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CONCEPT EVALUATION OF A NUCLEAR DESIGN FOR ELECTRONUCLEAR FUEL
PRODUCTION: EVALUATION OF ORNL'S PROPOSED TMF-ENFP

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HIGHLIGHTS

ORNL/TM-6828

CONCEPT EVALUATION OF A NUCLEAR DESIGN FOR ELECTRONUCLEAR FUEL

PRODUCTION: EVALUATION OF ORNL'S PROPOSED TMF-ENFP

T. J. Burns, D. E. Bartine, and J.-P. Renier

The use of an external neutron source, generated via an accelerator, represents a potential means for producing the fissile isotopes ^{233}U and ^{239}Pu . One promising accelerator-breeder concept employs a sodium-cooled ternary metal fertile blanket (^{239}Pu , ^{238}U , and ^{232}Th) surrounding a central column of sodium. A preliminary assessment of the neutronic characteristics and the relevant economic parameters for this system, designated the TMF-ENFP (Ternary Metal Fueled Electronuclear Fuel Producer) is described here.

The rationale for selecting the particular configuration utilized was addressed from the standpoints of fissile production, heat generation, and target interaction. The basic assumption assumed a 300-mA beam of 1-GeV protons from the accelerator. A calculational model of the target blanket assembly was developed which met certain prespecified design limitations. A preliminary assessment of the nuclear performance, thermal-hydraulic, and economic parameters was made. For the latter, the use of the TMF-ENFP in conjunction with the denatured ($^{233}\text{U}/^{238}\text{U}$) fuel cycle in both PWRs and CANDUs was considered.

The basic accelerator-driven blanket concept requires a tradeoff between fissile production and the heat-removal capability. Moreover, the centrally located proton beam results in a significant gradient in the power density (Peak/Average ~ 2). Fissile production of the TMF-ENFP configuration considered was 5.8 kg/day of ^{233}U . Such a production rate would support a maximum of 5.58 GW(e) of denatured PWR capacity [or alternatively 11.58 GW(e) of denatured CANDU capacity]. The cost of the TMF-ENFP/thermal reactor system is judged to be competitive with current-generation LEU-PWRs operating from a $>\$80/\text{lb } \text{U}_3\text{O}_8$ resource base.

1. INTRODUCTION

Of the several accelerator breeder concepts conceived and examined by Oak Ridge National Laboratory,¹ the most recent and most promising is one that employs a sodium-cooled ternary metal blanket (^{239}Pu , ^{238}U , and ^{232}Th) surrounding a flowing column of liquid sodium. Called the Ternary Metal Fueled Electronuclear Fuel Producer (TMF-ENFP), the system would burn ^{239}Pu in a subcritical mode to produce ^{233}U at a rate of approximately 5.8 kg per day. Not only is this production greater by a factor of approximately 2 than that from a reactor of the same thermal power rating, the TMF-ENFP is also self-sufficient in terms of its own energy balance.

The TMF-ENFP is viewed as having a high potential in reduced proliferation scenarios in that it can be coupled with dispersed nuclear power reactors operating on a denatured ^{233}U fuel cycle. It will sacrificially destroy all the plutonium produced from the ^{238}U denaturant of these reactors to provide both excess neutrons (to produce ^{233}U) and energy (to power the accelerator). In the scenarios visualized, the reactors would be dispersed throughout the country (or the world) to supply power needs and the TMF-ENFPs would be located in secure energy parks, along with the associated fuel reprocessing and fabrication facilities. Thus no plutonium in the system would exist outside the parks except that built up in the spent fuel elements and transported to the parks for destruction, and the high level of radioactivity of the spent fuel elements would act as a deterrent during their transport.

The TMF-ENFP is also attractive from an environmental viewpoint. Because ^{235}U will not be utilized in the system, it will no longer be necessary to mine large quantities of uranium ore in order to utilize a

relatively small percentage (0.7%) in a reactor. Sufficient quantities of the ^{238}U required for the denaturant have already been mined, and, since the thorium ore required to produce the ^{233}U can be fully utilized, the thorium ore requirements will be much less than the uranium ore requirements for reactors utilizing ^{235}U as their fuel. Thus the destruction of the environment due to mining will be greatly lessened.

It also appears that the coupling of the TMF-ENFP with reactors operating on denatured ^{233}U fuel will reduce a number of problems related to nuclear safety. In particular, the possibility of a transient overpower excursion will be extremely remote since the TMF-ENFP will be a subcritical system and, moreover, it will be a system that can be shut down simply by turning off the accelerator power. Also, the subcritical mode of operation will allow safety features to be included in the design that might not be acceptable in a reactor because they would interfere with its nuclear performance.

In Section 2 we briefly describe the nuclear physics aspects of the TMF-ENFP and the engineering considerations required for its development. We do not include a discussion of the associated accelerator, it being assumed that the technology is already available for providing a proton beam of the intensity required. Neither do we discuss the calculations of the interactions of the protons and the neutrons they produce, since the most recent studies are reported in detail in Reference 2.

