to create and burn additional fissile material, the leakage losses would reduce to about 1% and the single-rod cell results would be reduced at most by 6.1% for UO_2 and 4.9% for ThO_2.

7. Evaluation of sensitivity of E to power level has shown that a factor of 3 reduction in power level results in an increase in E of 9.1% for UO_2 and of 30.9% for ThO_2.

Table 5 lists the maximum value of E for each fuel system and each type of reactivity control for batch loading, fuel management based on infinite-medium calculations, fuel management corrected for neutron leakage typical of present-generation PWRs, fuel management with a 1% leakage correction, and fuel management normalized to the CE System 80 design. The results containing no correction for leakage should be interpreted on a relative basis because of the nature of the calculational model. The

TABLE 5. SINGLE-ROD CELL MAXIMUM E VALUE COMPARISON

(0 to 50,000 Mwd/mt Average Depletion)

	E, Mwy/st U_3O_8					
	Soluble Boron		Spectral Shift		Simulated Movable Fuel	
Fuel Cycle Strategy	UO_2	ThO_2	UO_2	ThO_2	UO_2	ThO_2
Batch						
5.34 kw(t)/ft-avg	11.0	9.3	13.4	11.3	14.9	12.2
1.78 kw(t)/ft-avg	-	-	14.6	14.8	-	-
Fuel management (5.34 kw(t)/ft-avg)						
Infinite medium	15.8	12.6	18.1	14.3	19.3	15.1
Adjusted for leakage typical of present-generation pressurized LWRs (~4% leakage)	13.0	11.1	13.6	11.7	15.0	12.8
Adjusted for leakage with an optimized core reflector (~1% leakage)	15.1	12.2	17.0	13.6	18.2	14.5
Normalized to detailed studies with leakage	12.4	10.7	-	-	-	-

fuel management values corrected for leakage typical of present-generation PWRs correspond to optimum values obtainable in a practical burner backfit design. The fuel management values with 1% leakage corrections are judged to be the maximum values obtainable in a practical system with redesigned reflectors. The fuel management values normalized to results for the CE System 80 design are used to qualify the infinite-medium fuel management calculations.

IV. SEED-BLANKET SURVEY STUDIES

Seed-blanket survey studies have been carried out to establish an upper limit on fuel utilization for once-through seed-blanket systems as a function of blanket average depletion. For the seed-blanket reactor concept considered, idealized movable fuel control systems and fuel management systems have been assumed to enable the reactor to meet the following criteria:

1. All neutrons leaving the seed are absorbed in the blanket.
2. The seed is depleted until the seed k_∞ drops to 1.0 before being discharged and replaced with a fresh reload seed.

Two types of blankets have been considered in the seed-blanket survey studies: a natural uranium oxide blanket and a natural thorium oxide blanket. Two types of seeds have been considered in the studies: a denatured uranium (20 wt % U-235, 80 wt % U-238) oxide fueled seed and a highly enriched uranium (assumed to be 100 wt % U-235) oxide fueled seed.

The water-to-metal ratios assumed are 1.43 in the seed and 1.24 in the blanket. These values were chosen to correspond to the W/M of the single-rod cell analysis described in Section III. The fissile fuel load is set at 1.51 kg/Mw(t) with a diluent of ZrO_2. The average seed linear heat generation rate is set at 5.34 kw(t)/ft. The average blanket linear heat generation rate is set at 4.8 kw(t)/ft, with the thoria blanket ramped from 0 to 4.8 kw(t)/ft at 7000 Mwd/mt.

The input data on which these survey calculations are based consist primarily of seed and blanket k_∞ values vs depletion. These data have been obtained from single-rod cell calculations using the methods of Section III. The resulting k_∞ values are shown in Figures 8 and 9. The calculation of E (Mwy/st U_3O_8) is carried out for seed-blanket systems in which seed and blanket modules are managed in such a way as to permit all seed modules to be driven until their k_∞ drops to 1.0 before they are discharged. The calculation based on the idealized fuel management scheme just described and on an idealized reactivity control scheme that eliminates neutron losses to control poisons consists of the following three steps:

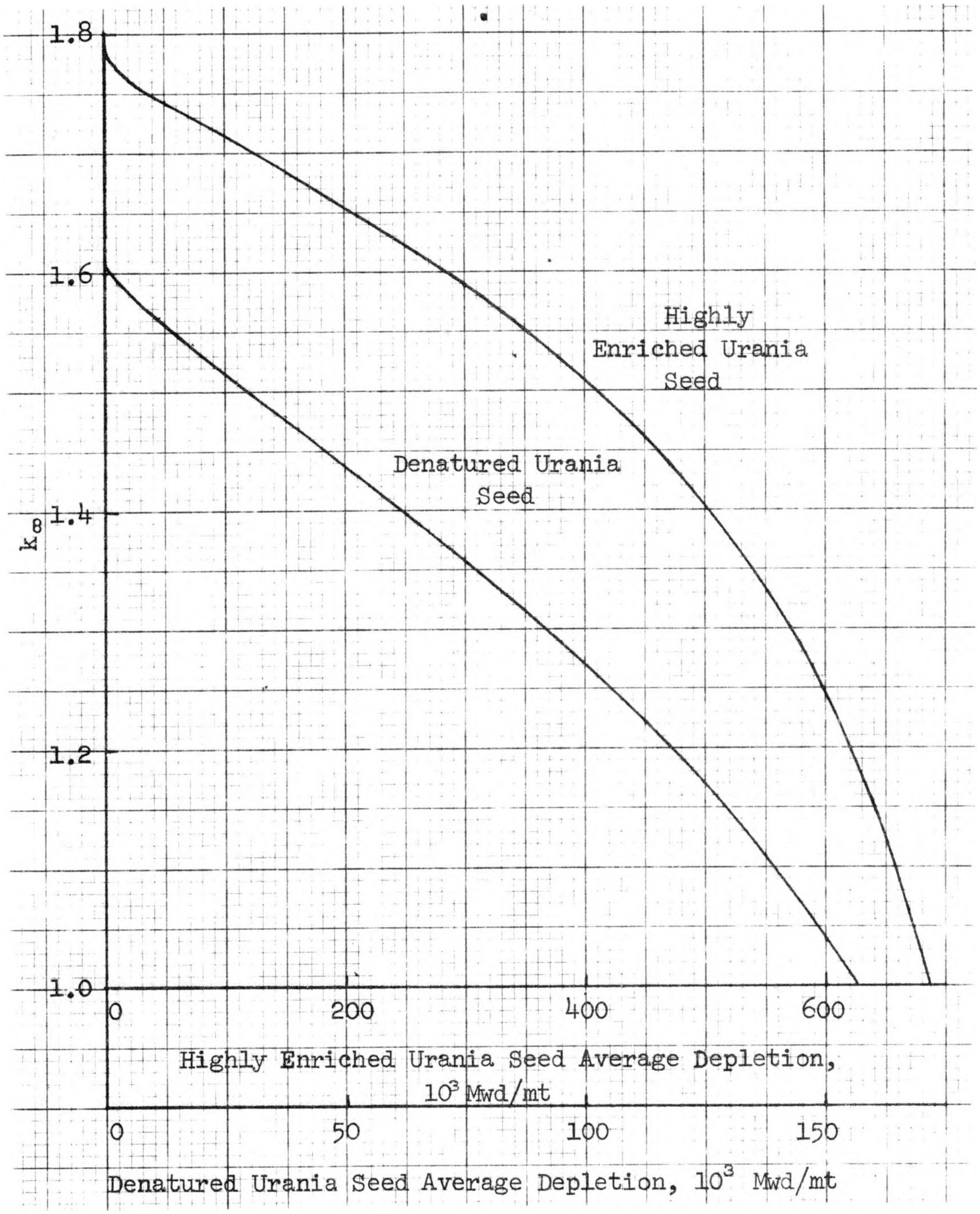


FIGURE 8. Seed Reactivity.
KS-75984

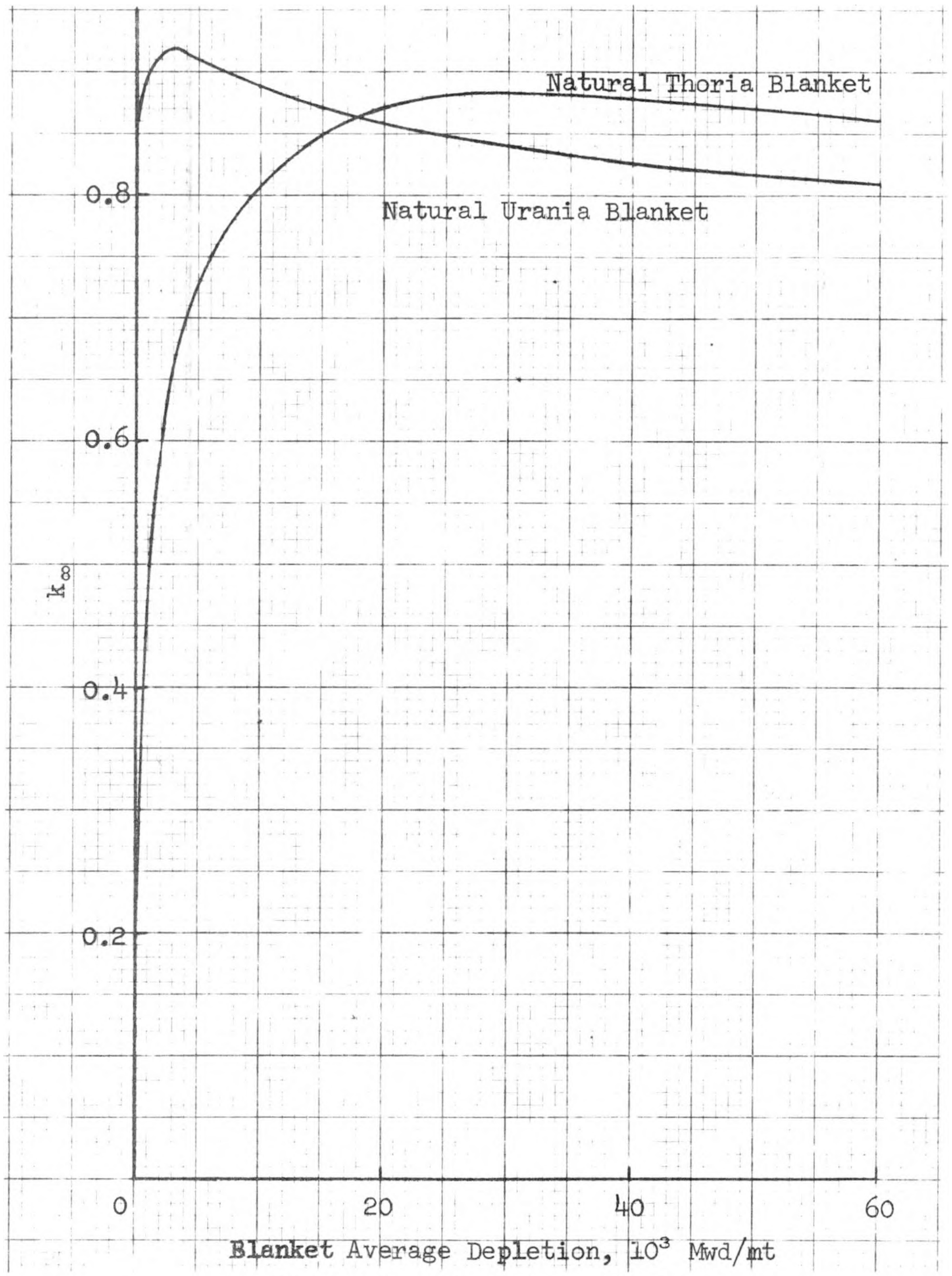


FIGURE 9. Blanket Reactivity.
KS-75985

1. Compute the cumulative energy produced and the net neutron leakage out of the seed until the k_∞ of the seed has dropped to 1.0. This calculation is based on a neutron balance for a critical system; the results are shown in Figure 10.

2. Compute the cumulative net neutron leakage into the blanket for the blanket average depletion (Mwd/mt) of interest. This calculation is based on the same neutron balance mentioned in Step 1. The results of this calculation are shown in Figure 11.

3. For the blanket average depletion of interest, determine the split in cumulative energy production between the seed and the blanket and determine the U_3O_8 mining requirement, with the assumption that the blanket remains in-core and that fresh reload seeds are supplied as needed to drive the blanket to the selected depletion. This calculation uses the neutron leakage data for the seed and blanket calculated in Steps 1 and 2 and uses the assumption that the number of neutrons leaving the seed equals the number of neutrons entering the blanket. The E values for the seed-blanket system follow directly from the seed-blanket split in energy production and the U_3O_8 mining requirement established here.

To qualify the seed-blanket survey analysis, a comparison of the results of that method has been made with the seed-blanket cell calculations described in Section V.B. The input k_∞ data for the survey method qualification calculations were obtained from two region seed-blanket unit cell studies having the same water/metal ratio and rod diameters as described in Section V.A. The survey method was used to compute the E values for a batch-loaded seed-blanket core at the blanket depletions of the seed-blanket cell analysis.

Table 6 presents the results of the comparison and shows that the maximum difference between the survey E value and the cell E value for the four seed-blanket cases considered is 5%. This close agreement provides support for the qualification of the survey analysis method.

The results of scoping calculations performed with the survey method for E vs blanket average depletion are given in Figures 12 through 14. The following observations are made:

1. Figure 12 shows results for denatured seed when the seed is depleted to a k_∞ of 1.0, thus simulating the case of fuel management. The maximum E for blanket average depletions <50,000 Mwd/mt that can be obtained with a denatured seed in this case is 17.3 Mwy/st U_3O_8 at 33,000 Mwd/mt for a natural urania blanket and is 16.4 Mwy/st U_3O_8 at 50,000 Mwd/mt for a natural thoria blanket.

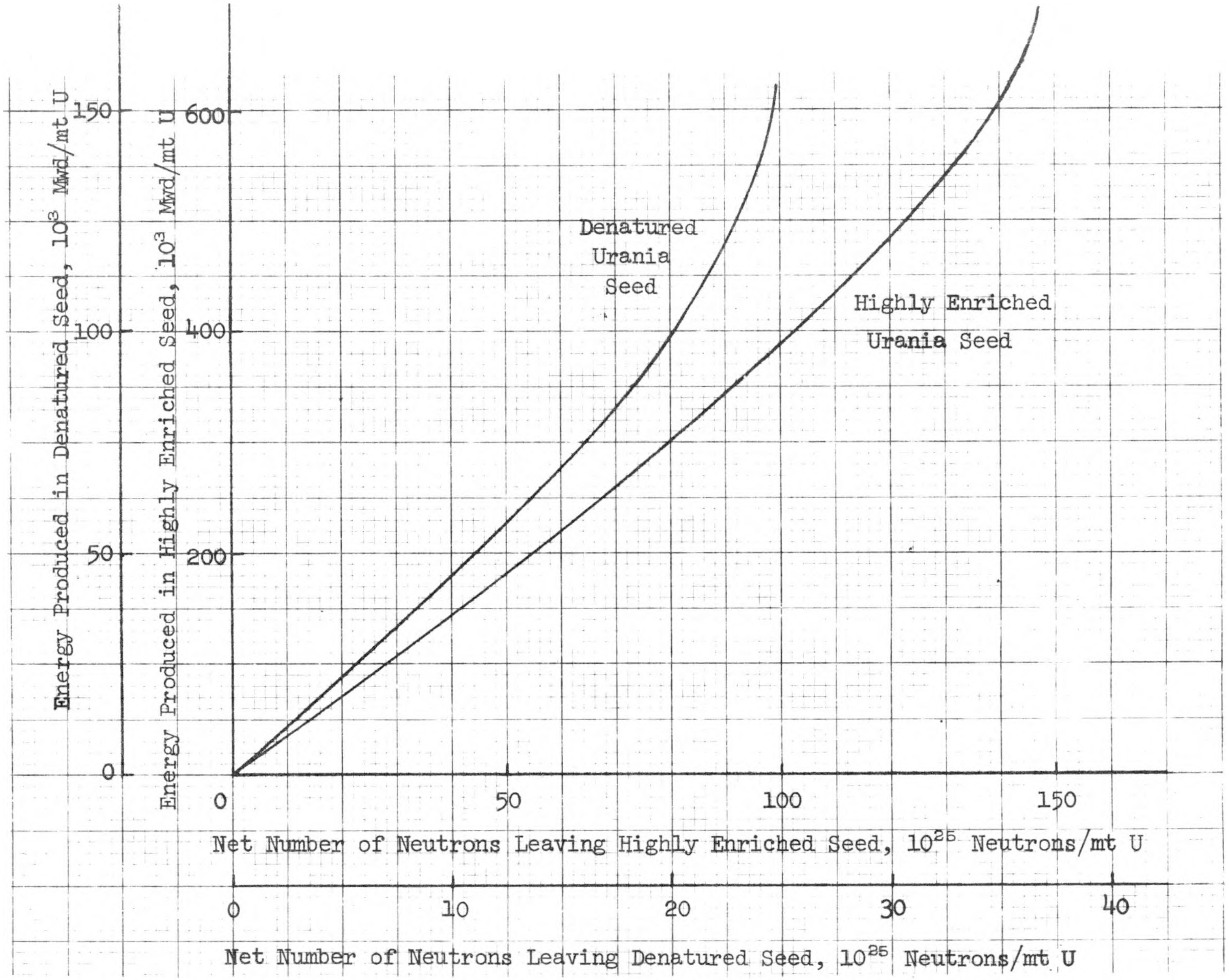


FIGURE 10. Energy Production of Denatured Urania Seed and Highly Enriched Urania Seed.

