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Preliminary Evaluation of Alternative Fuel Cycle Options Utilizing Fast Breeders

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ABSTRACT

Various "alternative" fuel cycles have been proposed to alleviate concerns about reactor fuel being diverted for the manufacture of nuclear weapons. This report addresses the impact of such fuel cycles on the performance of the classical (homogeneous) oxide-fueled Liquid Metal Fast Breeder Reactor. The primary fuel cycle analyzed is the $^{233}U/^{238}U/^{232}Th$ fuel cycle in which the fissile component ^{233}U is "denatured" with ^{238}U in order to restrict the use of reactor fuel as a source of weapons-usable material. The denatured reactor performance as a function of the fissile enrichment (% ^{233}U in U) is evaluated. The associated energy center concept in which dispersed denatured reactors are coupled to ^{233}U production reactors operating in such secure energy centers is described. The symbiotic system of dispersed/energy center reactors is analyzed both from the standpoint of energy growth supported and dispersed/centralized power production. Lastly, the effects of proposed changes in the nuclear data on both the reactor performance and symbiotic system characteristics are addressed.

I. INTRODUCTION

The primary rationale for considering the potentialities and limitations of alternate nuclear fuel cycles reflects a classic dilemma: a world-wide need for energy and the perceived economic/resource benefits of nuclear-based generating capacity versus a long-standing and growing international concern over nuclear weapons proliferation [1]. Although commercial nuclear reactor fuel may not be the most desirable source for nuclear weapons material, prudence demands that the possibility of reducing even this proliferation risk be examined. Moreover, the nuclear fuel cycle represents one of the few avenues for nuclear weapons proliferation over which international agencies are likely to have some control, unlike other avenues (such as uranium enrichment) which can be implemented on a solely national basis. Additionally, although a detailed study of the proliferation risks of commercial nuclear power must address the cost/benefit implications on all the nuclear fuel cycle components (i.e., enrichment, fabrication, reprocessing, etc.), this report is directed only at examining the impact of alternate fuel cycles on the most numerous components of the fuel cycle, the reactors themselves. Moreover, it is aimed solely at evaluating the impact on a particular type of reactor, the liquid metal fast breeder reactor (LMFBR).

The once-through low-enriched uranium cycle currently employed in light-water reactors (LWRs) displays certain proliferation-resistant characteristics which are useful in evaluating alternate fuel cycle scenarios. The fresh fuel consists of a mixture of two uranium isotopes, fissile ^{235}U in a fertile ^{238}U matrix, at an enrichment low enough to preclude direct use of the fuel material in weapons manufacture. Moreover, since the fissile component of the fuel is not chemically separable from the fertile component, production of weapons-grade material would require sophisticated isotope separation capability. The development of such a capability, which itself presumes a strategic decision due to the time and costs involved, also, of course, permits the utilization of natural uranium as a feed material. Further, although the spent fuel does contain a chemically separable fissile species, ^{239}Pu (bred from the ^{238}U matrix), extraction would require a shielded reprocessing facility due to the presence of highly radioactive fission products. Development of this capability, like that of the isotope separation capability above, would also require decisions well in advance of any weapons fabrication. Further, as a practical matter, the once-through low-enriched uranium cycle is currently being utilized worldwide.

Although nuclear-based energy production is the only regenerative energy source with a large-scale generating potential available at this time, the once-through low-enriched uranium cycle ignores this regenerative capacity, and hence is limited by resource considerations. Therefore, although short-term increases in nuclear electric generating capacity can be achieved by increasing the consumption rate of the resource base, long-term viability and growth of nuclear capacity must ultimately rest with recycle of fissile material and eventual implementation of breeder reactor technology. Investigation of techniques (such as the use of alternate fuel cycles), which have the possibility of minimizing the attractiveness of recycle fuel as a source of nuclear weapons material, is therefore essential to efforts aimed at limiting the long-term proliferation potential of nuclear power.

Many techniques for reducing the proliferation risk associated with breeder reactors have been suggested, including (1) controlled plutonium recycle, (2) fuel "spiking," (3) pre-irradiation, (4) radiological denaturing, and (5) isotopic denaturing. These techniques are addressed at limiting the availability of sensitive nuclear material through international institutional arrangements and/or reducing the attractiveness of reactor fuel as a weapons material source through technical means. Controlled plutonium recycle scenarios would rely on international controls as the means for limiting the availability of sensitive nuclear materials. Fuel "spiking" (the addition or nonremoval of highly radioactive contaminents as a deterrent), as well as pre-irradiation (producing the contaminant *in situ* after fabrication) would also require institutional arrangements to ensure the presence of the contaminant. The concept of radiological denaturing is analogous to that of "spiking" except that the contaminant is not chemically separable from the fissile material. It applies principally to ^{232}U contamination of thorium-derived fuels. The concept of isotopic denaturing (e.g., ^{233}U denatured with ^{238}U) attempts to extend some of the proliferation-resistant aspects of the low-enriched once-through cycle to the recycle/breeder-reactor scenario [1,2].

In the isotopically denatured cycle the fresh fuel would, like the low-enriched-uranium cycle, consist of fissile material (^{233}U) diluted with an isotope of the same element (^{238}U). Unlike the LEU cycle, however, the denatured fuel would also contain ^{232}Th. Due to the presence of the ^{238}U denaturant in the denatured fuel, plutonium would be produced and discharged but presumably in lesser amounts than in the LEU cycle. It should be emphasized that the rationale for investigating the denatured cycle is based primarily on possible resource constraints, and on the subsequent need for the employment of recycle fuel. In this context, the proliferation potential of the denatured fuel cycle is to be compared with that of the reference mixed-oxide (Pu/^{238}U) cycle. It is not apparent that the denatured cycle represents a significant reduction in the proliferation risk associated with commercial nuclear power when compared with the current LEU once-through cycle. [In fact, the increased mass difference inherent in the denatured cycle (^{233}U-^{238}U) relative to enriched natural uranium (^{235}U-^{238}U) may reduce the difficulty of isotopically separating fissile material.] The denatured cycle does, however, appear to have potential in minimizing the possible increase in proliferation risks associated with the introduction of recycle scenarios. It should be noted, however, that such a conclusion must, of necessity, be reached through a detailed analysis of the various factors involved in utilizing reactor fuel as a source of weapons material such as cost, time, etc. For example, how much more difficult is it to isotopically enrich fresh denatured fuel than to chemically remove plutonium from fresh mixed oxide fuel?

The answers to such questions are beyond the scope of this report. This report is predicated on the assumption that the denatured fuel cycle inherently results in a reduced proliferation risk compared to the reference mixed-oxide cycle, and addresses the impact of the denatured cycle on LMFBR performance. This report summarizes and extends the results of previously reported studies on LMFBRs [3,4].

II. IMPACT OF ALTERNATE FUELS ON LMFBR PERFORMANCE PARAMETERS

For the purposes of this scoping evaluation, a representative LMFBR model was adopted as the reference case. The model utilized was based on the design promulgated by the Large Core Code Evaluation Working Group (LCCEWG) as a calculational benchmark [5]. While such an LMFBR does not represent the commercial design of any specific organization, it was felt that the general performance parameters were typical of possible future commercial LMFBRs. As the calculational benchmark represented only the core midplane, the basic design was modified to account for the presence of axial blanket regions. The axial blanket regions were added to the model by assuming the same volume fractions as the corresponding core regions. Table 1 gives some of the principal features of the resulting model. Figure 1 depicts the geometrical configuration of the calculational model utilized in this study. As indicated in Fig. 1, the model represents a "classical" or homogeneous configuration comprised of two core zones (of different fissile enrichments) surrounded by blanket material. The use of a heterogeneous core design (i.e., interspersed fissile and blanket assemblies) was not addressed in the calculations done for this study. Certain other modifications were also incorporated into the model for the sake of later comparisons. The composite cladding and structural material specified in the benchmark was replaced by SS316. Further, the isotopic plutonium composition specified by the Large Heterogeneous Reference Fuel Design Study (LHRFDS) [6], given in Table 2, was utilized as the feed material in the reference case. The depleted uranium used for fertile material was assumed to contain 0.20% ^{235}U.

The principal neutronics analysis tool utilized in this study was the CITATION computer code [7]. A three-step calculational procedure was employed for the reference fuel burnup calculation. An initial depletion calculation was performed to determine the reload enrichment necessary to achieve an effective multiplication factor of unity at the end of the equilibrium cycle (i.e., cycle for which the discharge isotopics are identical with the preceding cycle). A fuel management scheme in which 1/3 of each core zone (and corresponding axial blanket extensions) and 1/6 of the radial blanket

Table 1. Parameters of Basic Homogeneous Core LMFBR Design Used in This Study

Power (MWe)	1200
Power (MWth)	3085
Fuel Density (% TD)	92
Core Height (in.)	42
Axial Blanket Height (in.)	13
Rods/Assembly	271
Spacers	wire
Channel Pitch (in.)	5.47
Rod Pitch/Diameter (P/D)	1.20
Rod OD (in.)	0.260
Cladding Thickness (in.)	0.012
Fuel Gap (in.)	0.0055
Channel Wall Thickness (in.)	0.080
Fuel Volume Fraction (%)	43.3
Structure Volume Fraction (%)	17.4
Sodium Volume Fraction (%)	39.3
Cylindrical Model	
Inner Core Max. Radius (in.)	40.3
Outer Core Max. Radius (in.)	56.4
Radial Blanket Thickness (in.)	15.3

assemblies were replaced annually was utilized. A capacity factor of 0.75 was assumed. Using a program written to be compatible with CITATION, additional computations were performed to determine the minimum first-cycle fissile loading consistent with criticality requirements over the initial cycles. Finally, using the initial core and reload specifications determined in the above two steps, a final depletion calculation was done in which the control material was adjusted so as to maintain a just-critical configuration at each time during the cycles.

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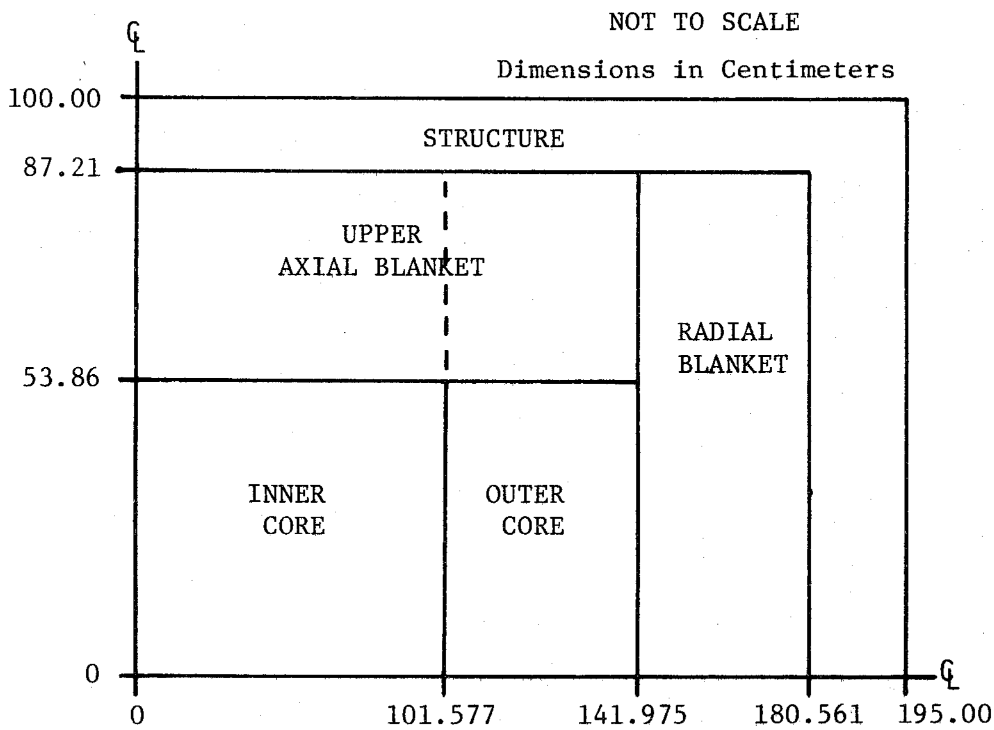

Fig. 1. Geometric Model Utilized in the Alternate Fast Breeder Fuel Cycle Study.

Various proposed alternative fissile/fertile fuel combinations were also analyzed using the above procedure. The fissile and fertile assembly volume fractions given in the reference case were maintained for each of the combinations analyzed. Atomic number densities were adjusted to reflect differences in the various oxides involved [e.g., $(Pu/U)O_2$, UO_2, ThO_2, etc.]. No attempt was made, however, to account for differences in the thermophysical properties (e.g. melting point, thermal conductivity, etc.) of the alternate materials relative to the reference system.

Nine-group cross-section sets were generated for various reactor regions based on ENDF/B-IV data and using the MINX/SPHINX/AMPX [8-10] code system. The adequacy of the nuclear data used, however--particularly for the significant isotopes involved in the alternative fuel cycles--is open to some question. Recent measurements of the capture cross-section of ^{232}Th [11], the primary alternate fertile material, indicate significant discrepancies between the measured and tabulated ENDF/B-IV cross sections for the energy

Table 2. Isotopic Plutonium Composition

Isotope	Wt %
^{238}Pu*	0.00997
^{239}Pu	0.67272
^{240}Pu	0.19209
^{241}Pu	0.10127
^{242}Pu	0.02395

*Not considered in this study.

range of interest. Additionally, the adequacy of the nuclear data for ^{233}U in the LMFBR spectral range has also been questioned [12]. In light of the possible uncertainties in the basic nuclear data and the lack of detailed design optimization for the reactors operating on the alternative fuels, it is thus prudent to regard the results reported below as preliminary evaluations, subject to revision as more precise data become available.

Table 3 presents a summary of some overall performance characteristics of LMFBRs operating on certain alternative fuels. The detailed mass flows for each case listed in Table 3 are given in the Appendix. The breeding ratios given in Table 3 were calculated utilizing the following definition:

$$BR = \frac{\text{fertile capture rate}}{\text{fissile absorption rate}} \qquad (1)$$

It should be noted, however, that Eq. (1) does in fact represent an approximation in that it assumes that the fissile production rate is equal to the fertile capture rate. For the reference Pu/^{238}U system, the difference between the two rates is insignificant due to the low interaction rate of the intermediate nuclide, ^{239}Np, resulting from its short half-life (large decay constant). For the alternative fuels involving ^{232}Th, however, the use of Eq. (1) does represent a slight approximation due to the much longer half-life of ^{233}Pa (relative to ^{239}Np). Additionally, the common European practice of assigning "equivalence factors" [13] to the various fissile and fertile isotopes based on their relative worth with respect to k_{eff} has not been followed.

Table 3. Alternate Oxide-Fueled Homogeneous LMFBR Performance Parameters (1 year ex-reactor inventory, 75% capacity, r = 0.98, f = 1.00)

Case	Core Fissile	Core Fertile	Uranium Enrichment	Axial Blanket	Radial Blanket	Breeding Ratio (MEC)[a]	CFDT (yr)
1	Pu	^{238}U	–	^{238}U	^{238}U	1.272	12.7
2	Pu	^{238}U	–	^{238}U	^{232}Th	1.272	13.0
3	Pu	Th	–	^{232}Th	^{232}Th	1.144	36.2
4	^{233}U	^{238}U	∿12%	^{238}U	^{232}Th	1.127	24.3
5	^{233}U	^{238}U	∿12%	^{232}Th	^{232}Th	1.121	26.6
6	^{233}U	$^{238}U/^{232}Th$	20%	^{232}Th	^{232}Th	1.086	43.2
7	^{233}U	$^{238}U/^{232}Th$	40%	^{232}Th	^{232}Th	1.048	116.4
8	^{233}U	^{232}Th	100%	^{232}Th	^{232}Th	1.020	-696.4

[a]Middle of equilibrium cycle.

Since some of the fissile isotopes produced in many of the alternate fuels do not correspond to the fissile materials in the fresh fuel itself, a "compound fissile doubling time" (CFDT) for the various systems is given in Table 3. Although the fissile doubling time is defined in a manner similar to that of a "reactor" or "system" doubling time, i.e.,

$$\text{CFDT} = \frac{(\ln 2)\ f\,(M^{\text{Initial core}} + M^{\text{Reload}})}{(r\ M^{\text{discharge}}_{\text{eq. cycle}} - f\ M^{\text{charge}}_{\text{eq. cycle}})} \qquad (2)$$

the modified terminology is employed to emphasize the difference in the charge and bred material isotopics. It should also be noted that Eq. (2), when coupled with the assumption of annual refueling, presumes a one-year out-of-reactor reprocessing/refabrication delay. The parameter r used in Eq. (2) represents the reprocessing recovery factor assumed, and f is the fuel fabrication loss factor.

As indicated in Table 3, the overall breeding performance of the alternative fuels is reduced relative to the reference Pu/^{238}U fuel (case 1).

The replacement of plutonium by ^{233}U as the fissile material (compare cases 2 and 4, 3 and 8) results in a significant decrease in the breeding ratio and increase in the fissile doubling time. Similarly, the replacement of ^{238}U by ^{232}Th as the core fertile material also results in a significant breeding ratio penalty (cases 1 and 3, 5 and 8). It is also evident from Table 3 that the choice of the radial blanket fertile material does not affect the overall breeding performance significantly.

In discussing the neutronic properties of the various fissile isotopes it is useful to distinguish between the two functions of the fissile material in a fast breeder reactor: energy production and excess neutron production. As indicated by Fig. 2, the fission cross section of ^{233}U is significantly higher than that of ^{239}Pu over the energy range of interest. Moreover, as shown in Fig. 3, the capture-to-fission ratio of ^{233}U is significantly lower than that of ^{239}Pu. Thus, if the minor differences in the energy release per fission of the two isotopes is ignored, ^{233}U is clearly the superior isotope in terms of energy production per atom destroyed. However, regarding the second function of the fissile material--the production of the excess neutrons required for breeding--^{239}Pu has a clear advantage. Figure 4 indicates the superior breeding potential (represented by η, the number of neutrons produced per fissile absorption) of ^{239}Pu relative to ^{233}U, especially at the higher neutron energies. This effect is due to the higher ν value (neutrons produced per fission) of ^{239}Pu; an effect which outweighs the smaller capture-to-fission ratio of ^{233}U. It should be noted that this fissile replacement effect is blurred somewhat for some of the fuels examined, since fissile isotopes (other than the fuel isotope) are both produced and consumed over the cycle.

The reduced breeding potential of the alternate fuels is further accentuated by the fission cross section differences between ^{238}U and ^{232}Th. As illustrated by Fig. 5, the fission cross section of ^{232}Th is significantly lower (by approximately a factor of 4) than that of ^{238}U. Fertile fissions, in addition to contributing to the overall energy production, also result in neutron production which is not at the expense of a fissile nucleus. In a fuel utilizing ^{238}U as the fertile material, 15-20% of the fissions

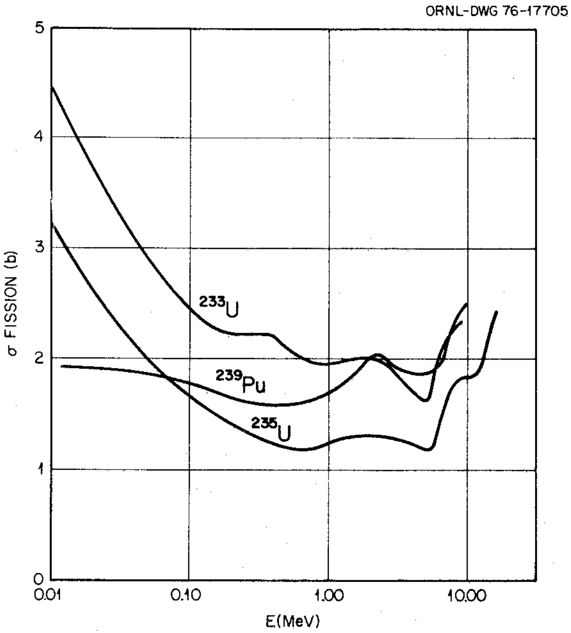

Fig. 2. Fission Cross Section (σ_f) of ^{233}U, ^{235}U, and ^{239}Pu.[14]

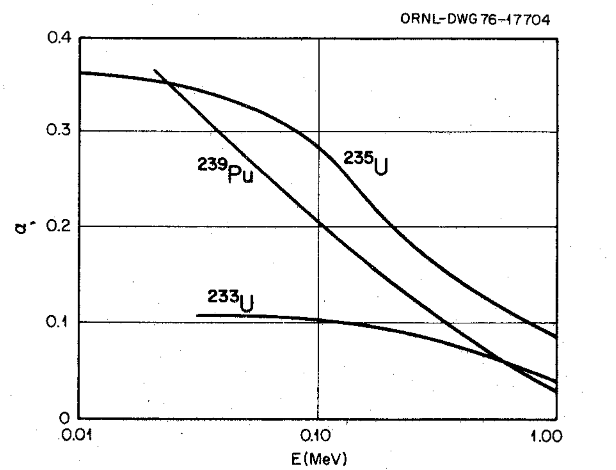

Fig. 3. Capture-to-Fission Ratio (α) of ^{233}U, ^{235}U, and ^{239}Pu.[14]

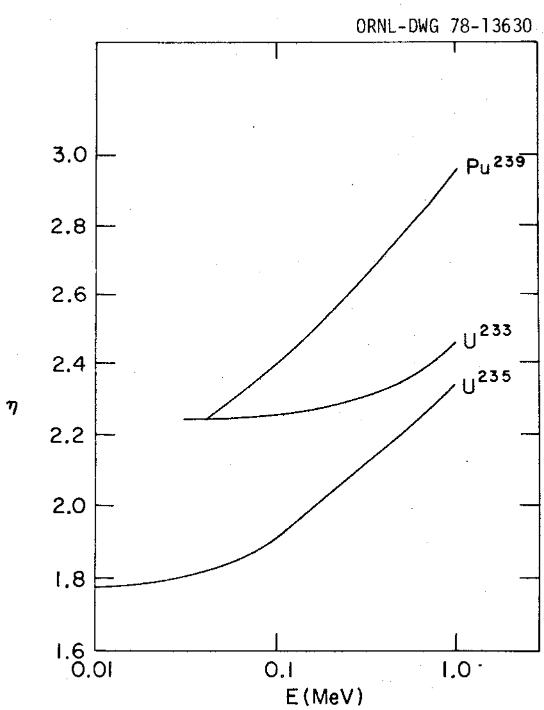

Fig. 4. Neutrons Produced/Absorption Ratio (η) for ^{233}U, ^{235}U, and ^{239}Pu.[14]

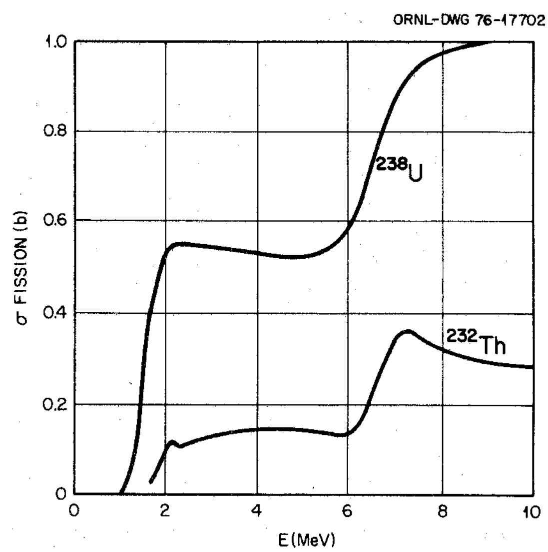

Fig. 5. Fission Cross Section (σ_f) of ^{232}Th and ^{238}U.[14]

occur in the fertile material versus 4-5% in a ^{232}Th system. In the denatured fuels, however, this effect is mitigated somewhat since the fuel material typically contains both fertile nuclides.

The denatured reactors (cases 4-7), of course, reflect the breeding ratio penalty inherent in replacing ^{239}Pu by ^{233}U as the primary fissile material, and, in some of the systems addressed, partial replacement of ^{238}U by ^{232}Th as the core fertile material. Possibly more significant than the overall breeding ratio of the denatured systems, however, is the fraction of the breeding ratio which is attributable to ^{233}U production since it is this component which determines the degree of self-sufficiency of the denatured systems. Both ^{233}U and ^{239}Pu are produced in the denatured fuels, but, as mentioned above, the bred ^{239}Pu cannot be recycled back into the denatured systems.

Figure 6 schematically illustrates the two components of the breeding ratio (mid-equilibrium cycle) for some of the denatured reactors. (It should be noted that the ^{233}U/^{232}Th system given on Fig. 6 is not denatured. It is included only since it represents an upper bound on the ^{233}U/U enrichment.) As depicted in Fig. 6, the ^{233}U component of the breeding ratio increases as the allowable degree of denaturing is increased, thereby permitting more ^{232}Th in the fuel material. More importantly, however, the magnitude of the ^{233}U component of the breeding ratio is very sensitive to the allowable degree of denaturing at the lower enrichments (i.e., between 12% and 20%). Thus, even though the overall breeding ratio decreases as the allowable enrichment is raised, the decrease in the required ^{233}U makeup is even more drastic and presents a strong incentive from a performance standpoint to set the allowable enrichment as high as practicable. It should also be noted that the 12% value for the ^{233}U/U enrichment represents the minimum average enrichment required for the reactor model utilized. The fuel consisted solely of UO_2 (no ThO_2), and hence the system requires considerable ^{233}U as makeup material. Moreover, the 12% value represents an average between the inner core region (10%) and the outer core region (14%).

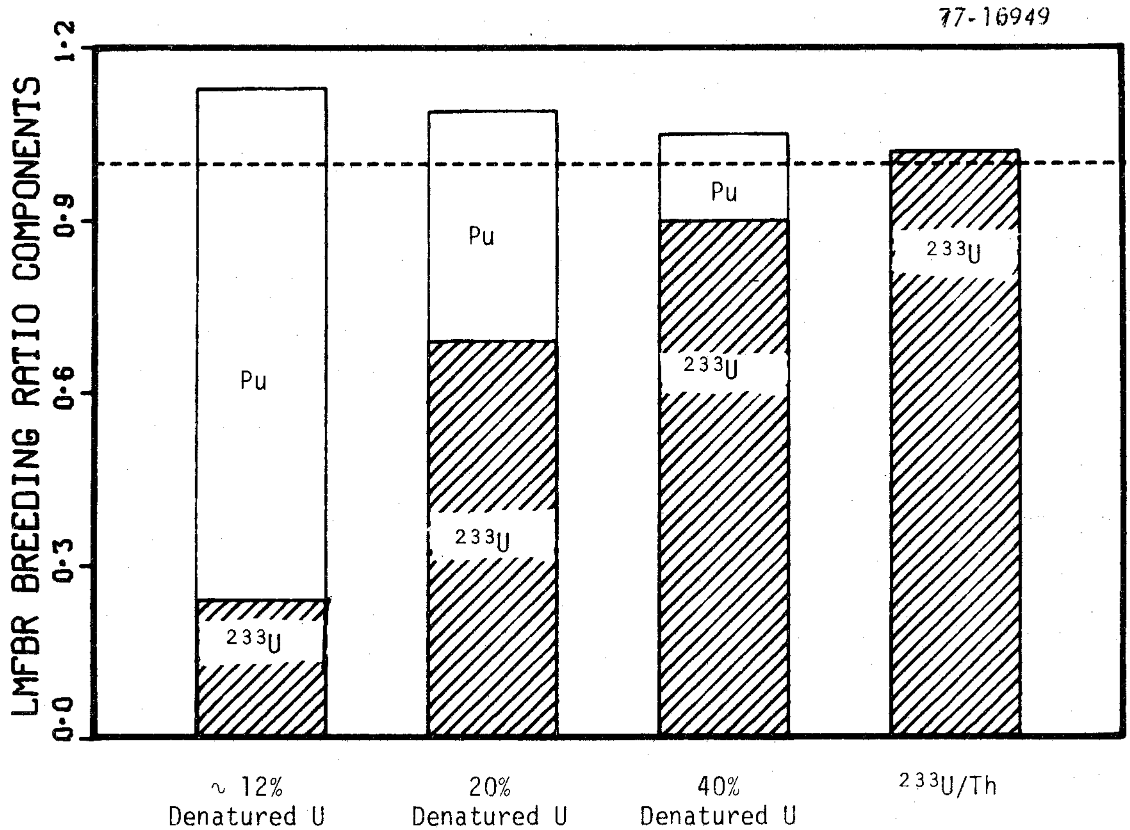

Fig. 6. Alternate Fast Breeder Fuel Cycle Breeding Ratio Components.

Since all of the denatured systems require an initial inventory of ^{233}U, as well as varying amounts of ^{233}U as makeup material, a second class of reactors, whose purpose is to produce the requisite ^{233}U, must be considered in evaluating the denatured fuel cycle. Three possible LMFBR candidates for this role are listed in Table 3: a Pu/^{238}U reactor with a ThO_2 radial blanket; a Pu/Th LMFBR; and a $^{233}U/^{232}Th$ breeder. Figure 7 illustrates the isotopic fissile production (destruction) properties of the three systems. Clearly, each system has its own unique features. From the standpoint of ^{233}U production capability, the hybrid Pu/Th LMFBR is clearly superior. However, it does require a significant quantity of fissile plutonium as makeup material since it essentially "transmutes" plutonium into ^{233}U. The $(Pu/^{238}U)O_2$ + ThO_2 radial blanket LMFBR generates significantly less ^{233}U than the Pu/Th reactor, but also markedly reduces the required plutonium feed. In fact, for the case illustrated, this system actually produces a slight excess of plutonium. The third possible LMFBR system considered is the

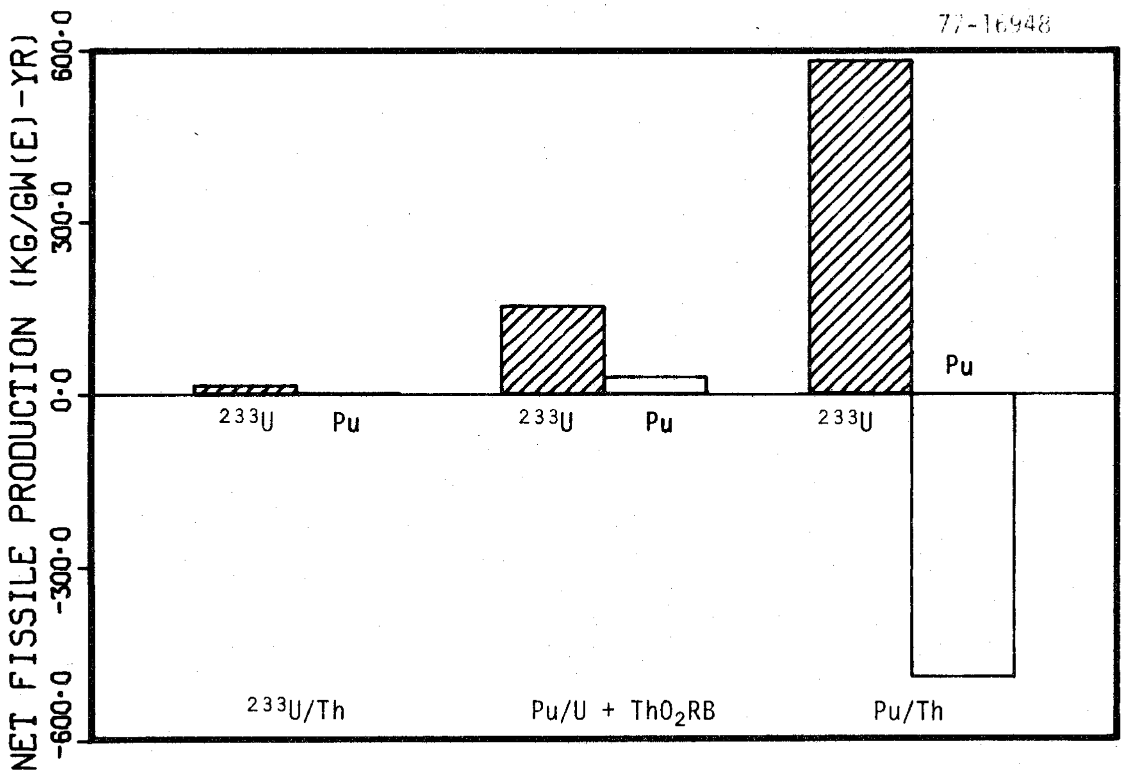

Fig. 7. Equilibrium Cycle Net Fissile Production for Alternate Oxide-Fueled ^{233}U Production Reactors.