KS-75986

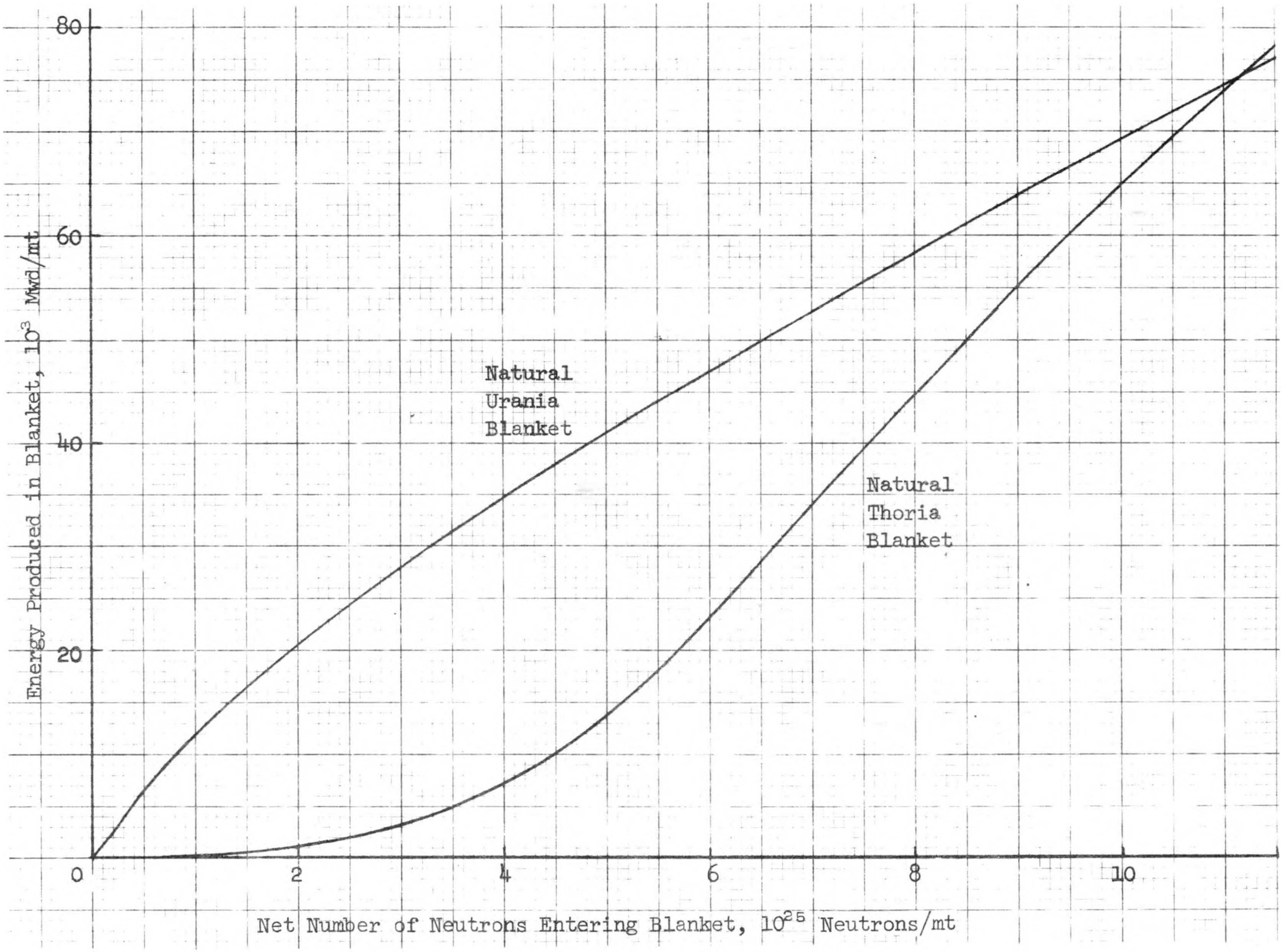


FIGURE 11. Energy Production of Natural Urania Blanket and Natural Thoria Blanket.

KS-75987

TABLE 6. COMPARISON OF SEED-BLANKET CELL ANALYSIS WITH SEED-BLANKET SURVEY ANALYSIS

	Blanket Average Depletion, Mwd/mt	E, Mwy/st U_3O_8 Cell Analysis	Survey Analysis	Survey/Cell
Highly enriched seed				
Natural thoria blanket	8,950	10.1	10.3	1.02
Natural urania blanket	8,650	9.8	9.7	0.99
Denatured seed*				
Natural thoria blanket	14,114	10.8	11.3	1.05
Natural urania blanket	7,585	8.6	9.0	1.05

*Denatured seed is comprised of 20 wt % U-235 and 80 wt % U-238.

2. Figure 12 also shows results for a denatured seed when the seed is depleted to a k_∞ of 1.3. This calculation was chosen to simulate the case of no fuel management. Comparison of the data in Figure 12 indicates that for this case fuel management would increase E by about 5 Mwy/st U_3O_8 for a denatured seed.

3. Figure 13 is the same as Figure 12 except for the case of a highly enriched seed. It is seen that fuel management would increase E by about 1.7 Mwy/st U_3O_8 for a highly enriched seed.

4. Figure 14 is a plot of selected data from Figures 12 and 13. Figure 14 illustrates the comparison between highly enriched seeds and denatured seeds with a natural urania blanket and a natural thoria blanket. In the case of a natural urania blanket, the use of a highly enriched seed instead of a denatured seed would increase E by <0.5 Mwy/st U_3O_8. In the case of a natural thoria blanket, the use of a highly enriched seed would cause differences in E by <0.3 Mwy/st U_3O_8 at blanket average depletions <50,000 Mwd/mt.

5. Figures 12 through 14 indicate that the uranium resource utilization potential for natural thoria blankets would be greater than for natural urania blankets for blanket average depletions beyond 60,000 Mwd/mt. These data are reasonably consistent with the results of the single rod cell analysis; bearing in mind that the single rod cell data are based on core average depletion, while the survey seed-blanket data are based on blanket average depletion.

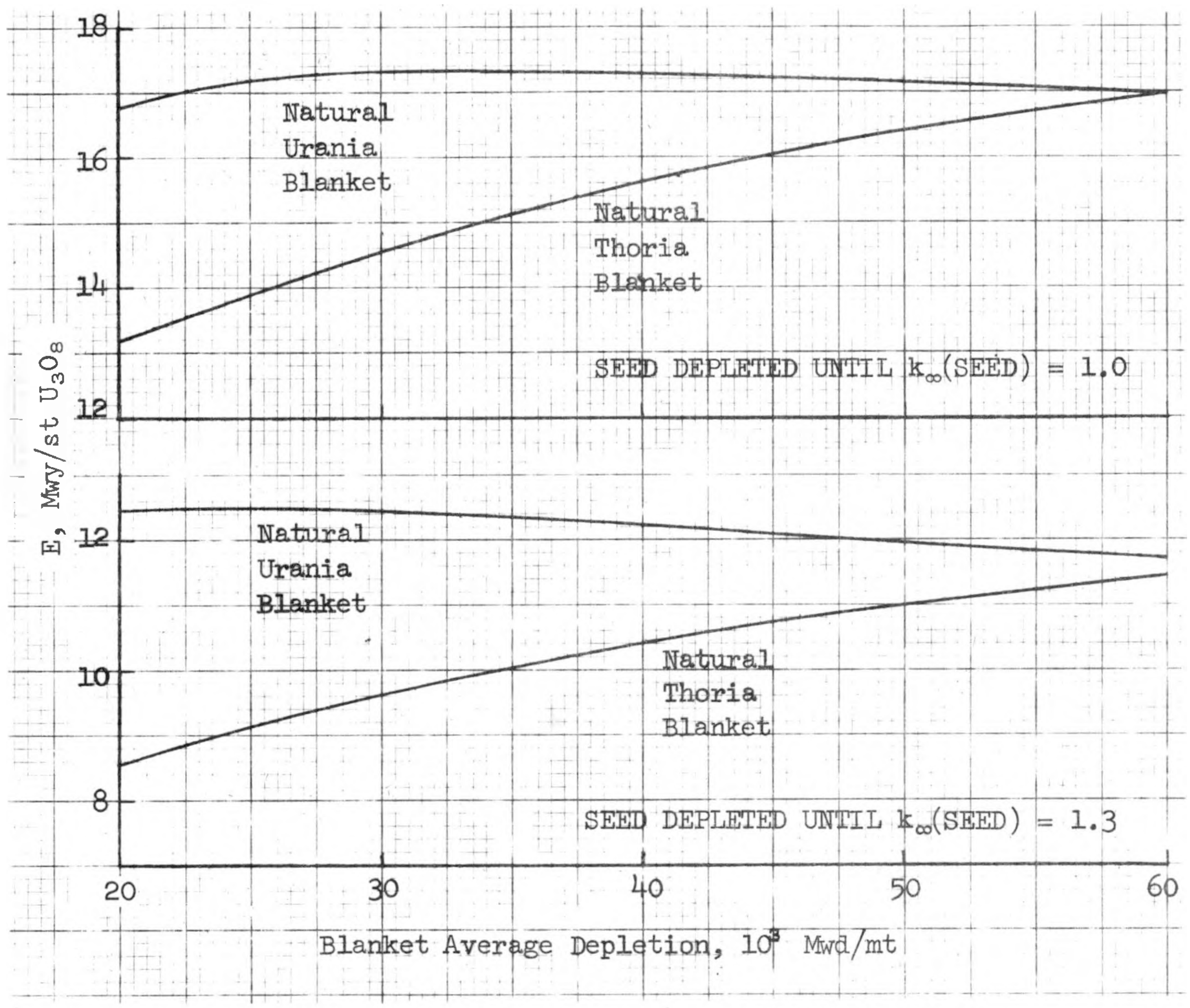


FIGURE 12. Fuel Management Seed-Blanket Survey Calculations for Denatured Urania Seed.
KS-75988

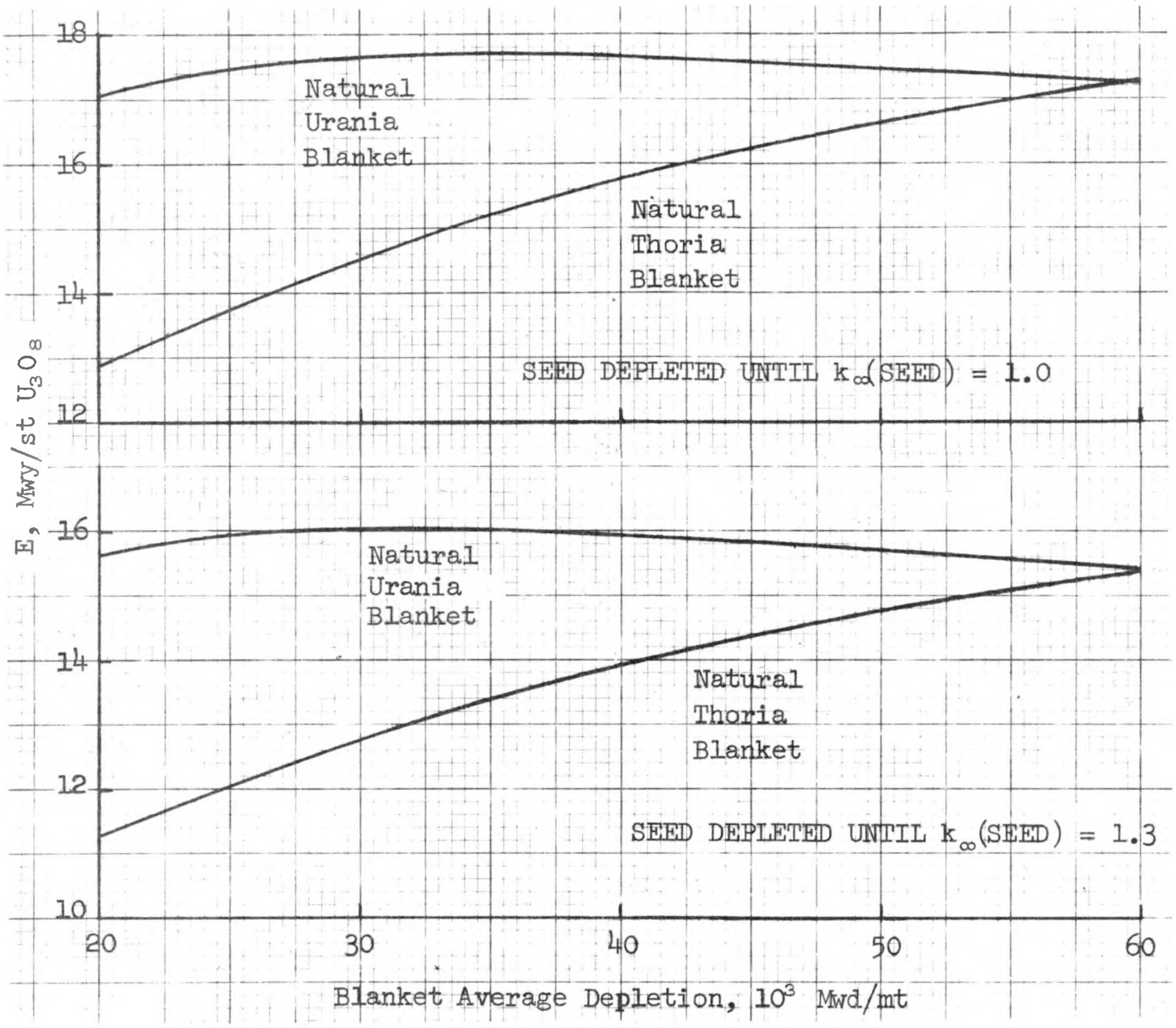


FIGURE 13. Fuel Management Seed-Blanket Survey Calculations for Highly Enriched Urania Seed.
KS-75989

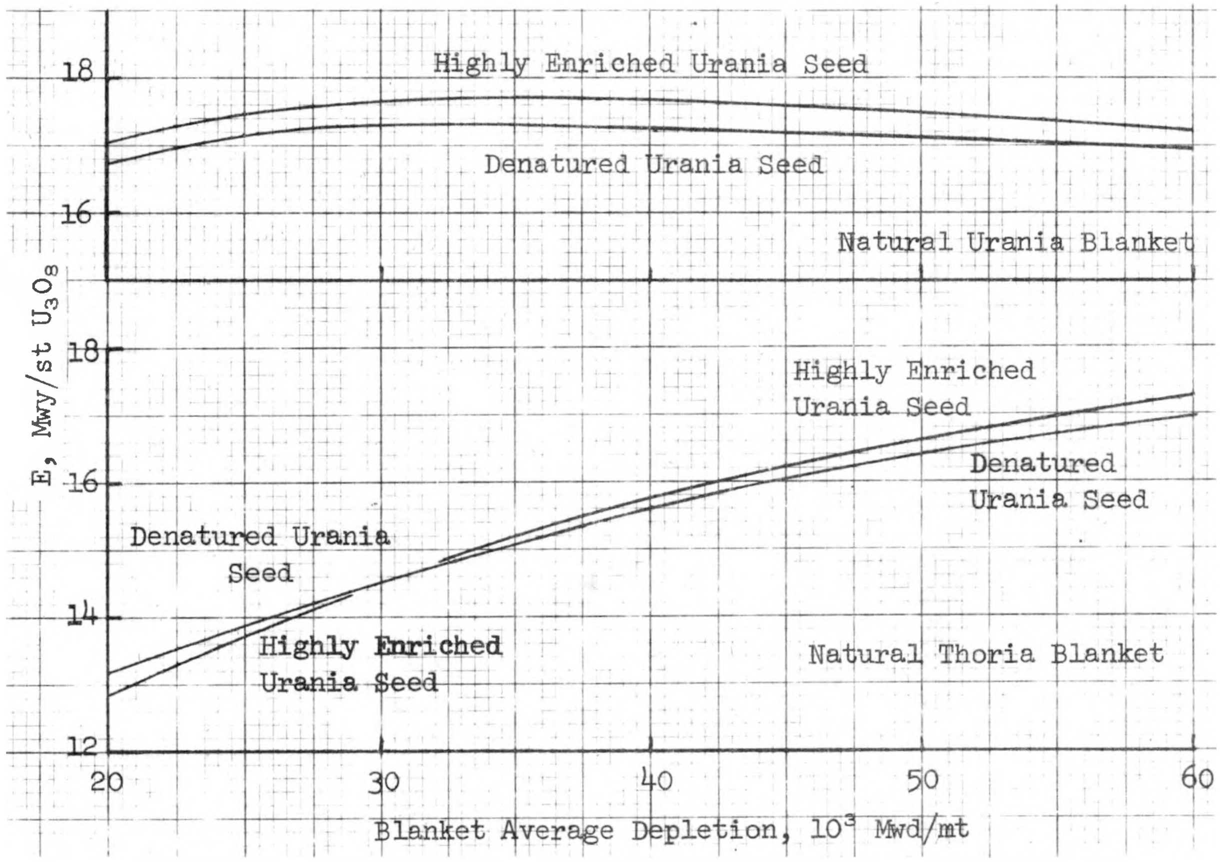


FIGURE 14. Fuel Management Seed-Blanket Survey Calculations for Seed Depleted until k_∞(Seed) = 1.0.

KS-75990

Figure 15 shows the results of a study to determine the effect on E of seed k_∞ at discharge. The data indicate that as the seed discharge k_∞ decreases below unity, the value of E approaches an upper limit of <18 Mwy/st U_3O_8 for seed k_∞ values in the range of 0.8 to 0.9. If seeds are driven to lower values of k_∞, fuel utilization actually reduces. Seed discharge k_∞ values <1.0 can be obtained by placing depleted seed modules in the blanket.

The above results show that resource utilization can be improved by fuel management. However, it is difficult to achieve additional resource utilization gain solely from improved fuel management practices because present-generation pressurized light water reactors have quite efficient fuel management procedures. The gains due to highly enriched seed vs denatured seed and due to increased blanket depletions are also comparatively small.

V. SEED-BLANKET UNIT CELL RESULTS

This section describes an evaluation of fuel utilization using an explicit seed-blanket cell analysis of highly enriched seeds and denatured seeds with natural thoria and natural urania blankets. The main purposes of the evaluation are as follows:

1. To provide additional comparisons of a highly enriched seed with a denatured seed.
2. To provide a check for the survey seed-blanket analysis.
3. To determine the sensitivity of blanket power fraction to seed-blanket volume variation.

A. Model Description

The cell analysis is based on a homogenized diffusion theory model of a single module containing a seed region and a blanket region. The computer model uses R-Z cylindrical geometry with unit height in the Z direction and represents an infinite array. The W/M ratio is 1.0 in both seed and blanket regions with a rod outer diameter of 0.350 in. The average linear heat generation rate for each seed fuel rod is assumed to be 5.34 kw(t)/ft rod. The model is normalized based on BOL unit cell RCP Monte Carlo calculations. The following assumptions are made for the highly enriched seed calculations:

1. It is comprised of UO_2/ZrO_2 fuel pellets with 4.3 vol % UO_2.
2. The UO_2 is highly enriched U-235.

For the denatured calculations, it is assumed that:

1. The denatured seed is comprised of UO_2/ZrO_2 fuel pellets with 21 vol % UO_2.
2. The UO_2 is comprised of 20% U-235 and 80% U-238.
3. BOL U-235 density is identical for the denatured and highly enriched systems.