^{233}U/^{232}Th breeder, which is characterized by a very small excess ^{233}U production. Further, it does not provide a means for utilizing the plutonium which will be bred in the denatured system.

It should also be noted that all three of these ^{233}U "production" reactors, since they are not denatured, must be subject to rigorous safeguards; that is, operated only in nuclear weapons states or in internationally controlled energy centers. The energy center concept is depicted in Fig. 8. The principal feature of such a scenario is that no plutonium-bearing fresh fuel is ever required to leave the energy center complex. All fuel shipped from the center is denatured, and the plutonium-bearing spent fuel shipped back from the denatured systems is highly radioactive. As shown in Fig. 8 such an energy center would contain, in addition to the "transmuters," other sensitive nuclear fuel cycle activities such as reprocessing. The mature system as depicted by Fig. 8 is a true symbiotic one in which the "transmuters" consume the plutonium generated in the denatured systems and produce ^{233}U. The denatured reactors in turn consume the ^{233}U and produce plutonium.

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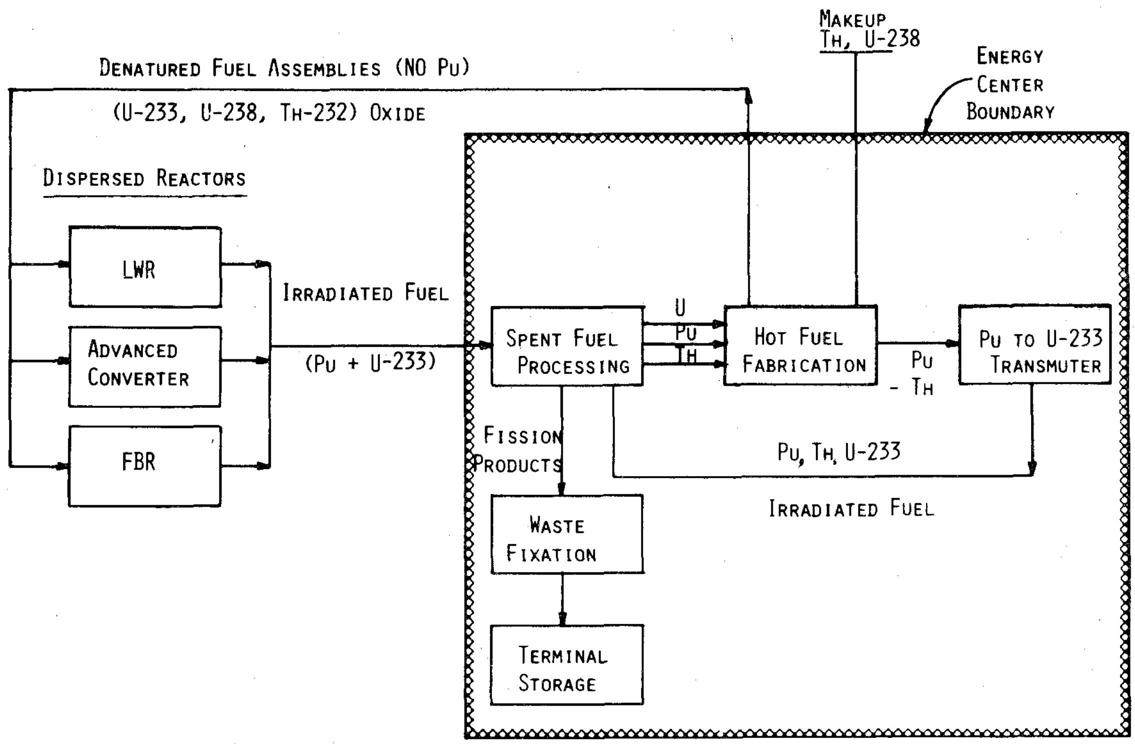

Fig. 8. Schematic Fuel Flow for Denatured Dispersed/Energy Center Concept.

III. SYMBIOTIC BREEDER REACTOR SYSTEMS

For symbiotic systems involving two types of reactors such as that depicted by Fig. 8, two characteristic parameters are of interest: the growth potential and the support (or power) ratio--the number of dispersed reactors supported per transmuter. To develop appropriate measures for these parameters, the mass flow of each type of fissile material within the system must be considered. The time rate of change (indicated using "prime" notation) of each fissile component is given by

$$I_i'(t) = \sum_j p_{ij} R_j(t) \quad , \tag{3}$$

where $I_i(t)$ is the inventory of the ith fissile material, $R_j(t)$ is the number of reactors of type j operating, and p_{ij} is the net production rate (per year) of the ith fissile material in reactor type j. Generally, p_{ij} is a time-dependent quantity to allow for changes in the production rate due to the approach to equilibrium cycle, variable capacity factor, etc. For a constant capacity factor, however, the equilibrium cycle production rates are representative of the long-term history of the reactor. Hence, p_{ij} is defined as

$$p_{ij} = (r_i M_{ij}^{discharge} - f_i M_{ij}^{charge}) \cdot N \quad , \tag{4}$$

where $M_{ij}^{discharge}$ is the mass of ith fissile component discharged/cycle for reactor type j, M_{ij}^{charge} is the mass of the ith material charged/cycle, r is the reprocessing recovery factor for material i, f is the fabrication factor, and N is the number of cycles/year. Defining c_{ij} as the quantity of fissile material i associated with a type j reactor, a relationship between the number of reactors operating and the fissile inventories can be established, i.e.,

$$\underline{I}(t) = \underline{\underline{C}}\ \underline{R}(t) \quad . \tag{5}$$

Substituting Eq. (5) into Eq. (3) and specializing to a 2 x 2 system of reactors/fissile materials yields

$$R_1'(t) = a_{11}\ R_1(t) + a_{12}\ R_2(t) \quad , \tag{6a}$$

$$R_2'(t) = a_{21}\ R_1(t) + a_{22}\ R_2(t) \quad , \tag{6b}$$

where

$$\underline{\underline{A}} = \underline{\underline{C}}^{-1}\ \underline{\underline{P}} \quad . \tag{7}$$

Equations (6a-6b) represent a coupled set of first-order differential equations, the solution of which is given by

$$R_1(t) = A_1\ e^{\lambda_1 t} + A_2\ e^{\lambda_2 t} \quad , \tag{8a}$$

$$R_2(t) = \left(\frac{\lambda_1 - a_{11}}{a_{12}}\right) A_1\ e^{\lambda_1 t} + \left(\frac{\lambda_2 - a_{11}}{a_{12}}\right) A_2\ e^{\lambda_2 t} \quad , \tag{8b}$$

where

$$A_1 = \frac{(a_{11} - \lambda_2)\ R_1(0) + a_{12}\ R_2(0)}{(\lambda_1 - \lambda_2)} \quad , \tag{9a}$$

$$A_2 = \frac{(a_{11} - \lambda_1)\ R_1(0) + a_{12}R_2(0)}{(\lambda_2 - \lambda_1)} \quad , \tag{9b}$$

and

$$\lambda_1,\lambda_2 = \frac{(a_{11} + a_{22}) \pm \sqrt{(a_{11} - a_{22})^2 + 4a_{12}\ a_{21}}}{2} \quad . \tag{10}$$

$R_1(0)$ and $R_2(0)$ are the initial number of reactors of each type. It should be noted that the assumption that $\lambda_1 \neq \lambda_2$ has also been made. Using Eqs. (8a-8b), the ratio of type 1 reactors to the number of type 2 reactors, $P(t)$, is given by

$$P(t) = \frac{R_1(t)}{R_2(t)} = \frac{\dfrac{a_{12}}{\lambda_1 - a_{11}} \left[1 + \dfrac{A_2}{A_1}\ e^{(\lambda_2-\lambda_1)t}\right]}{\left[1 + \dfrac{A_2(\lambda_2 - a_{11})}{A_1(\lambda_1 - a_{11})}\ e^{(\lambda_2-\lambda_1)t}\right]} \quad . \tag{11}$$

Associating the positive sign of Eq. (10) with λ_1, the asymptotic value of Eq. (11) is just

$$\lim_{t\to\infty} P(t) = P_\infty = \frac{a_{12}}{\lambda_1 - a_{11}} = \left(\frac{\lambda_1 - a_{22}}{a_{21}}\right) \quad . \tag{12}$$

As indicated by Eq. (12), the asymptotic value is independent of the initial conditions, and thus can be viewed as a "characteristic" ratio for the given system -- approached regardless of the initial configuration.

While P_∞ represents the natural or characteristic support ratio, two additional ratios are useful in evaluating the system. These ratios define the operating limits of the true (i.e., self-sufficient) symbiotic system. For a self-sufficient system, the rate of change of each component must be greater (or equal to) zero. Thus, from Eqs. (6a-6b)

$$P_1 = \frac{R_1}{R_2} = -\frac{a_{12}}{a_{11}} \tag{13a}$$

and

$$P_2 = \frac{R_1}{R_2} = -\frac{a_{22}}{a_{21}} \tag{13b}$$

For the purposes of illustration, it is useful to consider a system consisting of Pu/Th transmuters and dispersed 20% denatured reactors. Figure 9 illustrates the time-dependent support ratios using each of the limits given by Eqs. (13a-13b) as the initial condition. As indicated by Fig. 9, the asymptotic ratio is approached in either case. It should be emphasized, however, that although the support ratio asymptotes to a constant value, the total system power continues to increase. Figure 10 depicts the power production of each symbiotic component as well as the total system power for the case where $P(0) = P_1$. The power from each component as well as the overall system power increases with time for this system. The analogous case for $P(0) = P_2$ is given by Fig. 11.

Both of these initial conditions [i.e., Eqs. (13a-13b)], however, reflect an implicit assumption that enough ^{233}U will be available to fuel the initial dispersed reactors. More realistically, while plutonium (from reprocessed LWR fuel, for example) may be available, it is unlikely that significant stockpiles of ^{233}U will exist when such a system is introduced. A more realistic implementation scenario for the denatured fuel cycle would contemplate the construction of an initial transmuter followed by construction of dispersed denatured reactors only as the required ^{233}U became available. The transmuter, of course, would require an exogeneous source of plutonium until the minimum sustainable symbiotic ratio is attained. To address this scenario, the number of transmuters, $R_2(t)$, in Eq. (6a), is taken as constant (=1 for convenience). Therefore, the number of dispersed reactors (and the support ratio) is given by

$$R_1(t) = P(t) = P_1[1 - e^{a_{11}t}] . \tag{14}$$

The time required to attain the minimum support ratio is given by

$$t_{min} = \frac{1}{a_{11}} \ln [1 - (P_2/P_1)] . \tag{15}$$

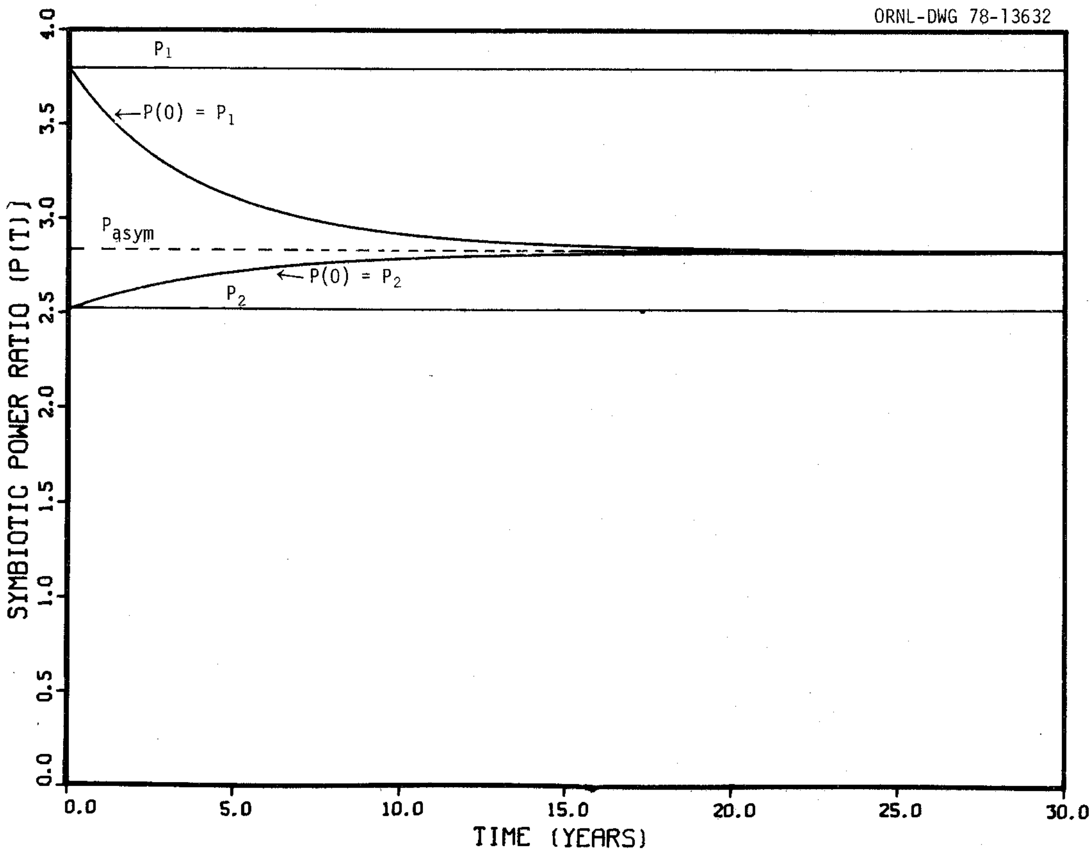

Fig. 9. Power Ratio Behavior for a Symbiotic System of Pu/Th Transmuters and 20% Denatured Reactors, Initiated Within the Symbiotic Window.

Thus, for $0 < t < t_{min}$, the support ratio is given by Eq. (14). For $t > t_{min}$, Eq. (11) with the initial condition that $P(t_{min}) = P_2$ is the pertinent solution. Figure 12 indicates this combined solution. As illustrated, $t_{min} \simeq 22$ years for this particular combination of reactors. Figure 13 gives the power of each symbiotic component as well as the overall power for this scenario. For such an implementation strategy in which the initial condition does not lie within the symbiotic "window" defined by P_1 and P_2, the amount of feed required from the exogeneous source is also of concern. Figure 14 depicts the plutonium feed required for the scenario just discussed [i.e., $P(0) = 0$].