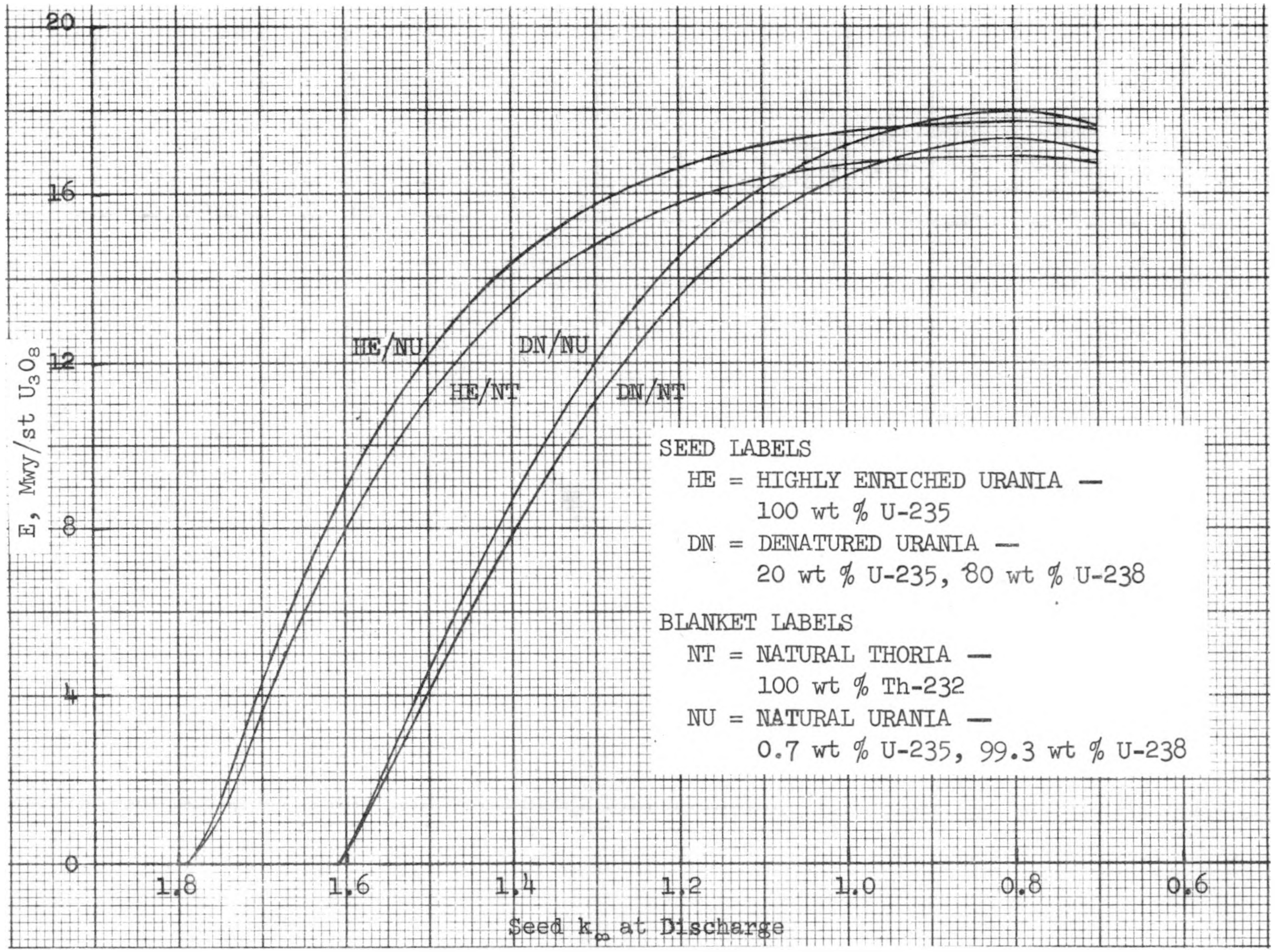


FIGURE 15. Fuel Management Seed-Blanket Survey Calculations for E vs Seed Discharge Reactivity at 50,000 Mwd/mt Blanket Average Depletion.
KS-75991

For the conceptual designs using a natural thoria blanket, the radial thickness of the seed region is assumed to be 6 in., which results in a cross-sectional area of 113.1 in.^{2} and a total of ~590 seed rods per module. The radial thickness of the blanket is varied until the BOL unrodded no-xenon reactivity level is ~1.10. For the highly enriched seed, this procedure results in a radial blanket thickness of 3.21 in., a cross-sectional area of 153.4 in.^{2}, and a total of ~800 blanket rods per module. For the denatured seed, this procedure results in a radial blanket thickness of 2.3 in., a cross-sectional area of 103.3 in.^{2}, and a total of ~540 blanket rods per module.

For the highly enriched seed with a natural urania blanket, the radial thicknesses are assumed to be 3.5 in. for the seed and 10.2 in. for the blanket. This volume combination results in a BOL unrodded no-xenon reactivity level of ~1.10, cross-sectional areas of 38.5 in.^{2} for the seed and 535.8 in.^{2} for the blanket, and a total of ~200 seed rods and ~2790 blanket rods per module.

For the denatured seed with a natural urania blanket, the radial thicknesses assumed are 5.0 in. for the seed and 6.52 in. for the blanket. This volume combination results in a BOL unrodded no-xenon reactivity level of ~1.10, cross-sectional areas of 78.5 in.^{2} for the seed and 338.5 in.^{2} for the blanket, and a total of ~410 seed rods and ~1760 blanket rods per module.

The control systems examined consist of soluble boron and movable fuel control. The soluble boron is used in the seed region only and is permitted to vary from timestep to timestep, with the study being terminated when the boron is no longer needed as the reactivity level becomes subcritical. This can be thought of as representing a core in which poison control rods and burnable poison are used in the seed only. The movable fuel consists of either a thoria or a urania control rod that is not explicitly represented in the model. The actual reactivity of the problem is always supercritical except at EOL. The conversion ratio, fissile fuel load, and core life are externally adjusted to describe a system that is critical.

For the seed-blanket analysis, batch loading was the only type of fuel cycle strategy analyzed. For this type of fuel cycle, no fuel management is assumed, and the model is depleted until the system can no longer maintain criticality.

A summary of the model characteristics for the seed-blanket analyses is shown in Table 7. A model schematic is presented in Figure 16.

TABLE 7. SEED-BLANKET UNIT CELL MODEL CHARACTERISTICS

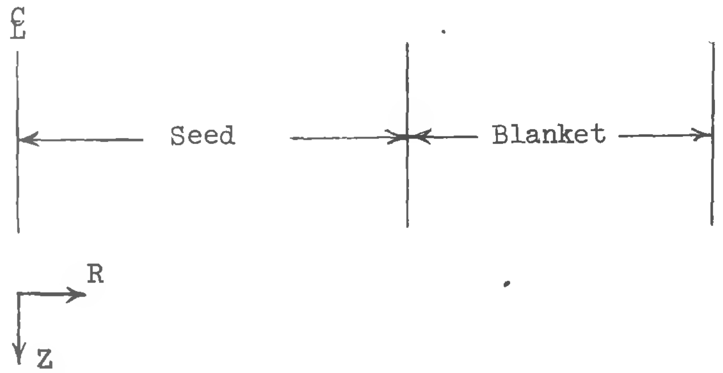
Geometry	Single module, R-Z cylindrical geometry, unit height, infinite array
Computer code	Diffusion theory
Normalization	RCP model at BOL
Average seed linear heat generation rate, kw(t)/ft rod	5.34
Water/metal ratio	1.0
Fuel rod dimensions, in.	
Pitch	0.471
Rod OD	0.350
Pellet OD	0.293
Cladding thickness	0.025
Gap thickness	0.0035
Fuel pellet composition	
Seed	
Highly enriched U	UO_2/ZrO_2 (U-235)
Denatured U	UO_2/ZrO_2 (20% U-235, 80% U-238)
Blanket	ThO_2, UO_2 (natural)
Control systems	Movable fuel, soluble boron
Region temperature, °F	550

B. Highly Enriched and Denatured Seed Comparison

This section describes a comparison of a denatured seed with a highly enriched seed using the four seed blanket models described in Section V.A. The highly enriched and denatured cases with a natural thoria blanket have been analyzed with both soluble boron and movable fuel reactivity control. The highly enriched and denatured cases with a natural urania blanket have been analyzed with only movable fuel reactivity control.

The results of a one-seed-life analysis are presented in Table 8 for the various seeds, blankets, and control systems described above. These data show that with a natural thoria blanket, denatured fuel would maximize core life and the value of E; the difference between the denatured and highly enriched systems is 5 to 7% in E, depending on the control system employed. However, with a natural urania blanket, the highly enriched fuel would maximize core life and increase the E value by ~14% compared with denatured fuel.

Unit Height, R-Z Cylindrical Geometry



	Highly Enriched Seed		Denatured Seed	
	ThO_2 Blanket	UO_2 Blanket	ThO_2 Blanket	UO_2 Blanket
Radial Thickness, in.				
Seed	6.0	3.5	6.0	5.0
Blanket	3.2	10.0	2.3	6.5
Cross-Sectional Area, in^2				
Seed	113.1	38.5	113.1	78.5
Blanket	153.4	535.8	103.3	338.5

FIGURE 16. Seed-Blanket Model Schematic.
KS-75992

TABLE 8. E COMPARISONS FOR VARIOUS SEEDS, BLANKETS, AND CONTROL SYSTEMS BASED ON A SEED-BLANKET UNIT CELL ANALYSIS AT END OF ONE SEED LIFE

	Highly Enriched Seed			Denatured Seed*		
	Core Life, efph	E Mwy/st U_3O_8	Mwd/mt**	Core Life, efph	Mwy/st U_3O_8	Mwd/mt**
Natural thoria blanket						
Movable fuel	20,900	10.1	8,950	23,600	10.8	14,114
Soluble boron	19,000	8.9	8,300	20,000	9.4	11,600
Natural urania blanket						
Movable fuel	70,000	9.8	8,650	28,000	8.6	7,585

*Denatured seed is comprised of 20 wt % U-235 and 80 wt % U-238.
**Average Mwd/mt of blanket region only.

The one-seed-life comparison just presented does not take into account the depletion capability of the blanket. In a practical reactor system, multiple seeds would be used to drive the blanket to a much higher depletion than that obtained by a one-seed-life model. To make a fair comparison between highly enriched seeds and denatured seeds, therefore, each case should be compared at the same blanket depletion. Such a comparison is presented in Table 9 for an average depletion of 10,000 Mwd/mt. These data show in all cases that the highly enriched fuel maximizes the value of E.

C. Isotopic Power Fraction Comparison

An isotopic power fraction comparison is shown in Figure 17 for the denatured-seed, natural thoria blanket system with movable fuel control and in Figure 18 for the highly enriched seed, natural thoria blanket system with movable fuel control. As can be seen from these curves, the U-235 power fraction decreases to 0.44 at EOL for the denatured seed and to 0.50 at EOL for the highly enriched seed. For the denatured case, the U-233 buildup in the blanket accounts for an EOL U-233 power fraction of 0.38, and the Pu-239 buildup in the seed accounts for an EOL Pu-239 power fraction of 0.12. For the highly enriched case, the U-233 buildup in the blanket accounts for an EOL U-233 power fraction of 0.48.

A corresponding isotopic power fraction comparison is shown in Figure 19 for the denatured-seed, natural urania blanket system with movable fuel control and in Figure 20 for the highly enriched seed, natural urania blanket system with movable fuel control. As can be seen from these curves, the U-235 and Pu-239 power fractions are very similar for the highly enriched and denatured systems. The buildup of Pu-239 in the denatured system results in an EOL Pu-239 power fraction of 0.04 for the seed region and of 0.42 for the blanket region. The highly enriched system results in an EOL blanket Pu-239 power fraction of 0.45.

D. Sensitivity of Blanket Power Fraction to Seed-Blanket Size

A study was performed on a natural urania blanket with a denatured seed to determine how the blanket power fraction (Pf^B) will vary with seed-blanket size. The model used in the analysis is the same as that described in Section V.A with the exception that the fuel lattice is a homogenization of the CE System 80 lattice, as described in Section III.A.

Only BOL calculations were performed on the seed-blanket configuration. These calculations were carried out by first selecting a seed region size and then by adjusting the blanket region size to obtain core reactivities (k_∞) of 1.0 and 1.11.

TABLE 9. E COMPARISONS FOR VARIOUS SEEDS, BLANKETS, AND CONTROL SYSTEMS BASED ON A SEED-BLANKET UNIT CELL ANALYSIS AT BLANKET AVERAGE DEPLETIONS OF 10,000 Mwd/mt

	E, Mwy/st U_3O_8	
	Highly Enriched Seed	Denatured Seed*
Natural thoria blanket		
Movable fuel	10.3	10.1
Soluble boron	9.3	9.1
Natural urania blanket		
Movable fuel	10.5	9.8

*Denatured seed is comprised of 20 wt % U-235 and 80 wt % U-238.

As a result of this study, the curves in Figure 21 were produced, and the following conclusions are made:

1. The blanket power fraction is insensitive to the seed radius at a particular BOL core reactivity.
2. The blanket power fraction is very sensitive to changes in the BOL core reactivity.
3. The maximum seed radius obtainable for a specific BOL core reactivity increases with increasing BOL k_∞; no lower limit on seed radius appears to exist at any BOL reactivity.

E. Sensitivity of E to Percentage of Denaturing in Seed

The denatured results reported in Section V.B. were based on a denatured seed with a UO_2/ZrO_2 fuel pellet. The UO_2 in the pellet is assumed to be comprised of 20% U-235 and 80% U-238. This section describes a sensitivity study that evaluated E for a UO_2 system with 10% U-235 and 90% U-238. The comparison is made for a denatured seed assuming a natural thoria blanket and employing movable fuel as the reactivity control system.

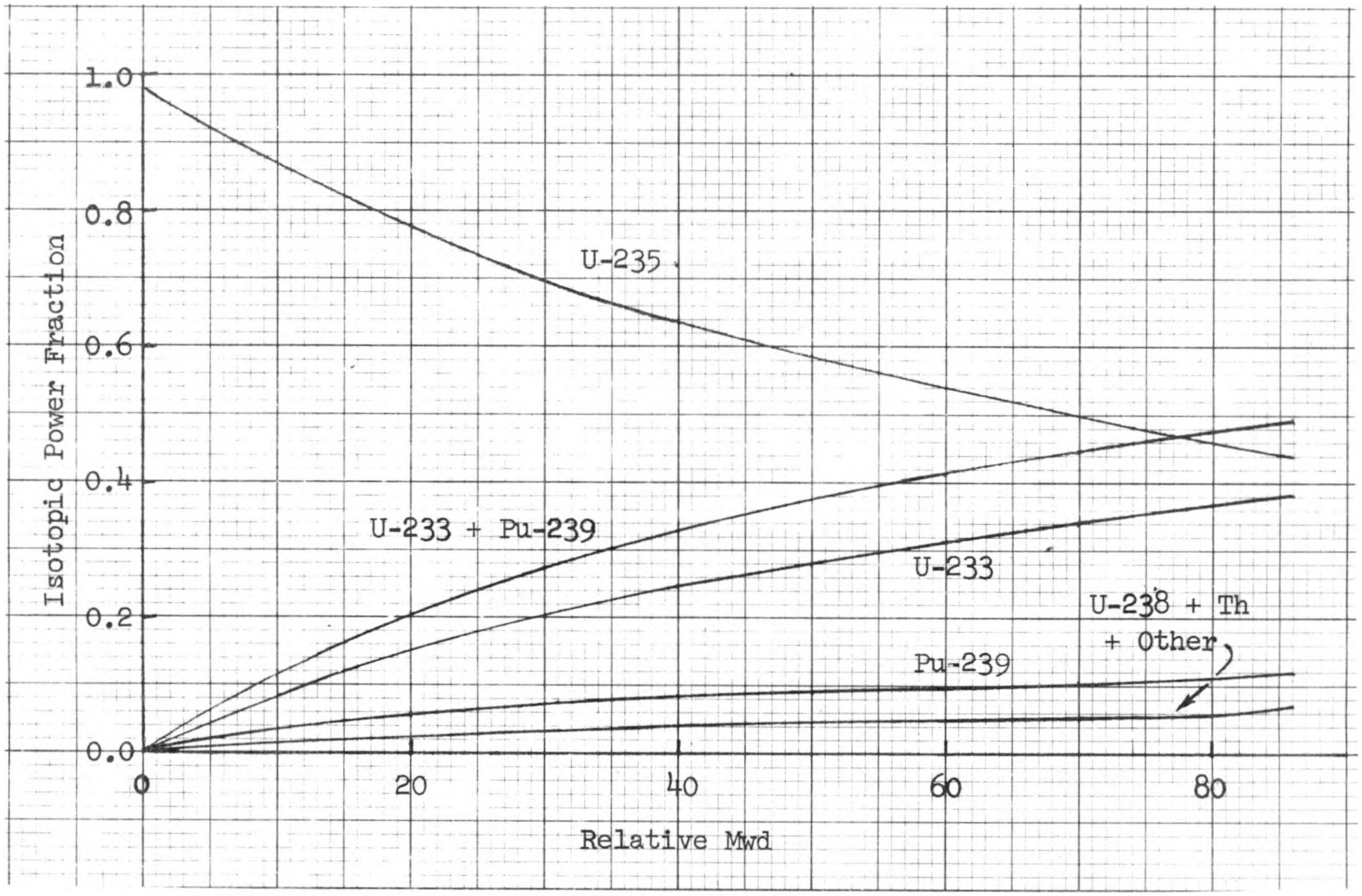


FIGURE 17. Isotopic Power Fraction Comparison for Denatured Seed (20% U-235, 80% U-238), Natural Thoria Blanket Seed-Blanket Cell. KS-75993

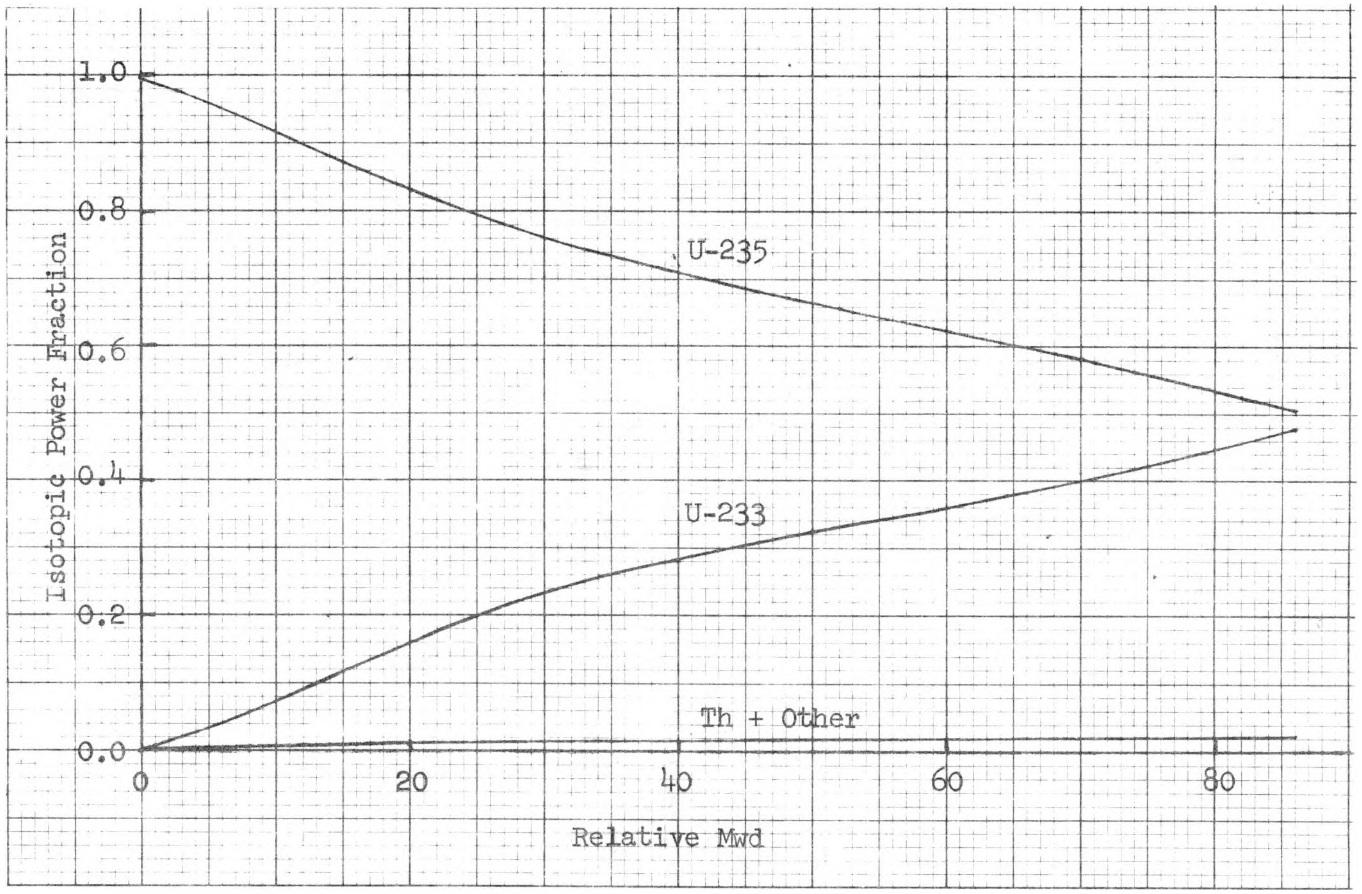


FIGURE 18. Isotopic Power Fraction Comparison for Highly Enriched Seed, Natural Thoria Blanket Seed-Blanket Cell.