In addition to the characteristic support ratios of various symbiotic systems, the growth potential of breeder-based systems must be addressed. To characterize the growth potential of the symbiotic system, the asymptotic support ratio represents the logical reference point. For a system operating at the asymptotic value of the support ratio, both components grow in proportion to the number operating, i.e.,

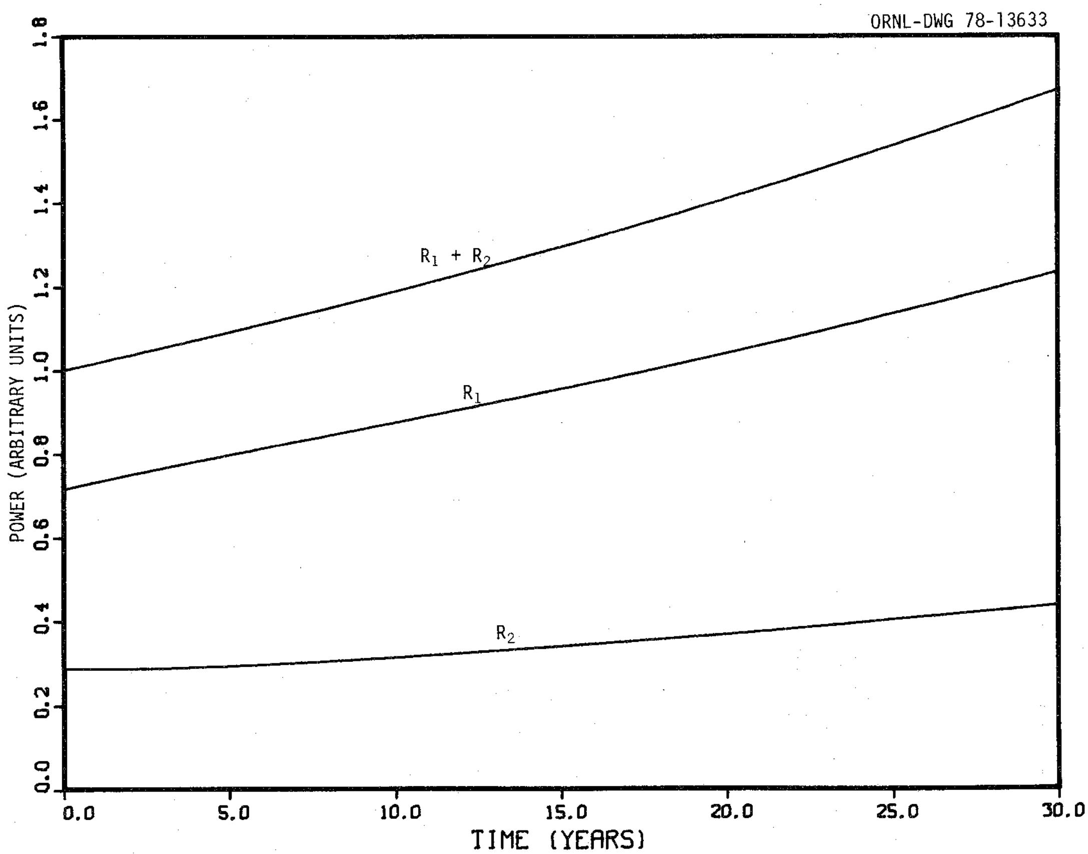

Fig. 10. Power Characteristics for Pu/Th-20% Denatured Symbiotic System; $P(0) = P_1$.

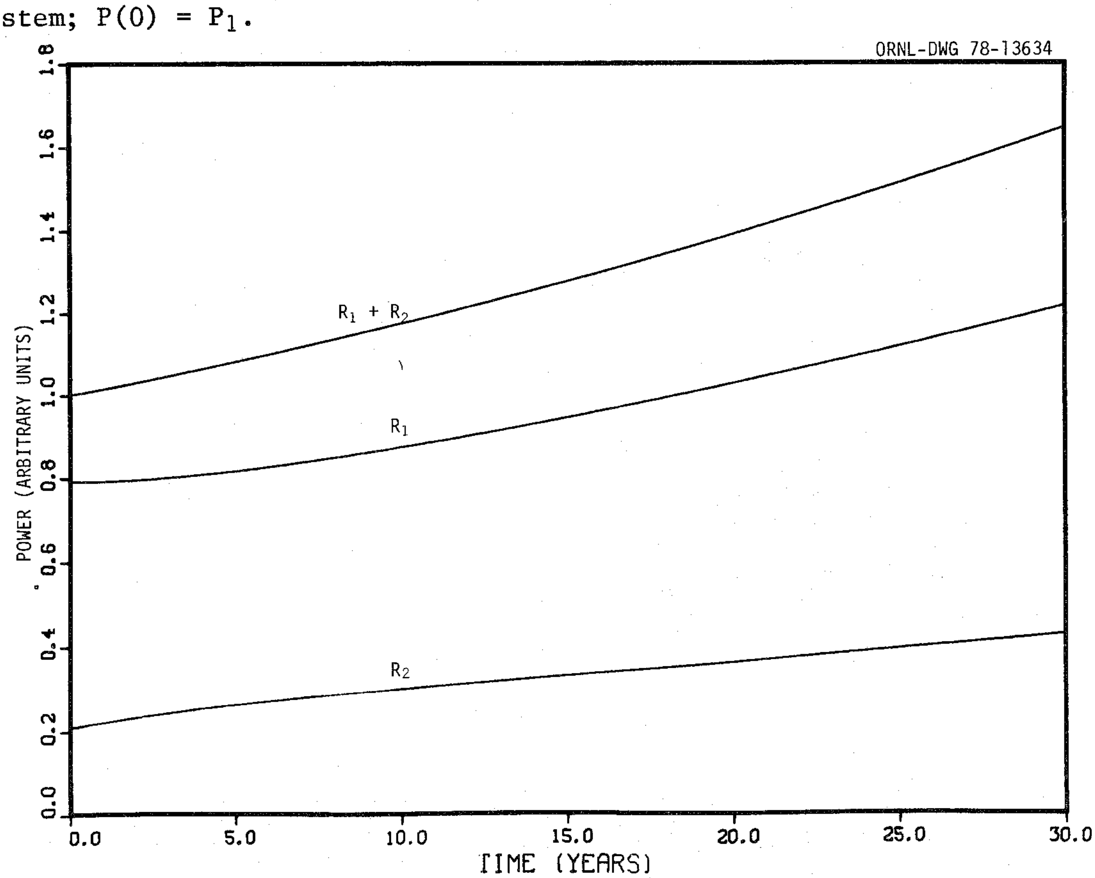

Fig. 11. Power Characteristics of Pu/Th-20% Denatured Symbiotic System; $P(0) = P_2$.

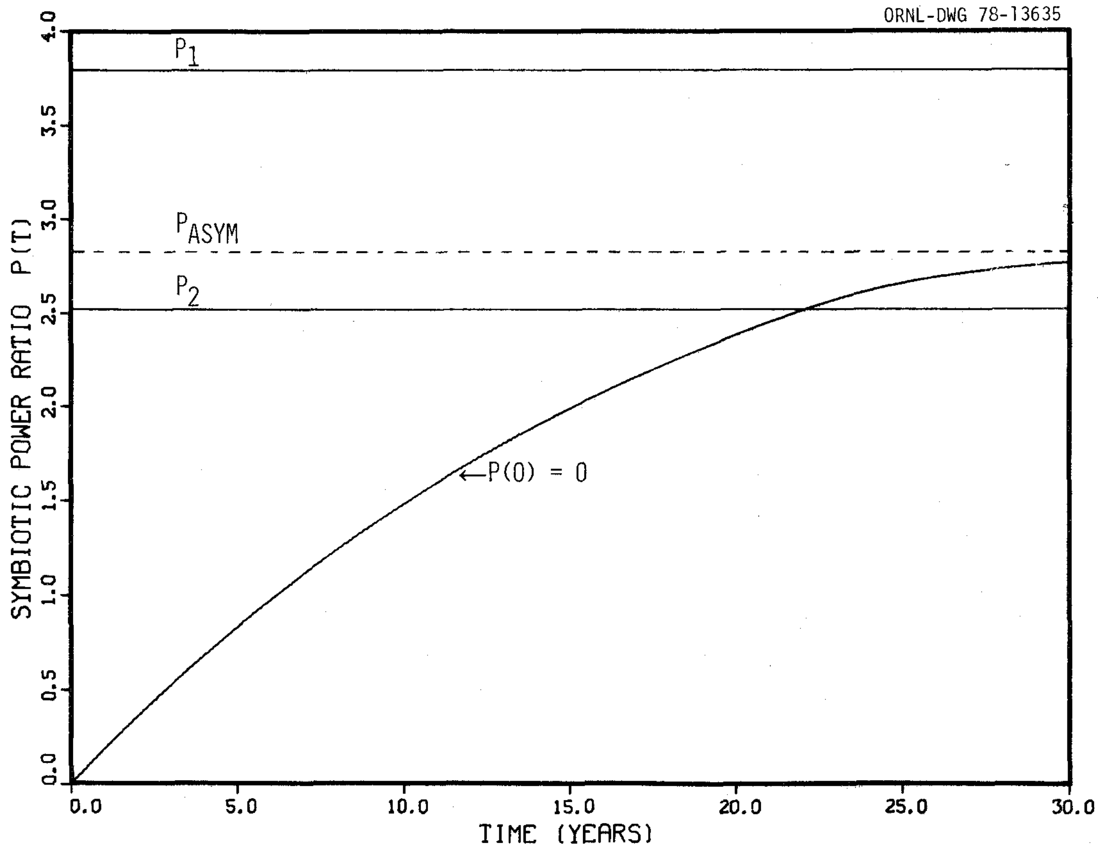

Fig. 12. Power Ratio Behavior of Symbiotic System of Pu/Th Transmuters and 20% Denatured Reactors Assuming an Exogenous Source of Pu.

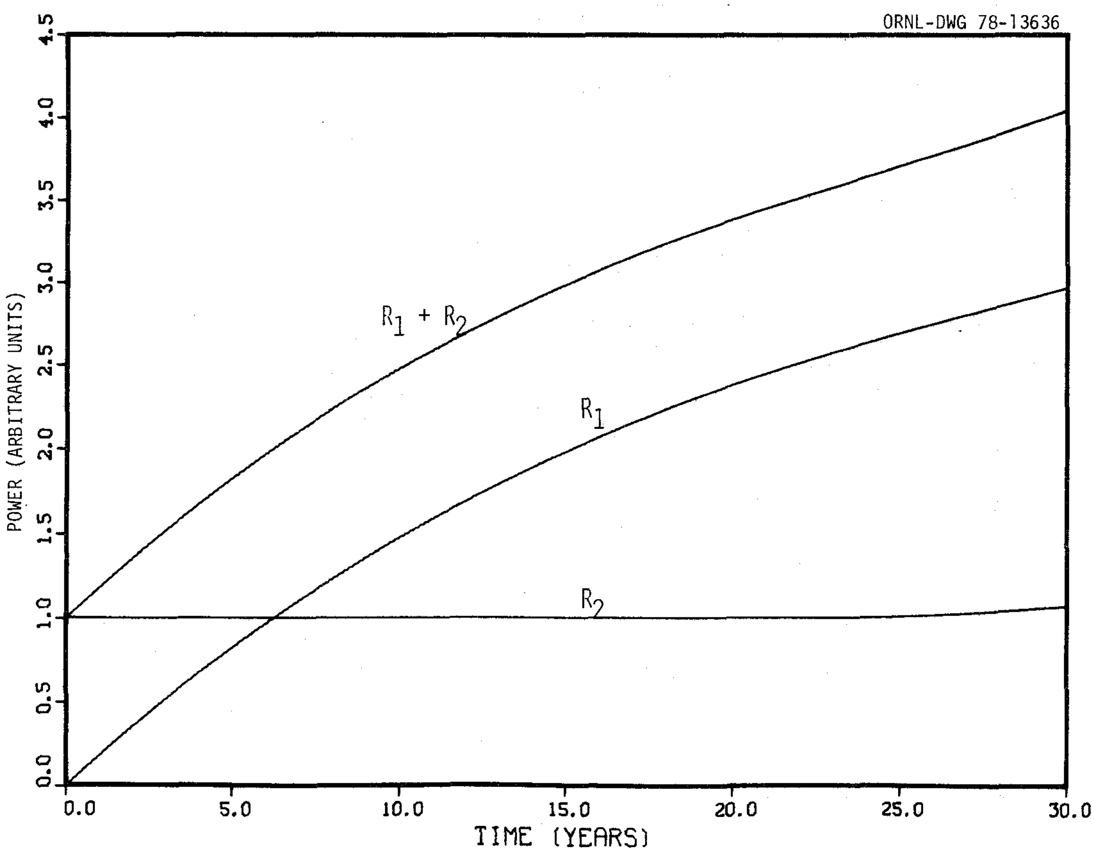

Fig. 13. Power Characteristics of Pu/Th-20% Denatured Symbiotic System Assuming an Exogenous Source of Pu.

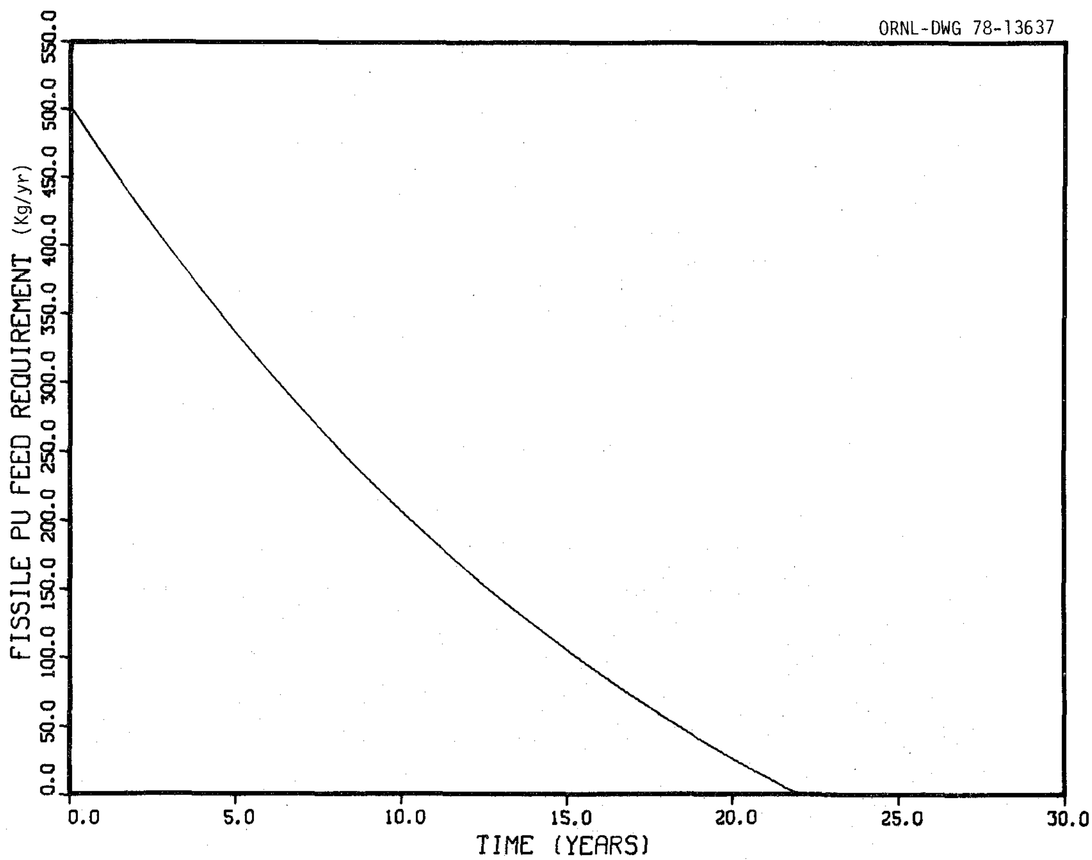

Fig. 14. Fissile Pu Fuel Requirements for Pu/Th-20% Denatured Symbiotic System Initiated with P(0) = 0.0.

$$\frac{R_1'(\infty)}{R_2'(\infty)} = P_\infty \quad . \tag{16}$$

At the asymptotic value of support ratio, the corresponding "asymptotic compound system doubling time" is given by

$$t_D = \frac{\ell n2}{\lambda_1} = \frac{(\ell n2)\ P_\infty}{\left|a_{12} - P_\infty\ a_{11}\right|} = \frac{(\ell n2)}{\left|P_\infty\ a_{21} + a_{22}\right|} \tag{17}$$

Considering reactor charged with a single fissile material (i.e., $c_{ij} = 0$ if $i \neq j$), taking

$$c_{ii} = f_i\ [M_{ii}^{\text{initial core}} + mM_{ii}^{\text{reload}}] \quad , \tag{18}$$

and assuming annual refueling, the doubling time is given by

$$t_D = \frac{(\ell n2)\ P_\infty\ f(M_{11}^{IC} + mM_{11}^{Reload})}{r\cdot(M_{12}^{discharge} + P_\infty\ M_{11}^{discharge}) - f\ P_\infty\ M_{11}^{charge}} \quad (19a)$$

$$= \frac{(\ell n2)\ f(M_{22}^{IC} + mM_{22}^{Reload})}{r(P_\infty\ M_{21}^{discharge} + M_{22}^{discharge}) - f\ M_{22}^{charge}} \quad (19b)$$

where M_{ij} denotes the mass of isotope i associated with reactor j, and m is the lag (in years) assumed for reprocessing and refabrication.

Table 4 lists the symbiotic parameters discussed above for various combinations of denatured/transmuter LMFBRs. Additionally, the corresponding parameters for the reference case Pu/^{238}U fuel are listed for comparison. For the reference case, it has been assumed that no restrictions on reactor location are appropriate, yielding an infinite support ratio. As indicated by the data in Table 4, an obvious tradeoff exists between support ratio and doubling time. Location of nuclear generating stations near load centers in order to minimize transmission losses is an obvious incentive to maximize the power generated in the dispersed systems. Moreover, the reactor types proposed for use in the energy centers represent a higher proliferation risk than do the dispersed reactors. Although this risk can be minimized by adequate controls on the energy center itself, the number of such centers which must be monitored is clearly a concern. However, as indicated by Table 4, maximizing the support ratio also implies large increases in the system doubling times. Such a tradeoff exists due to the dominance of different types of reactors in the various symbiotic systems. At low support ratios, the dominant reactors are the transmuters which are inherently higher breeding performance machines. Hence the doubling time reflects this influence. On the other hand, at the higher support ratios the lower performance denatured systems become the dominant influence, markedly increasing the system doubling time. It should be emphasized that the data presented in Table 4 is for non-optimized, oxide-fueled, homogeneous LMFBRs. Improvement in both the support ratio and the doubling time characteristics should be attainable.

Table 4. Equilibrium-Cycle Symbiotic Parameters for Oxide Fueled Homogeneous LMFBRS (1 year out-of-reactor time, 75% capacity, r = 0.98, f = 1.00)

Energy Center Reactor LMFBR		Support Ratio			CSDT[a]	Energy Growth Rate Supported (%)
		Min.	Asym.	Max.		
$(Pu/U)O_2 + ThO_2$ RB	∿12% denatured	0.00	0.30	0.42	14.5	4.8
$(Pu/U)O_2 + ThO_2$ RB	20% denatured	0.00	0.55	1.00	17.3	4.0
$(Pu/U)O_2 + ThO_2$ RB	40% denatured	0.00	1.00	2.63	23.7	2.9
Pu/Th	∿12% denatured	1.13	1.32	1.57	29.0	2.4
Pu/Th	20% denatured	2.52	2.83	3.79	40.9	1.7
Pu/Th	40% denatured	6.55	6.92	9.93	88.7	0.8
-	(Pu/U)		∞		12.7	5.4

[a]CSDT = Compound System Doubling Time.

The parameters given in Table 4 were calculated by specifying a single reactor type for each component of the symbiosis. A more dramatic illustration of the tradeoff of growth rate versus dispersion ratio can be derived by considering combinations of the various transmuter options. Using $(Pu/U)O_2$, $(Pu/U)O_2 + ThO_2$ radial blankets, and $(Pu/Th)O_2$ transmuters (i.e., cases 1, 2, and 3 in Table 3), the asymptotic "operational envelope" for the symbiotic system (growth potential versus dispersion ratio) can be generated once a particular dispersed system is selected. Figure 15 indicates such an envelope for arbitrary combinations of the three transmuter options coupled with 12% denatured LMFBRs. The points labeled A, B, and C correspond to specification of a single transmuter as the option (cases 1, 2, and 3 respectively); the curves connecting these three points consist of combinations of the two options defined by the end points. Points within the envelope correspond to combinations of all three types in different proportions.

Figures 16 and 17 indicate the analogous operational envelopes for 20% and 40% denatured LMFBRs respectively. Comparison of Figs. 15-17 permits the effects of the denaturing criterion on the symbiosis to be

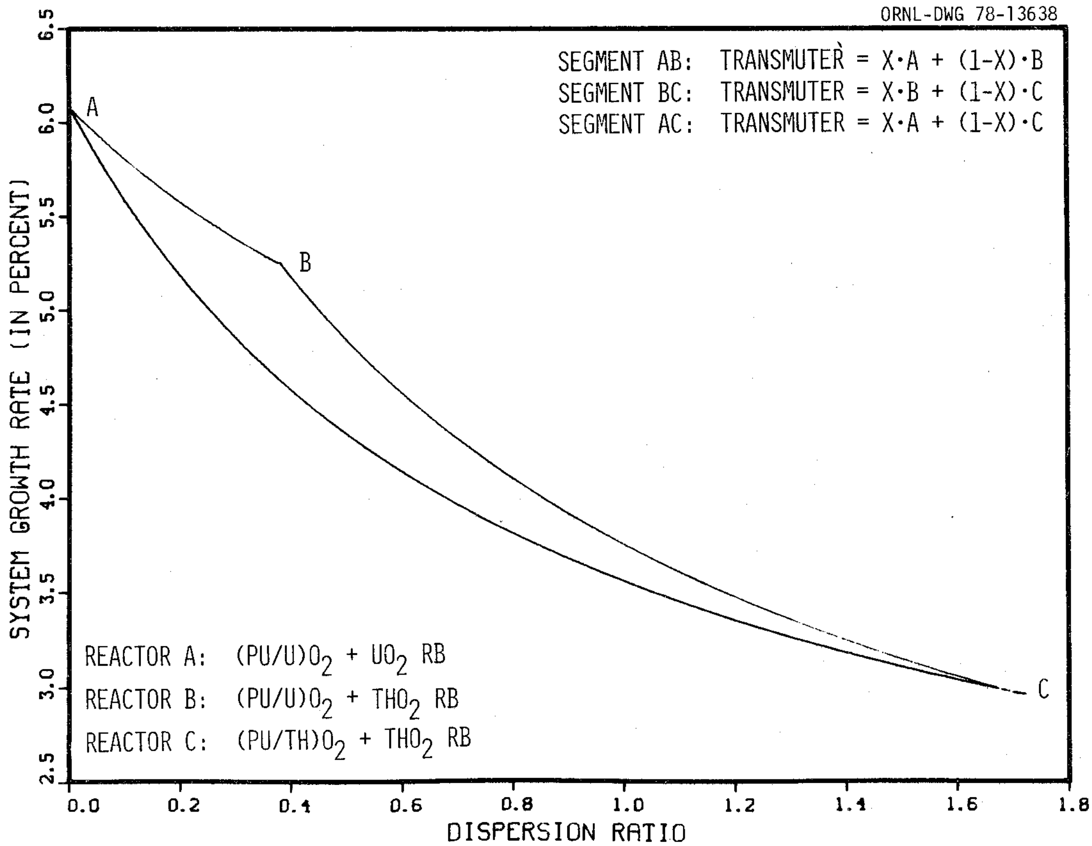

Fig. 15. Operational Envelope for "Mixed" Transmuter Options Coupled with 12% Denatured Fast Breeder Reactors.

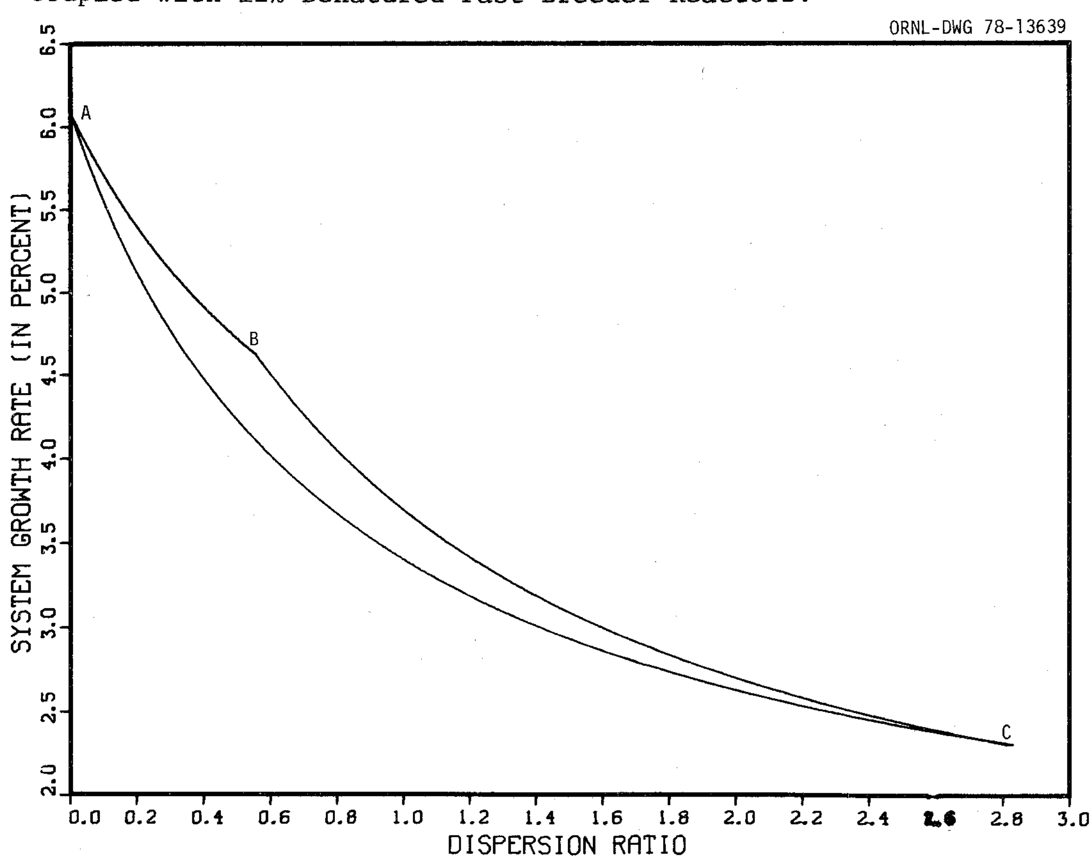

Fig. 16. Operational Envelope for "Mixed" Transmuter Options Coupled with 20% Denatured Fast Breeder Reactors.

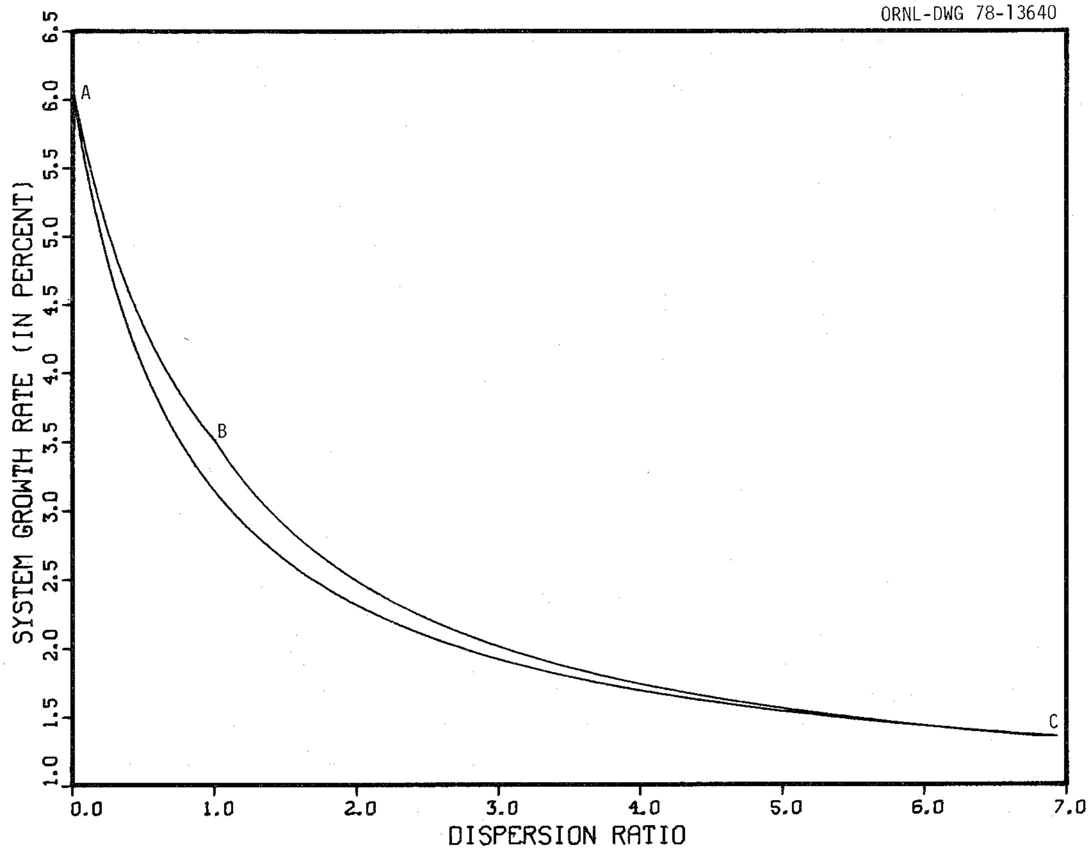

Fig. 17. Operational Envelope for "Mixed" Transmuter Options Coupled with 40% Denatured Fast Breeder Reactors.

addressed. Clearly, increasing the allowable enrichment increases the range of possible dispersion ratios for the system. Increasing the allowable enrichment, however, decreases the growth rate attainable for a specified dispersion ratio. Finally, increasing the allowable enrichment tends to decrease the growth flexibility of the symbiosis by narrowing the permissible range of growth rates for a desired dispersion ratio.

IV. NUCLEAR DATA UNCERTAINTIES

As indicated in Section II, recent measurements of the capture cross section of ^{232}Th [12] have indicated significant discrepancies between measured and tabulated ENDF/B-IV cross sections. In an effort to ascertain the projected impact of such a cross-section change on the LMFBR performance parameters, the original nine-group ^{232}Th capture cross sections were modified to approximate the revised evaluation given in Fig. 18. As indicated by Fig. 11, the revised evaluation is significantly lower in the energy range of interest.

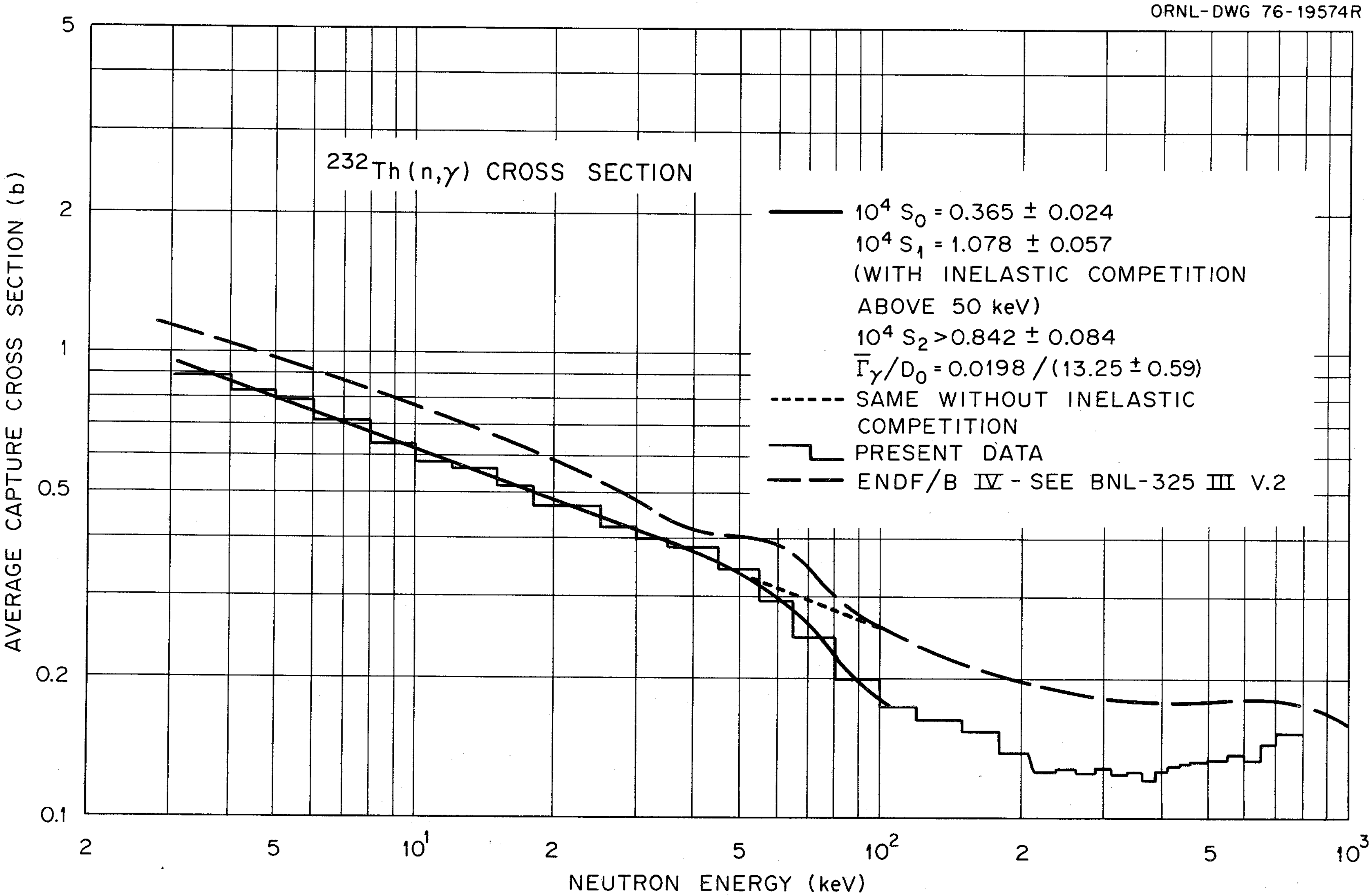

Fig. 18. Comparison of $^{232}Th(n,\gamma)$ Cross Section Values from ENDF/B-IV and Evaluation of Macklin and Halperin.[11]

The results of calculations for two of the alternative fueled LMFBRs, the 20% denatured case and the Pu/Th transmuter, are presented in Table 5. (It should be noted that due to differences in the control pattern employed, minor discrepancies between Tables 3 and 5 exist for corresponding systems.) The revised ^{232}Th capture data results in somewhat smaller fissile loading requirements by both systems. Much more significant, however, is the sharp reduction in breeding gain (breeding ratio minus 1) in both reactors of approximately 50%. Due to the similar percentage change in the breeding gain of both systems, it is not surprising that the support ratio for a symbiotic system consisting of these reactor types changes only slightly. The marked decrease in the breeding gain, however, sharply increases the doubling time for the system by approximately a factor of 3.