KS-75994

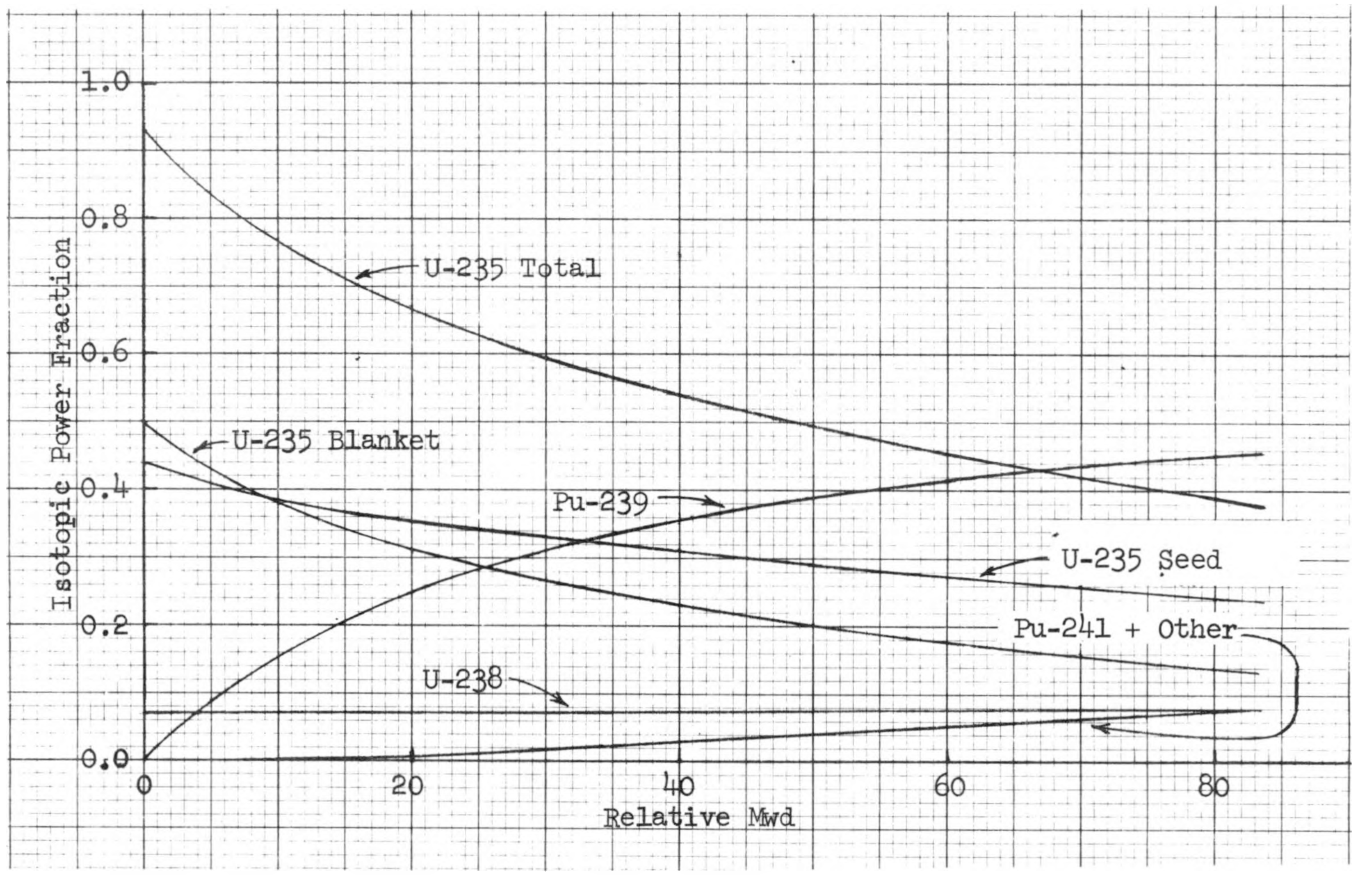


FIGURE 19. Isotopic Power Fraction Comparison for Denatured Seed (20% U-235, 80% U-238), Natural Urania Blanket Seed-Blanket Cell. KS-75995

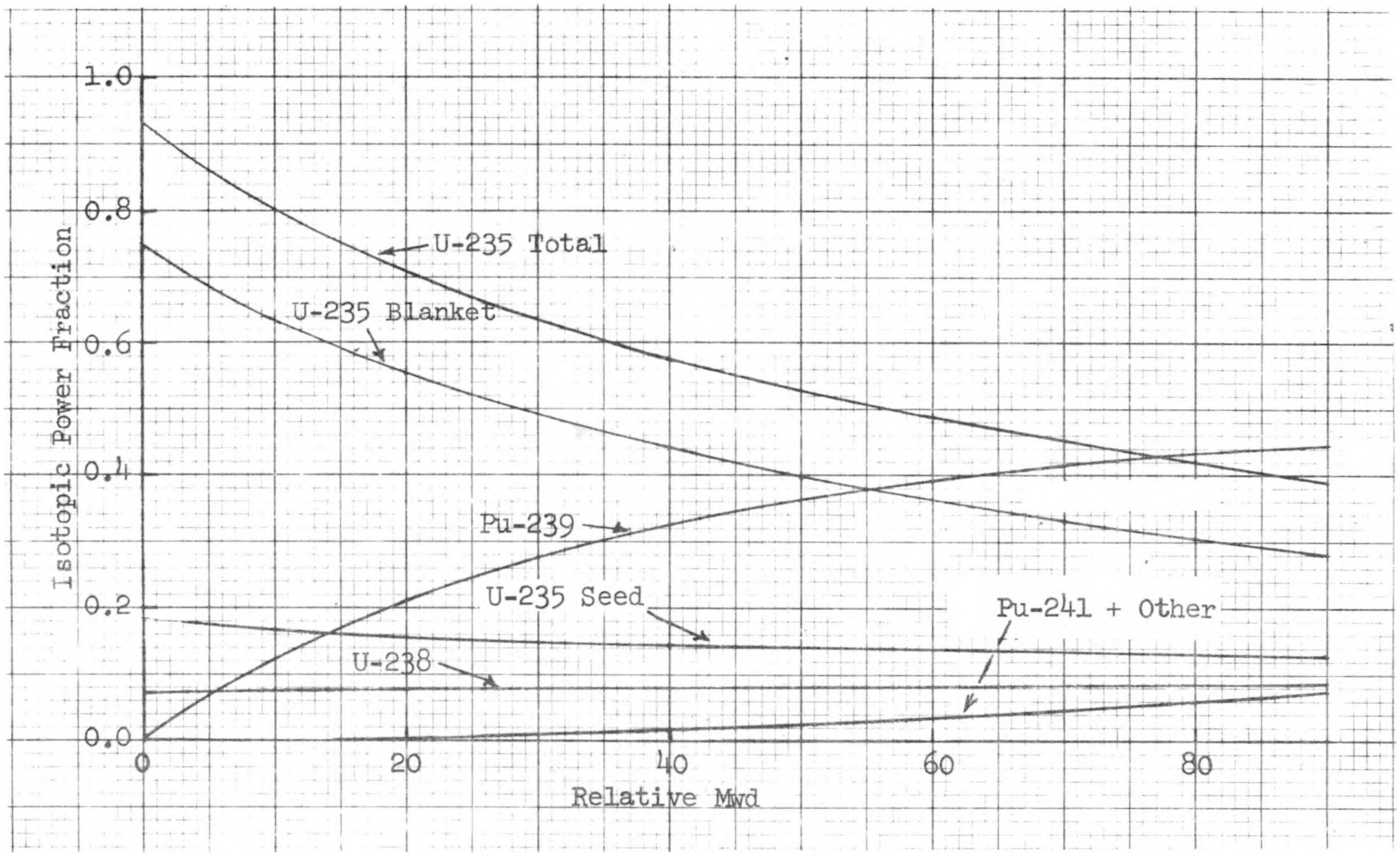


FIGURE 20. Isotopic Power Fraction Comparison for Highly Enriched Seed, Natural Urania Blanket Seed-Blanket Cell.