It should be emphasized that Table 5 represents the effects of one particular cross-section change. Since the isotopes involved in the ^{233}U/^{232}Th cycle have not received scrutiny comparable to those involved in the Pu/^{238}U cycle, other data adjustments may have compensating effects. Moreover, while the impact of certain nuclear data on the alternative fuel cycles may be unfavorable, it is also important to note that many changes can be accommodated by design modifications. In any case, however, it is apparent that determination of more accurate nuclear data for the ^{232}Th-based fuels is of prime importance.

Table 5. Changes in Nuclear Data Can Significantly Affect the Denatured/Symbiotic Fuel Cycle

	ENDF/B IV	Revised Th Capture
20% denatured		
Initial inventory (kg/GWe)	2192	2050
Equilibrium reload (kg/GWe)	813	796
Breeding ratio	1.096	1.050
Pu/Th "transmuter"		
Initial inventory (kg/GWe)	2517	2280
Equilibrium reload (kg/GWe)	894	865
Breeding ratio	1.158	1.078
Symbiotic power ratio	2.83	2.59
Compound system doubling time (yrs)	33.8	89.7

V. SUMMARY AND CONCLUSIONS

Although the results given in this study must be regarded as preliminary evaluations, certain generic effects and trends concerning the alternative fuel cycles are indicated.

(a) The use of ^{233}U as the fissile material and/or the use of ^{232}Th as the fertile material in oxide-fueled LMFBRs incurs a significant breeding ratio penalty relative to Pu/^{238}U. This penalty can be traced to basic differences in the neutronic behavior of the various isotopes. Hence, although design improvements can potentially improve the alternative fuel performance, it is unlikely that the breeding performance can exceed that of an analogous Pu/^{238}U system.

(b) For the denatured fuels ($^{233}U/^{238}U/^{232}Th$) considered, both the overall breeding ratio and the isotopic composition are related to the degree of denaturing (% ^{233}U in U). In particular, the ^{233}U component of the denatured LMFBR breeding ratio is very sensitive to this parameter, increasing rapidly between 12% and 20% denaturing. Since the ^{233}U component of the breeding ratio is of primary importance in determining the required amount of makeup (and hence the support ratio), there is an obvious incentive from a performance viewpoint to set the allowable enrichment as high as possible.

(c) An enrichment of $\sim$12% corresponds to a minimum enrichment required by criticality for the LMFBR configuration considered in this study.

(d) In considering symbiotic systems of dispersed denatured reactors and energy center "transmuters," a trade-off between support ratio and energy growth rate was found to exist. Although there are incentives for both a high growth rate and a large support ratio, the results indicate that these goals are somewhat mutually exclusive for the reactor types analyzed.

(e) The effect of proposed nuclear data changes was found to significantly impact the viability of the denatured fuel cycle. Clearly the accuracy of the nuclear data for ^{232}Th and ^{233}U requires improvement.

As a final note, it should be emphasized that only one particular model of a reactor was utilized in this study. In order to fully address the denatured fuel cycle, other possible reactor designs and reactor types need to be addressed. For example, preliminary calculations for heterogeneous (i.e., interspersed core and blanket assemblies) indicate that significant improvements in both the support ratio and the system doubling time are possible.

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APPENDIX

Table A-1. Initial Core Loadings for LMFBRs Utilizing Alternate Oxide Fuels (kg)

Case	Region	^{232}Th	^{233}U	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Fissile Pu	Total HM
1	Core			39.8	19558.3	2364.4	678.2	359.0	120.8	2723.4	23120.5
	Axial			28.7	14274.6						14303.3
	Radial			75.7	37683.6						37759.2
	Total			144.1	71516.5	2364.4	678.2	359.0	120.8	2723.4	75182.9
2	Core			39.7	19527.0	2387.0	684.5	362.4	121.9	2749.5	23122.6
	Axial			28.7	14274.6						14303.3
	Radial	35102.7									35102.7
	Total	35102.7		68.4	33801.7	2387.0	684.5	362.4	121.9	2749.5	72528.4
3	Core	17880.0				2828.8	807.7	425.7	143.3	3254.5	22085.5
	Axial	13594.5									13594.5
	Radial	35891.0									35891.0
	Total	67365.5				2828.8	807.7	425.7	143.3	3254.5	71570.9
4	Core		2446.0	41.8	20560.3						23048.2
	Axial			28.7	14274.6						14303.3
	Radial	35891.0									25891.0
	Total	35891.0	2446.0	70.5	34835.0						73242.4
5	Core		2466.8	41.8	20537.5						23046.1
	Axial	13594.5									13594.5
	Radial	35891.0									35891.0
	Total	49485.5	2466.8	41.8	20537.5						72531.6
6	Core	9092.3	2649.0	22.0	10805.0						22568.2
	Axial	13594.5									13594.5
	Radial	35891.0									35891.0
	Total	58577.8	2649.0	22.0	10805.0						72053.7
7	Core	15181.1	2786.6	8.7	4263.8						22240.2
	Axial	13594.5									13594.5
	Radial	35891.0									35891.0
	Total	64666.6	2786.6	8.7	4263.8						71725.6
8	Core	19063.5	2902.8								21966.3
	Axial	13594.5									13594.5
	Radial	35891.0									35891.0
	Total	68548.9	2902.8								71451.8

Table A-2. Equilibrium Cycle Reloadings for LMFBRs Utilizing Alternate Oxide Fuels (kg)

Case	Region	^{232}Th	^{233}U	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	Fissile Pu	Total HM
1	Core			13.1	6446.7	837.8	240.3	127.2	42.8	964.9	7707.7
	Axial			9.6	4758.2						4767.8
	Radial			12.6	6280.6						6293.2
	Total			35.3	17485.5	837.8	240.3	127.2	42.8	964.9	18768.7
2	Core			13.1	6416.7	834.1	245.9	130.2	43.7	964.3	7683.6
	Axial			9.6	4758.2						4767.8
	Radial	5850.5									5850.5
	Total	5850.5		22.6	11174.9	834.1	245.9	130.2	43.7	964.3	18301.8
3	Core	5936.0				960.0	274.1	144.4	48.6	1104.4	7363.1
	Axial	4531.5									4531.5
	Radial	5981.8									5981.8
	Total	16449.4				960.0	274.1	144.4	48.6	1104.4	17876.5
4	Core		953.5	13.7	6712.3						7679.5
	Axial			9.6	4758.2						4767.8
	Radial	5981.8									5981.8
	Total	5981.8	953.5	23.2	11470.5						18429.1
5	Core		961.6	13.6	6703.8						7679.1
	Axial	4531.5									4531.5
	Radial	5981.8									5981.8
	Total	10513.3	961.6	13.6	6703.8						18192.4
6	Core	2441.5	1001.4	7.8	4084.3						7535.0
	Axial	4531.5									4531.5
	Radial	5981.8									5981.8
	Total	12954.8	1001.5	7.8	4084.3						18048.4
7	Core	4757.7	1050.5	3.3	1607.3						7418.8
	Axial	4531.5									4531.5
	Radial	5981.8									5981.8
	Total	15271.1	1050.5	3.3	1607.3						17932.2
8	Core	6230.0	1092.6								7322.6
	Axial	4531.5									4531.5
	Radial	5981.8									5981.8
	Total	16743.4	1092.6								17836.0

Table A-3. Discharge Data for 1200-MWe Pu-U/U LMFBR with Pu/UO_2 Core, UO_2 Axial and Radial Blankets (RZ Model, Two Core Zones)

Discharge Data (kg)

Reactor Core

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PJ	TOTAL HM
1	0.7500	0.0	0.0	9.6	6334.4	746.6	230.6	83.6	38.3	830.2	7443.0
2	0.7500	0.0	0.0	6.8	5911.9	811.0	288.1	79.5	44.7	890.4	7141.9
3	0.7500	0.0	0.0	5.0	5686.8	785.1	301.6	67.7	43.2	852.7	6889.3
4	0.7500	0.0	0.0	5.0	5667.9	793.5	307.1	69.7	44.4	863.3	6887.7
5	0.7500	0.0	0.0	5.1	5671.5	793.8	306.8	69.8	44.4	863.6	6891.4
6	0.7500	0.0	0.0	5.1	5675.9	793.9	306.3	69.8	44.4	863.7	6895.4
7	0.7500	0.0	0.0	5.1	5677.3	793.9	306.2	69.9	44.4	863.8	6896.7
8	0.7500	0.0	0.0	5.1	5677.8	793.9	306.1	69.9	44.4	863.8	6897.3
9	0.7500	0.0	0.0	5.1	5677.9	793.9	306.1	69.9	44.4	863.8	6897.4
10	0.7500	0.0	0.0	5.1	5678.0	793.9	306.1	69.9	44.4	863.8	6897.5
11	0.7500	0.0	0.0	5.1	5678.1	793.9	306.1	69.9	44.4	863.8	6897.5
12	0.7500	0.0	0.0	5.1	5678.1	793.9	306.1	69.9	44.4	863.8	6897.5
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	0.0	0.0	21.6	17780.6	2439.6	859.0	248.9	133.9	2687.5	21482.6

Axial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	0.0	0.0	8.6	4701.5	51.5	0.8	0.0	0.0	51.5	4762.4
2	0.7500	0.0	0.0	7.7	4645.5	97.5	3.0	0.1	0.0	97.5	4753.6
3	0.7500	0.0	0.0	6.9	4589.4	138.9	5.3	0.2	0.0	139.1	4741.8
4	0.7500	0.0	0.0	6.9	4589.2	138.8	5.3	0.2	0.0	139.0	4741.3
5	0.7500	0.0	0.0	6.9	4589.2	138.7	6.3	0.2	0.0	138.9	4741.3
6	0.7500	0.0	0.0	6.9	4590.0	138.2	6.2	0.2	0.0	138.4	4741.5
7	0.7500	0.0	0.0	6.9	4590.3	138.0	6.2	0.2	0.0	138.1	4741.5
8	0.7500	0.0	0.0	6.9	4590.5	137.9	6.2	0.2	0.0	138.0	4741.6
9	0.7500	0.0	0.0	6.9	4590.5	137.8	6.2	0.2	0.0	138.0	4741.6
10	0.7500	0.0	0.0	6.9	4590.5	137.8	6.2	0.2	0.0	138.0	4741.6
11	0.7500	0.0	0.0	6.9	4590.5	137.8	6.2	0.2	0.0	138.0	4741.6
12	0.7500	0.0	0.0	6.9	4590.5	137.8	6.2	0.2	0.0	138.0	4741.6
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	0.0	0.0	23.2	13937.9	285.8	9.8	0.2	0.0	285.0	14256.9

Radial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	0.0	0.0	11.9	6242.6	35.4	0.3	0.0	0.0	35.4	6290.2
2	0.7500	0.0	0.0	11.2	6205.6	68.1	1.1	0.0	0.0	68.1	6286.0
3	0.7500	0.0	0.0	10.6	6168.7	98.9	2.4	0.0	0.0	99.0	6280.7
4	0.7500	0.0	0.0	10.0	6131.1	128.8	4.1	0.1	0.0	128.8	6274.1
5	0.7500	0.0	0.0	9.5	6093.3	157.2	6.2	0.2	0.0	157.4	6266.4
6	0.7500	0.0	0.0	8.9	6055.5	184.2	8.8	0.3	0.0	184.5	6257.7
7	0.7500	0.0	0.0	8.9	6054.6	184.6	8.8	0.2	0.0	184.9	6257.1
8	0.7500	0.0	0.0	8.9	6052.8	185.7	8.8	0.3	0.0	185.9	6256.5
9	0.7500	0.0	0.0	8.9	6051.1	186.7	8.9	0.3	0.0	187.0	6255.9
10	0.7500	0.0	0.0	8.9	6050.4	187.2	9.0	0.3	0.0	187.4	6255.7
11	0.7500	0.0	0.0	8.9	6050.1	187.3	9.0	0.3	0.0	187.6	6255.6
12	0.7500	0.0	0.0	8.9	6050.0	187.4	9.0	0.3	0.0	187.6	6255.5
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	0.0	0.0	62.1	36872.1	687.8	23.5	0.5	0.0	688.3	37645.9

Entire Reactor

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	0.0	0.0	30.1	17278.5	833.6	231.7	83.6	38.3	917.1	18495.6
2	0.7500	0.0	0.0	25.7	16763.0	976.5	292.1	79.5	44.7	1055.0	18181.5
3	0.7500	0.0	0.0	22.5	16444.9	1022.9	310.3	67.9	43.2	1090.8	17911.7
4	0.7500	0.0	0.0	22.0	16388.2	1061.1	317.4	70.0	44.4	1131.1	17903.2
5	0.7500	0.0	0.0	21.4	16354.0	1089.8	319.3	70.1	44.4	1159.9	17899.1
6	0.7500	0.0	0.0	20.9	16321.4	1116.3	321.3	70.3	44.4	1186.5	17894.6
7	0.7500	0.0	0.0	21.0	16322.2	1116.5	321.1	70.3	44.4	1186.8	17895.5
8	0.7500	0.0	0.0	21.0	16321.1	1117.5	321.2	70.3	44.4	1187.7	17895.4
9	0.7500	0.0	0.0	20.9	16319.6	1118.5	321.2	70.3	44.4	1188.8	17894.9
10	0.7500	0.0	0.0	20.9	16318.9	1118.9	321.3	70.3	44.4	1189.2	17894.8
11	0.7500	0.0	0.0	20.9	16318.7	1119.1	321.3	70.3	44.4	1189.4	17894.7
12	0.7500	0.0	0.0	20.9	16318.6	1119.1	321.3	70.3	44.4	1189.4	17894.7
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	0.0	0.0	106.9	68590.6	3412.1	892.2	249.7	133.9	3661.8	73385.3

Table A-4. Discharge Data for 1200-MWe Pu-U/U-Th LMFBR with Pu/UO_2 Core, UO_2 Axial Blankets, and ThO_2 Radial Blanket (RZ Model, Two Core Zones)

Discharge Data (kg)

Reactor Core

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	0.0	0.0	9.6	6307.8	759.3	235.9	85.6	39.2	844.9	7437.4
2	0.7500	0.0	0.0	6.8	5926.6	804.8	284.0	78.1	43.9	882.9	7144.1
3	0.7500	0.0	0.0	4.9	5656.3	791.0	307.5	69.1	44.1	860.1	6873.0
4	0.7500	0.0	0.0	5.0	5634.5	791.3	310.8	71.0	45.3	862.2	6857.8
5	0.7500	0.0	0.0	5.0	5639.0	791.4	310.5	71.0	45.3	862.4	6862.2
6	0.7500	0.0	0.0	5.0	5643.1	791.3	310.1	71.1	45.3	862.3	6865.9
7	0.7500	0.0	0.0	5.0	5645.0	791.4	310.0	71.1	45.3	862.5	6867.8
8	0.7500	0.0	0.0	5.0	5645.9	791.4	309.9	71.1	45.3	862.5	6868.6
9	0.7500	0.0	0.0	5.0	5646.1	791.4	309.9	71.1	45.3	862.5	6868.9
10	0.7500	0.0	0.0	5.0	5646.3	791.4	309.9	71.1	45.3	862.5	6869.1
11	0.7500	0.0	0.0	5.0	5646.4	791.4	309.9	71.1	45.3	862.5	6869.1
12	0.7500	0.0	0.0	5.0	5646.4	791.4	309.9	71.1	45.3	862.5	6869.2
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	0.0	0.0	21.4	17687.1	2430.6	871.8	254.0	136.7	2684.6	21401.6

Axial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	0.0	0.0	8.6	4701.2	51.8	0.8	0.0	0.0	51.8	4762.3
2	0.7500	0.0	0.0	7.7	4644.6	98.0	3.0	0.1	0.0	98.1	4753.4
3	0.7500	0.0	0.0	6.9	4588.4	139.6	6.4	0.2	0.0	139.7	4741.4
4	0.7500	0.0	0.0	6.9	4588.2	139.4	6.4	0.2	0.0	139.6	4741.1
5	0.7500	0.0	0.0	6.9	4588.4	139.2	6.3	0.2	0.0	139.4	4741.1
6	0.7500	0.0	0.0	6.9	4589.3	138.7	6.3	0.2	0.0	138.8	4741.3
7	0.7500	0.0	0.0	6.9	4589.7	138.4	6.2	0.2	0.0	138.5	4741.4
8	0.7500	0.0	0.0	6.9	4589.9	138.2	6.2	0.2	0.0	138.4	4741.4
9	0.7500	0.0	0.0	6.9	4590.0	138.2	6.2	0.2	0.0	138.3	4741.4
10	0.7500	0.0	0.0	6.9	4590.0	138.1	6.2	0.2	0.0	138.3	4741.5
11	0.7500	0.0	0.0	6.9	4590.0	138.1	6.2	0.2	0.0	138.3	4741.5
12	0.7500	0.0	0.0	6.9	4590.0	138.1	6.2	0.2	0.0	138.3	4741.5
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	0.0	0.0	23.2	13936.9	286.4	9.9	0.2	0.0	286.7	14256.7

Radial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5815.8	33.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5849.3
2	0.7500	5781.3	65.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5846.4
3	0.7500	5746.2	95.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5841.6
4	0.7500	5710.0	124.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5834.9
5	0.7500	5673.7	152.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5826.4
6	0.7500	5637.3	179.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5816.3
7	0.7500	5634.5	180.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5815.4
8	0.7500	5631.7	182.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5814.5
9	0.7500	5629.6	184.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5813.8
10	0.7500	5628.8	184.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5813.6
11	0.7500	5628.4	185.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5813.4
12	0.7500	5628.3	185.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5813.4
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	34320.7	684.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	35004.7

Entire Reactor

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5815.8	33.5	18.1	11009.0	811.1	236.7	85.6	39.2	896.7	18049.0
2	0.7500	5781.3	65.1	14.5	10571.2	902.9	287.0	78.2	43.9	981.0	17743.9
3	0.7500	5746.2	95.4	11.8	10244.6	930.6	313.9	69.3	44.1	999.9	17456.0
4	0.7500	5710.0	124.8	11.9	10222.7	930.7	317.2	71.1	45.3	1001.8	17433.7
5	0.7500	5673.7	152.7	11.9	10227.4	930.6	316.8	71.2	45.3	1001.7	17429.6
6	0.7500	5637.3	179.0	11.9	10232.4	930.0	316.4	71.2	45.3	1001.2	17423.5
7	0.7500	5634.5	180.9	12.0	10234.7	929.7	316.2	71.3	45.3	1001.0	17424.6
8	0.7500	5631.7	182.8	12.0	10235.8	929.6	316.1	71.3	45.3	1000.9	17424.6
9	0.7500	5629.6	184.2	12.0	10236.1	929.6	316.1	71.3	45.3	1000.9	17424.2
10	0.7500	5628.8	184.8	12.0	10236.3	929.6	316.1	71.3	45.3	1000.8	17424.1
11	0.7500	5628.4	185.0	12.0	10236.4	929.5	316.1	71.3	45.3	1000.8	17424.0
12	0.7500	5628.3	185.1	12.0	10236.4	929.5	316.1	71.3	45.3	1000.8	17424.0
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	34320.7	684.0	44.6	31624.0	2717.1	881.7	254.2	136.7	2971.3	70662.9

Table A-5. Discharge Data for 1200-MWe Pu-Th/Th LMFBR with $Pu/ThUO_2$ Core, ThO_2 Axial and Radial Blankets (RZ Model, Two Core Zones)

Discharge Data (kg)

Reactor Core

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5741.6	198.7	0.0	0.0	694.7	281.5	107.4	48.4	802.1	7072.4
2	0.7500	5471.6	319.0	0.0	0.0	548.5	295.4	90.4	50.2	639.0	6775.3
3	0.7500	5256.3	395.1	0.0	0.0	420.5	290.8	76.4	49.4	496.9	6488.6
4	0.7500	5258.6	390.8	0.0	0.0	431.0	293.1	77.6	49.8	508.5	6500.9
5	0.7500	5262.7	389.7	0.0	0.0	433.1	293.3	77.8	49.9	510.8	6506.4
6	0.7500	5265.8	389.0	0.0	0.0	434.3	293.4	77.8	49.9	512.1	6510.1
7	0.7500	5267.1	388.7	0.0	0.0	434.8	293.4	77.9	49.9	512.7	6511.7
8	0.7500	5267.6	388.6	0.0	0.0	435.0	293.4	77.9	49.9	512.9	6512.4
9	0.7500	5267.7	388.6	0.0	0.0	435.0	293.4	77.9	49.9	512.9	6512.5
10	0.7500	5267.8	388.5	0.0	0.0	435.1	293.4	77.9	49.9	513.0	6512.6
11	0.7500	5267.8	388.5	0.0	0.0	435.1	293.4	77.9	49.9	513.0	6512.6
12	0.7500	5267.9	388.5	0.0	0.0	435.1	293.4	77.9	49.9	513.0	6512.7
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	16453.4	886.7	0.0	0.0	1736.0	882.4	283.2	150.8	2019.2	20392.4

Axial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	4477.2	50.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4528.1
2	0.7500	4423.4	95.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4519.1
3	0.7500	4370.0	135.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4505.2
4	0.7500	4369.9	135.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4505.0
5	0.7500	4370.1	135.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4505.1
6	0.7500	4370.7	134.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4505.3
7	0.7500	4371.0	134.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4505.4
8	0.7500	4371.1	134.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4505.4
9	0.7500	4371.2	134.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4505.5
10	0.7500	4371.2	134.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4505.5
11	0.7500	4371.2	134.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4505.5
12	0.7500	4371.2	134.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4505.5
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	13272.6	280.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	13552.6

Radial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5946.2	34.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5980.6
2	0.7500	5912.7	65.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5977.8
3	0.7500	5879.3	94.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5973.3
4	0.7500	5845.3	121.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5967.3
5	0.7500	5811.1	148.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5959.6
6	0.7500	5776.8	173.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5950.6
7	0.7500	5777.1	173.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5950.6
8	0.7500	5775.6	174.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5950.1
9	0.7500	5774.1	175.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5949.7
10	0.7500	5773.4	176.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5949.4
11	0.7500	5773.1	176.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5949.4
12	0.7500	5773.1	176.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5949.3
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	35156.5	648.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	35805.0

Entire Reactor

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	16165.1	284.0	0.0	0.0	694.7	281.5	107.4	48.4	802.1	17581.2
2	0.7500	15807.8	479.8	0.0	0.0	548.5	295.4	90.4	50.2	639.0	17272.2
3	0.7500	15505.6	624.4	0.0	0.0	420.5	290.8	76.4	49.4	496.9	16967.1
4	0.7500	15473.7	648.0	0.0	0.0	431.0	293.1	77.6	49.8	508.5	16973.2
5	0.7500	15443.8	673.3	0.0	0.0	433.1	293.3	77.8	49.9	510.8	16971.1
6	0.7500	15413.3	697.4	0.0	0.0	434.3	293.4	77.8	49.9	512.1	16966.0
7	0.7500	15415.2	696.6	0.0	0.0	434.8	293.4	77.9	49.9	512.7	16967.7
8	0.7500	15414.3	697.4	0.0	0.0	435.0	293.4	77.9	49.9	512.9	16967.9
9	0.7500	15413.0	698.4	0.0	0.0	435.0	293.4	77.9	49.9	512.9	16967.6
10	0.7500	15412.4	698.9	0.0	0.0	435.1	293.4	77.9	49.9	513.0	16967.5
11	0.7500	15412.1	699.0	0.0	0.0	435.1	293.4	77.9	49.9	513.0	16967.4
12	0.7500	15412.1	699.1	0.0	0.0	435.1	293.4	77.9	49.9	513.0	16967.5
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	64882.5	1815.3	0.0	0.0	1736.0	882.4	283.2	150.8	2019.2	69749.9

Table A-6. Discharge Data for 1200-MWe U/U-Th Minimum Denatured LMFBR UO_2 Core, UO_2 Axial and Radial Blankets (RZ Model, Two Core Zones)

Discharge Data (kg)

Reactor Core

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	0.0	454.5	10.7	6763.9	201.5	6.4	0.1	0.0	201.6	7437.0
2	0.7500	0.0	439.3	7.6	6306.5	330.7	20.3	0.7	0.0	331.4	7105.4
3	0.7500	0.0	347.2	5.9	5998.0	422.9	39.0	2.0	0.1	424.9	6815.1
4	0.7500	0.0	354.4	5.9	5998.1	418.9	37.9	1.9	0.1	420.8	6817.2
5	0.7500	0.0	355.4	5.9	6000.3	418.2	37.7	1.9	0.1	420.1	6819.5
6	0.7500	0.0	356.8	6.0	6003.9	416.9	37.3	1.9	0.1	418.8	6822.8
7	0.7500	0.0	357.6	6.0	6005.6	416.3	37.1	1.9	0.1	418.1	6824.5
8	0.7500	0.0	357.9	6.0	6006.4	416.0	37.1	1.8	0.1	417.8	6825.3
9	0.7500	0.0	358.0	6.0	6006.7	415.9	37.0	1.8	0.1	417.7	6825.5
10	0.7500	0.0	358.1	6.0	6006.8	415.8	37.0	1.8	0.1	417.7	6825.6
11	0.7500	0.0	358.1	6.0	6006.8	415.8	37.0	1.8	0.1	417.7	6825.7
12	0.7500	0.0	358.1	6.0	6006.9	415.8	37.0	1.8	0.1	417.7	6825.7
					YEARS 13 THRU 29 SAME AS ABOVE						
30	0.7500	0.0	1539.4	24.2	18708.2	919.0	61.2	2.6	0.1	921.6	21254.7

Axial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	0.0	0.0	8.7	4707.2	46.5	0.6	0.0	0.0	46.5	4763.0
2	0.7500	0.0	0.0	7.9	4656.7	88.4	2.4	0.0	0.0	88.5	4755.4
3	0.7500	0.0	0.0	7.1	4606.1	126.7	5.2	0.1	0.0	126.9	4745.2
4	0.7500	0.0	0.0	7.1	4605.8	126.8	5.1	0.1	0.0	126.9	4744.9
5	0.7500	0.0	0.0	7.1	4605.6	126.8	5.1	0.1	0.0	126.9	4744.8
6	0.7500	0.0	0.0	7.1	4606.3	126.3	5.1	0.1	0.0	126.5	4745.0
7	0.7500	0.0	0.0	7.2	4606.6	126.1	5.1	0.1	0.0	126.2	4745.0
8	0.7500	0.0	0.0	7.2	4606.7	126.0	5.1	0.1	0.0	126.1	4745.1
9	0.7500	0.0	0.0	7.2	4606.8	126.0	5.1	0.1	0.0	126.1	4745.1
10	0.7500	0.0	0.0	7.2	4606.8	126.0	5.1	0.1	0.0	126.1	4745.1
11	0.7500	0.0	0.0	7.2	4606.8	126.0	5.1	0.1	0.0	126.1	4745.1
12	0.7500	0.0	0.0	7.2	4606.8	126.0	5.1	0.1	0.0	126.1	4745.1
					YEARS 13 THRU 29 SAME AS ABOVE						
30	0.7500	0.0	0.0	23.7	13970.7	260.2	8.0	0.2	0.0	260.4	14262.8

Radial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5950.8	30.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5980.9
2	0.7500	5919.0	59.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5973.5
3	0.7500	5886.8	87.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5974.5
4	0.7500	5854.2	114.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5969.1
5	0.7500	5821.4	140.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5962.2
6	0.7500	5788.5	165.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5953.9
7	0.7500	5785.8	167.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5953.1
8	0.7500	5784.0	168.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.6
9	0.7500	5782.7	169.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.2
10	0.7500	5782.1	169.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.0
11	0.7500	5781.9	170.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5951.9
12	0.7500	5781.8	170.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5951.9
					YEARS 13 THRU 29 SAME AS ABOVE						
30	0.7500	35187.3	624.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	35811.9

Entire Reactor

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5950.8	484.7	19.4	11471.0	248.0	7.0	0.1	0.0	248.1	18180.9
2	0.7500	5919.0	498.8	15.7	10963.1	419.1	22.7	0.8	0.0	419.9	17839.3
3	0.7500	5886.8	434.9	13.0	10604.1	549.6	44.2	2.1	0.1	551.8	17534.8
4	0.7500	5854.2	469.2	13.0	10603.8	545.7	43.1	2.0	0.1	547.7	17531.2
5	0.7500	5821.4	496.2	13.1	10605.9	545.0	42.8	2.0	0.1	547.0	17526.5
6	0.7500	5788.5	522.2	13.1	10610.1	543.2	42.4	2.0	0.1	545.2	17521.7
7	0.7500	5785.8	524.9	13.1	10612.2	542.4	42.2	2.0	0.1	544.4	17522.7
8	0.7500	5784.0	526.5	13.1	10613.1	542.0	42.1	2.0	0.1	544.0	17522.9
9	0.7500	5782.7	527.5	13.1	10613.4	541.9	42.1	2.0	0.1	543.8	17522.8
10	0.7500	5782.1	528.0	13.1	10613.6	541.8	42.1	2.0	0.1	543.8	17522.7
11	0.7500	5781.8	528.2	13.1	10613.6	541.8	42.1	2.0	0.1	543.7	17522.7
12	0.7500	5781.8	528.3	13.1	10613.7	541.8	42.1	2.0	0.1	543.7	17522.7
					YEARS 13 THRU 29 SAME AS ABOVE						
30	0.7500	35187.3	2164.1	47.9	32679.0	1179.2	69.2	2.8	0.1	1182.0	71329.3

Table A-7. Discharge Data for 1200-MWe U/Th Minimum Denatured LMFBR with UO_2 Core, ThO_2 Axial and Radial Blankets (RZ Model, Two Core Zones)

Discharge Data (kg)