KS-75996

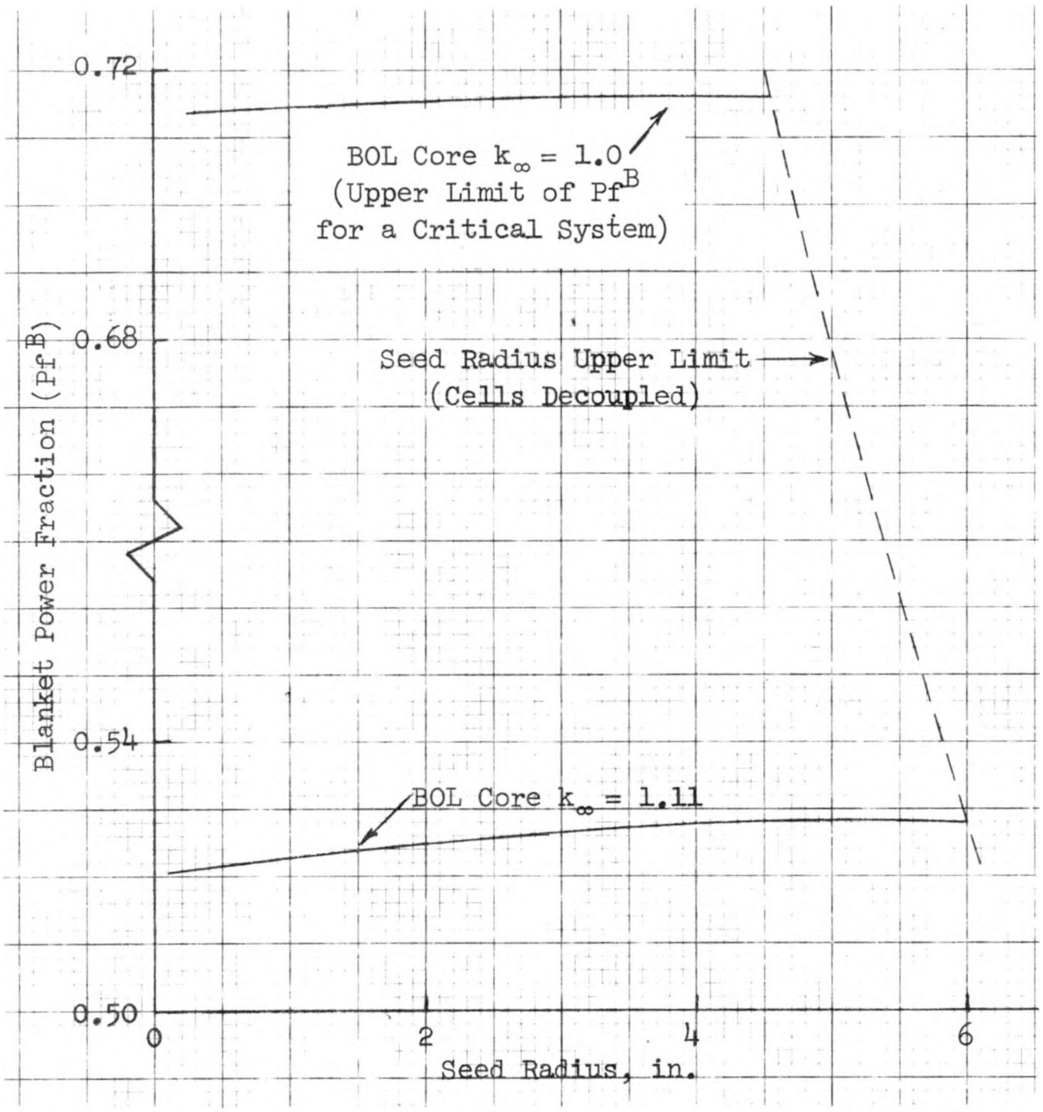


FIGURE 21. Blanket Power Fraction Comparison for Seed-Blanket Cell Analysis (Denatured Seed, Natural Urania Blanket).
KS-75997

The model used in the analysis is the same as that described in Section V.A. with the exception of the seed and blanket radial dimensions. In both the 10% and 20% denatured seeds, the fissile fuel load is assumed to be the same, and the radial seed thicknesses are assumed to be 5.0 in. The radial thicknesses of the natural thoria blanket, which were varied to achieve a BOL no-xenon reactivity level of 1.10, were assumed to be 1.86 in. for the 20% denatured case and 1.70 in. for the 10% denatured case.

The results of the analysis show that the 20% denatured seed is capable of maintaining criticality for 24,100 efph with a blanket average depletion of 15,153 Mwd/mt and that the 10% denatured seed is capable of maintaining criticality for 22,900 efph with a blanket average depletion of 14,357 Mwd/mt. This capability would result in a decrease of 4.4% in E (from 11.4 to 10.9 Mwy/st U_3O_8) for the 10% denatured system after one seed life. If both systems are compared at the same blanket average depletion, the 10% denatured system would result in a decrease of 2.9% in E at a blanket average depletion of 14,400 Mwd/mt.

F. Summary of Seed-Blanket Cell Analysis

The seed-blanket cell analysis can be summarized by the following:

1. After one seed life, the use of 20% denatured fuel in the seed would increase the value of E by 4 to 7% compared with the use of highly enriched fuel in the thoria blanket cases. For the urania blanket cases, the denatured fuel would decrease E by about 13%.
2. On the basis of the same blanket average depletion of 10,000 Mwd/mt, the use of highly enriched fuel in the seed would increase the value of E by 2 to 7% compared with the use of 20% denatured fuel in all cases.
3. The close agreement between the seed-blanket cell analysis and the survey analysis adjusted for batch loading provides support for the qualification of the survey method, as was described in Section IV.
4. The blanket BOL power fraction is insensitive to the seed radius for a constant reactivity level.
5. The blanket BOL power fraction is very sensitive to changes in the reactivity level.
6. The reduction in the percentage of denaturing in the seed from 20% to 10% U-235 would decrease the value of E by 4.4% after one seed life and by 2.9% on the basis of the same blanket average depletion of ~14,400 Mwd/mt.

VI. CONCLUSIONS

Studies have been performed to determine the maximum fuel utilization achievable in a once-through converter-burner pressurized water reactor. These studies show that the maximum gain in fuel utilization for a practical PWR that does not reduce neutron leakage is 30%. If a PWR were designed with a power-producing exterior blanket that reduced the neutron leakage from 4% (characterestic of present reactors) to 1%, a maximum overall gain of about 50 to 60% in fuel utilization might be realized.

An overall summary of the studies performed is presented in Table 10. This summary compares maximum calculated fuel utilization values from the seed-blanket survey analysis, the single-rod cell analysis, and the KAPL LSBR burner analysis for six fuel cycle strategies. These strategies are:

1. Batch loading at a power density consistent with pressurized LWRs.
2. Batch loading at a reduced power density.
3. Fuel management based on infinite-medium calculations.
4. Fuel management corrected for leakage typical of present-generation PWRs.
5. Fuel management with a 1% leakage correction to simulate the use of a radial blanket.
6. Fuel management normalized to results for the CE System 80 design.

The fuel management values corrected for leakage typical of present-generation PWRs represent the difference between a single-rod cell calculation and a detailed design based on an efficient present-generation PWR. The fuel management values with 1% leakage correction represent maximum values obtainable in a practical system with a radial blanket to reduce leakage losses.

The results of this work support the following specific conclusions:

1. Reactivity Control Systems

Movable fuel control would produce the highest fuel utilization while soluble boron control always results in the lowest fuel utilization. Movable fuel control would produce up to a 35% increase in fuel utilization for batch loading and up to a 21% increase in fuel utilization for fuel management when compared to soluble boron control. The reactivity worth of a movable fuel control system is enhanced by a seed-blanket arrangement that increases the neutron importance in the seed and thus reduces the amount of fuel movement required.

TABLE 10. MAXIMUM E VALUE COMPARISON

(0 to 50,000 Mwd/mt Average Depletion)*

	E, Mwy/st U_3O_8													
	Single-Rod Cell Analysis						Survey Seed-Blanket				Extrapolated LSBR Data			
	Soluble Boron		Spectral Shift		Simulated Movable Fuel		Denatured Seed		Highly Enriched Seed		Denatured Seed		Highly Enriched Seed	
Fuel Cycle Strategy	UO_2	ThO_2	UO_2	ThO_2	UO_2	ThO_2	UO_2	ThO_2	UO_2	ThO_2	UO_2	ThO_2	UO_2	ThO_2
Batch														
5.34 kw/ft-avg	11.0	9.3	13.4	11.3	14.9	12.2	12.5**	11.0**	16.0**	14.8**	9.9†	9.2†	12.7	12.4
1.78 kw/ft-avg			14.6	14.8										
Fuel management (5.34 kw/ft-avg)														
Infinite medium	15.8	12.6	18.1	14.3	19.3	15.1	17.3	16.4	17.7	16.7	16.1†	15.0†	16.5††	15.3††
Adjusted for leakage typical of present generation pressurized LWRs (~4% leakage)	13.0	11.1	13.6	11.7	15.0	12.8								
Adjusted for leakage with an optimized core reflector (~1% leakage)	15.1	12.2	17.0	13.6	18.2	14.5								
Normalized to detailed studies with leakage	12.4	10.7												

*Single-rod cell data are based on core average depletion; survey seed-blanket and extrapolated LSBR data are based on blanket average depletions.

**Calculated using EOL k_∞ of 1.3 in seed.

†Based on ratio of denatured/highly enriched seed from survey calculations.

††Based on ratio of fuel management/batch from unit cell calculations.

2. Comparison of Uranium and Thorium Fuel Cycles as a Function of Increased Depletion

UO_2 consistently yields a higher fuel utilization value than ThO_2 for the same control system at a power density consistent with pressurized LWRs for average fuel depletions up to 50,000 Mwd/mt and above. For UO_2, the gains in fuel utilization due to increased average depletion from 35,000 to 50,000 Mwd/mt are small. For ThO_2, gains of 8 to 10% may be realized. Average depletions of greater than ~60,000 Mwd/mt are necessary for the fuel utilization of a ThO_2 system to approach that of a UO_2 system.

3. Gains Due to Fuel Management

When compared with batch loading, fuel management would increase fuel utilization for both the ThO_2 and UO_2 concepts by up to 40% for fuel average depletions not exceeding 50,000 Mwd/mt. However, it would be difficult to achieve additional fuel utilization gain solely from improved fuel management practices because present pressurized LWRs have quite efficient fuel management schemes. Further gains might theoretically be obtainable in the limit of continuous fuel management but such gains are not representative of a practical system.

4. Power Density Effects

A reduction in power density would cause a greater increase in fuel utilization for ThO_2 systems than for UO_2 systems. In the specific example considered in this work, a reduction in power density by a factor of 3, while not practical from a design standpoint, would result in an increase in fuel utilization of 30.9% for ThO_2 and of 9.1% for UO_2.

5. Denatured vs Highly Enriched Fuel

For the same assumed blanket depletion, the use of highly enriched fuel in the seed would increase fuel utilization by up to 7% compared with 20% denatured fuel. A reduction of denaturing in the seed from 20% to 10% would result in an additional 3 to 4% decrease in fuel utilization.

6. Effect of W/M Ratio

With the exception of the case with UO_2 with ~5 wt % U-235, W/M ratios in the range of 1.0 to 1.5 would provide the maximum fuel utilization. For UO_2 with ~5 wt % U-235, a small increase in fuel utilization might be realized if the W/M is increased above the range of 1.0 to 1.5.

VII. REFERENCE

1. EPRI-NP-359. "Assessment of Thorium Fuel Cycles in Pressurized Water Reactors." February 17, 1977.