Reactor Core

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	0.0	457.9	10.7	6758.6	201.1	6.3	0.1	0.0	201.3	7434.8
2	0.7500	0.0	443.3	7.8	6299.3	329.8	20.2	0.7	0.0	330.5	7101.2
3	0.7500	0.0	350.6	5.9	5990.6	421.5	38.7	2.0	0.1	423.5	6809.3
4	0.7500	0.0	358.1	5.9	5992.1	417.3	37.6	1.9	0.1	419.2	6813.0
5	0.7500	0.0	359.3	5.9	5994.6	416.5	37.3	1.9	0.1	418.4	6815.6
6	0.7500	0.0	360.8	6.0	5998.2	415.2	37.0	1.8	0.1	417.1	6819.0
7	0.7500	0.0	361.5	6.0	5999.9	414.6	36.8	1.8	0.1	416.4	6820.7
8	0.7500	0.0	361.8	6.0	6000.7	414.3	36.7	1.8	0.1	416.1	6821.4
9	0.7500	0.0	361.9	6.0	6000.9	414.2	36.7	1.8	0.1	416.0	6821.6
10	0.7500	0.0	362.0	6.0	6001.0	414.2	36.7	1.8	0.1	416.0	6821.7
11	0.7500	0.0	362.0	6.0	6001.1	414.2	36.6	1.8	0.1	416.0	6821.8
12	0.7500	0.0	362.0	6.0	6001.1	414.2	36.6	1.8	0.1	416.0	6821.8
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	0.0	1554.8	24.2	18688.4	915.0	60.6	2.6	0.1	917.6	21245.7

Axial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	4482.4	46.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4528.7
2	0.7500	4433.8	87.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4521.3
3	0.7500	4384.9	124.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.7
4	0.7500	4384.4	125.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.5
5	0.7500	4384.2	125.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.5
6	0.7500	4384.8	124.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.7
7	0.7500	4385.1	124.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.7
8	0.7500	4385.3	124.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.8
9	0.7500	4385.3	124.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.8
10	0.7500	4385.3	124.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.8
11	0.7500	4385.3	124.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.8
12	0.7500	4385.3	124.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.8
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	13301.1	258.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	13559.7

Radial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5950.8	30.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5980.9
2	0.7500	5919.0	59.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5978.5
3	0.7500	5886.9	87.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5974.5
4	0.7500	5854.4	114.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5969.1
5	0.7500	5821.6	140.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5962.2
6	0.7500	5788.8	165.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5954.0
7	0.7500	5786.2	167.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5953.2
8	0.7500	5784.4	168.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.7
9	0.7500	5783.1	169.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.3
10	0.7500	5782.5	169.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.1
11	0.7500	5782.2	169.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.0
12	0.7500	5782.1	169.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.0
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	35188.7	623.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	35812.2

Entire Reactor

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	10433.2	534.3	10.7	6758.6	201.1	6.3	0.1	0.0	201.3	17944.4
2	0.7500	10352.8	590.3	7.8	6299.3	329.8	20.2	0.7	0.0	330.5	17601.0
3	0.7500	10271.9	563.0	5.9	5990.6	421.5	38.7	2.0	0.1	423.5	17293.6
4	0.7500	10238.8	598.0	5.9	5992.1	417.3	37.6	1.9	0.1	419.2	17291.6
5	0.7500	10205.8	625.1	5.9	5994.6	416.5	37.3	1.9	0.1	418.4	17287.3
6	0.7500	10173.6	650.8	6.0	5998.2	415.2	37.0	1.8	0.1	417.1	17282.6
7	0.7500	10171.3	653.1	6.0	5999.9	414.6	36.8	1.8	0.1	416.4	17283.6
8	0.7500	10169.7	654.6	6.0	6000.7	414.3	36.7	1.8	0.1	416.1	17283.9
9	0.7500	10168.4	655.6	6.0	6000.9	414.2	36.7	1.8	0.1	416.0	17283.7
10	0.7500	10167.8	656.1	6.0	6001.0	414.2	36.7	1.8	0.1	416.0	17283.7
11	0.7500	10167.6	656.3	6.0	6001.1	414.2	36.6	1.8	0.1	416.0	17283.6
12	0.7500	10167.5	656.4	6.0	6001.1	414.2	36.6	1.8	0.1	416.0	17283.7
YEARS 13 THRU 29 SAME AS ABOVE											
30	0.7500	48489.8	2436.9	24.2	18688.4	915.0	60.6	2.6	0.1	917.6	70617.4

Table A-8. Discharge Data for 1200-MWe U-Th/Th 20% Denatured LMFBR with U/ThO$_2$ Core, ThO$_2$ Axial and Radial Blankets (RZ Model, Two Core Zones)

Discharge Data (kg)

Reactor Core

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	3404.8	668.8	4.9	3068.8	86.5	2.5	0.0	0.0	86.5	7236.3
2	0.7500	2677.2	620.5	4.4	3452.4	173.0	10.0	0.3	0.0	173.3	6937.8
3	0.7500	2365.3	532.3	3.5	3504.2	237.0	20.5	1.0	0.0	238.0	6663.9
4	0.7500	2174.6	538.5	3.5	3668.5	245.7	20.9	1.0	0.0	246.7	6652.6
5	0.7500	2176.9	540.1	3.5	3671.3	244.6	20.6	1.0	0.0	245.6	6657.9
6	0.7500	2179.1	541.5	3.5	3673.8	243.7	20.3	1.0	0.0	244.6	6662.9
7	0.7500	2179.8	542.1	3.5	3674.7	243.3	20.2	0.9	0.0	244.3	6664.7
8	0.7500	2180.1	542.4	3.5	3675.1	243.1	20.2	0.9	0.0	244.1	6665.4
9	0.7500	2180.2	542.4	3.5	3675.3	243.1	20.2	0.9	0.0	244.0	6665.7
10	0.7500	2180.3	542.5	3.5	3675.3	243.1	20.1	0.9	0.0	244.0	6665.8
11	0.7500	2180.3	542.5	3.5	3675.4	243.1	20.1	0.9	0.0	244.0	6665.9
12	0.7500	2180.3	542.5	3.5	3675.4	243.0	20.1	0.9	0.0	244.0	6665.9
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	6795.4	1997.2	14.1	11425.3	534.4	33.2	1.3	0.0	535.7	20801.0

Axial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	4482.6	46.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4528.7
2	0.7500	4433.7	87.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4521.3
3	0.7500	4384.9	124.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.7
4	0.7500	4384.5	125.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.5
5	0.7500	4384.8	124.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.6
6	0.7500	4385.6	124.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.9
7	0.7500	4386.0	124.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
8	0.7500	4386.1	123.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
9	0.7500	4386.2	123.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
10	0.7500	4386.1	123.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
11	0.7500	4386.2	123.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
12	0.7500	4386.2	123.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	13302.8	257.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	13560.0

Radial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5950.0	30.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5980.8
2	0.7500	5918.1	60.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5978.3
3	0.7500	5885.9	88.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5974.3
4	0.7500	5853.1	115.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5968.8
5	0.7500	5820.1	141.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5961.8
6	0.7500	5787.0	166.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5953.4
7	0.7500	5784.9	167.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.8
8	0.7500	5783.0	169.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.2
9	0.7500	5781.5	170.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5951.8
10	0.7500	5780.9	170.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5951.6
11	0.7500	5780.7	170.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5951.5
12	0.7500	5780.6	170.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5951.5
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	35183.1	627.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	35810.8

Entire Reactor

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	13837.4	745.8	4.9	3068.8	86.5	2.5	0.0	0.0	86.5	17745.9
2	0.7500	13029.0	768.3	4.4	3452.4	173.0	10.0	0.3	0.0	173.3	17437.4
3	0.7500	12636.1	745.5	3.5	3504.2	237.0	20.5	1.0	0.0	238.0	17147.9
4	0.7500	12412.2	779.2	3.5	3668.5	245.7	20.9	1.0	0.0	246.7	17130.9
5	0.7500	12381.7	806.7	3.5	3671.3	244.6	20.6	1.0	0.0	245.6	17129.3
6	0.7500	12351.7	832.2	3.5	3673.8	243.7	20.3	1.0	0.0	244.6	17126.2
7	0.7500	12350.7	834.0	3.5	3674.7	243.3	20.2	0.9	0.0	244.3	17127.4
8	0.7500	12349.2	835.5	3.5	3675.1	243.1	20.2	0.9	0.0	244.1	17127.7
9	0.7500	12347.9	836.6	3.5	3675.3	243.1	20.2	0.9	0.0	244.0	17127.5
10	0.7500	12347.3	837.1	3.5	3675.3	243.1	20.1	0.9	0.0	244.0	17127.4
11	0.7500	12347.1	837.3	3.5	3675.4	243.1	20.1	0.9	0.0	244.0	17127.4
12	0.7500	12347.0	837.3	3.5	3675.4	243.0	20.1	0.9	0.0	244.0	17127.4
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	55281.2	2882.2	14.1	11425.3	534.4	33.2	1.3	0.0	535.7	70171.6

Table A-9. Discharge Data for 1200-MWe U-Th/Th 40% Denatured LMFBR with U/ThO_2 Core, ThO_2 Axial and Radial Blankets (RZ Model, Two Core Zones)

Discharge Data (kg)

Reactor Core

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5101.8	760.5	2.0	1229.8	33.9	1.0	0.0	0.0	33.9	7128.9
2	0.7500	4636.2	751.5	1.7	1360.5	66.6	3.8	0.1	0.0	66.7	6820.4
3	0.7500	4374.0	692.3	1.4	1372.8	90.7	7.7	0.4	0.0	91.1	6539.3
4	0.7500	4264.6	707.7	1.5	1447.3	94.5	7.8	0.4	0.0	94.8	6523.8
5	0.7500	4269.0	709.2	1.5	1448.6	94.0	7.7	0.4	0.0	94.3	6530.3
6	0.7500	4272.7	710.3	1.5	1449.6	93.6	7.5	0.3	0.0	93.9	6535.7
7	0.7500	4273.9	710.8	1.5	1449.9	93.5	7.5	0.3	0.0	93.8	6537.4
8	0.7500	4274.4	711.0	1.5	1450.1	93.4	7.5	0.3	0.0	93.8	6538.2
9	0.7500	4274.5	711.0	1.5	1450.1	93.4	7.5	0.3	0.0	93.7	6538.4
10	0.7500	4274.6	711.0	1.5	1450.1	93.4	7.5	0.3	0.0	93.7	6538.5
11	0.7500	4274.6	711.0	1.5	1450.1	93.4	7.5	0.3	0.0	93.7	6538.6
12	0.7500	4274.7	711.1	1.5	1450.2	93.4	7.5	0.3	0.0	93.7	6538.6
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	13295.3	2415.3	6.0	4504.0	204.9	12.3	0.5	0.0	205.4	20438.3

Axial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	4482.4	46.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4528.7
2	0.7500	4433.5	87.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4521.2
3	0.7500	4384.9	124.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.5
4	0.7500	4384.8	124.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.5
5	0.7500	4385.2	124.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.6
6	0.7500	4386.1	123.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.9
7	0.7500	4386.4	123.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
8	0.7500	4386.6	123.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
9	0.7500	4386.6	123.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
10	0.7500	4386.6	123.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
11	0.7500	4386.6	123.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
12	0.7500	4386.6	123.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.0
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	13303.6	256.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	13560.0

Radial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5950.8	30.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5980.9
2	0.7500	5919.8	58.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5978.5
3	0.7500	5888.4	86.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5974.7
4	0.7500	5856.4	113.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5969.4
5	0.7500	5824.3	138.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5962.7
6	0.7500	5792.1	162.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5954.7
7	0.7500	5790.1	164.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5954.2
8	0.7500	5788.3	165.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5953.6
9	0.7500	5786.9	166.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5953.2
10	0.7500	5786.3	166.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5953.1
11	0.7500	5786.0	166.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5953.0
12	0.7500	5786.0	167.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.9
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	35202.1	612.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	35814.6

Entire Reactor

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	15535.0	836.9	2.0	1229.8	33.9	1.0	0.0	0.0	33.9	17638.5
2	0.7500	14989.5	897.9	1.7	1360.5	66.6	3.8	0.1	0.0	66.7	17320.1
3	0.7500	14647.3	903.3	1.4	1372.8	90.7	7.7	0.4	0.0	91.1	17023.5
4	0.7500	14505.8	945.4	1.5	1447.3	94.5	7.8	0.4	0.0	94.8	17002.6
5	0.7500	14478.5	972.0	1.5	1448.6	94.0	7.7	0.4	0.0	94.3	17002.6
6	0.7500	14450.9	996.8	1.5	1449.6	93.6	7.5	0.3	0.0	93.9	17000.3
7	0.7500	14450.4	998.4	1.5	1449.9	93.5	7.5	0.3	0.0	93.8	17001.6
8	0.7500	14449.2	999.7	1.5	1450.1	93.4	7.5	0.3	0.0	93.8	17001.8
9	0.7500	14448.0	1000.8	1.5	1450.1	93.4	7.5	0.3	0.0	93.7	17001.6
10	0.7500	14447.5	1001.2	1.5	1450.1	93.4	7.5	0.3	0.0	93.7	17001.6
11	0.7500	14447.3	1001.4	1.5	1450.1	93.4	7.5	0.3	0.0	93.7	17001.6
12	0.7500	14447.2	1001.5	1.5	1450.2	93.4	7.5	0.3	0.0	93.7	17001.6
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	61801.0	3284.1	6.0	4504.0	204.9	12.3	0.5	0.0	205.4	69812.7

Table A-10. Discharge Data for 1200-MWe ^{233}U-Th/Th LMFBR with $^{233}U/ThO_2$ Axial and Radial Blankets (RZ Model, Two Core Zones)

Discharge Data (kg)

Reactor Core

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PJ	TOTAL HM
1	0.7500	6223.1	818.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	7041.5
2	0.7500	5884.4	839.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6724.2
3	0.7500	5639.1	799.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6437.1
4	0.7500	5603.6	818.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6421.6
5	0.7500	5609.1	819.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6428.4
6	0.7500	5613.6	820.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6433.8
7	0.7500	5615.1	820.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6435.6
8	0.7500	5615.7	820.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6436.4
9	0.7500	5615.8	820.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6436.6
10	0.7500	5615.9	820.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6436.7
11	0.7500	5616.0	820.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6436.8
12	0.7500	5616.0	820.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6436.8
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	17447.6	2694.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	20142.3

Axial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	4482.6	46.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4528.7
2	0.7500	4433.9	87.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4521.3
3	0.7500	4385.5	124.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.7
4	0.7500	4385.5	124.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.7
5	0.7500	4386.0	123.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4509.8
6	0.7500	4386.8	123.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.1
7	0.7500	4387.2	123.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.2
8	0.7500	4387.3	122.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.2
9	0.7500	4387.3	122.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.2
10	0.7500	4387.3	122.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.2
11	0.7500	4387.3	122.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.2
12	0.7500	4387.3	122.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4510.2
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	13305.1	255.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	13560.2

Radial Blanket

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	5950.5	30.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5980.9
2	0.7500	5919.4	59.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5978.4
3	0.7500	5888.0	86.6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5974.6
4	0.7500	5856.0	113.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5969.3
5	0.7500	5823.8	138.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5962.6
6	0.7500	5791.6	163.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5954.5
7	0.7500	5789.8	164.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5954.0
8	0.7500	5788.0	165.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5953.5
9	0.7500	5786.7	166.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5953.1
10	0.7500	5786.0	166.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.9
11	0.7500	5785.8	167.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.8
12	0.7500	5785.7	167.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5952.8
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	35201.3	612.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	35814.2

Entire Reactor

YEAR	CAPACITY FACTOR	TH232	U233	U235	U238	PU239	PU240	PU241	PU242	FISSILE PU	TOTAL HM
1	0.7500	16656.2	894.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	17551.0
2	0.7500	16237.7	986.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	17223.9
3	0.7500	15912.5	1008.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16921.3
4	0.7500	15845.1	1055.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16900.6
5	0.7500	15818.9	1081.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16900.8
6	0.7500	15792.0	1106.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16898.4
7	0.7500	15792.0	1107.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16899.8
8	0.7500	15791.0	1109.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16900.1
9	0.7500	15789.9	1110.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16899.9
10	0.7500	15789.3	1110.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16899.8
11	0.7500	15789.1	1110.7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16899.8
12	0.7500	15789.1	1110.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16899.8
				YEARS 13 THRU 29 SAME AS ABOVE							
30	0.7500	65953.9	3562.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	69516.7

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