The most recent calculations support the assumptions we have made here regarding nuclear performance of the target-blanket system. Also, we do not include here a discussion on radiation damage studies since these are discussed in considerable detail in Reference 3.

In Section 3 we present the results of an analysis of the fuel resource utilization and fuel cycle costs for the TMF-ENFP in several scenarios in which the ^{233}U produced by the TMF-ENFP is used in a denatured form in pressurized water reactors or in CANDU reactors. On the basis of this analysis and the promise of the design itself, we believe that further studies to evaluate the TMF-ENFP should be pursued.

2. ELECTRONUCLEAR FISSILE PRODUCTION USING A TERNARY METAL BLANKET

2.1. Generic Physics Considerations

Owing to the significant differences in the physics behavior of critical (i.e., self-sustaining) reactor systems and multiplying systems that are driven by external sources, a brief discussion of the generic physics considerations of the latter is presented here.

Since before a neutron can be utilized to breed fissile material it must first be produced, the discussion begins by considering the total number of neutrons produced in a source-driven multiplying system, and hence the total number of neutrons available for capture by fertile material. The total number of neutrons produced per second, N , is just

$$N = S + \langle \nu \Sigma_f \phi \rangle, \quad (1)$$

where S is the accelerator-produced neutron source and the brackets indicate integration over space and energy. Noting that $\Sigma_f \phi$ is just the fission rate, Eq. (1) becomes

$$N = S + \bar{\nu} F. \quad (2)$$

This equation embodies a significant concept related to electronuclear fuel production. The total number of neutrons available for fertile capture is directly related to the system's thermal power, here expressed as the fission rate. Moreover, once the thermal power level is chosen, Eq. (2) indicates that the only nuclear parameter available for optimization is $\bar{\nu}$, the average number of neutrons produced per fission of a heavy-metal atom.

However, an additional parameter needs to be addressed. Using the basic definition of the effective multiplication constant, k , the overall system fission rate (and hence the thermal fission power) can be expressed as

$$F = (S/\bar{\nu}) [k/(1 - k)] . \quad (3)$$

If it is assumed that all absorptions in fertile materials are capture reactions and all absorptions in fissile materials are fission reactions, and further that these two types of reactions are the only ones that occur, the net fertile capture rate is given by

$$C = (S/\bar{\nu}) [(\bar{\nu} - k)/(1 - k)] . \quad (4)$$

Additionally, the net fissile production rate is given by

$$P = (S/\bar{\nu}) [(\bar{\nu} - 2k)/(1 - k)] . \quad (5)$$

Considering Eqs. (3) through (5), it is evident that all the reaction rates increase with increasing values of k . Perhaps not as obvious, however, is the fact that the three reaction rates do not increase at the same rate. For example, if a representative value of 2.46 is assumed for $\bar{\nu}$, the relative values of these three reaction rates for increasing values of k are as tabulated in Table 1.

Table 1. Comparison of Reaction Rates in an ENFP

k	Fissions per Source Neutron	Fertile Captures per Source Neutron	Net Fissile Production (atoms per Source Neutron)
0	0	1.0	1.0
0.10	0.045	1.066	1.021
0.20	0.102	1.148	1.047
0.30	0.174	1.254	1.080
0.40	0.271	1.396	1.125
0.50	0.407	1.593	1.187
0.60	0.610	1.890	1.280
0.70	0.949	2.385	1.436
0.80	1.626	3.374	1.748
0.90	3.659	6.341	2.683
0.95	7.724	12.276	4.553
0.99	40.244	59.756	10.512

In order to emphasize the different behavior of these reaction rates as a function of k , these values are plotted in Fig. 1. Instead of the relative fission rate, the total fission power (assuming a source of 5×10^{19} neutrons/sec) is plotted. Clearly, of the three reaction rates, the system's power level is most sensitive to changes in k and the net fissile production rate is the least sensitive. From the standpoint of maximizing the fertile capture rate (or the net fissile production rate), the system should be operated as close to criticality as feasible from a safety viewpoint. However, as Fig. 1 indicates, this implies a high thermal power generation rate and hence a large investment in heat removal equipment.

To consider some of the possible options, it is convenient to normalize the fertile capture rate (and net fissile production rate) by

Fig. 1. Comparison of Reaction Rates in an ENFP.

the fission rate to indicate the gain in each per fission (i.e., per unit investment in heat removal equipment). The results

$$C/F = (\bar{\nu}/k) - 1 \quad (6)$$

and

$$P/F = (\bar{\nu}/k) - 2 \quad (7)$$

indicate that the means of optimizing the amount of fissile fuel produced per fission lies in $\bar{\nu}$. Since $\bar{\nu}$ typically increases as the spectrum is hardened, maximizing Eqs. (6) and (7) implies the hardest spectrum attainable.

2.2. TMF-ENFP Blanket/Target Design

In view of the desire to achieve the hardest possible spectrum for the assembly, a metal was chosen as the basic form of the fertile/fissile material. Moreover, considering the possible role of an ENFP as a source of fissile material for thermal converters (particularly in a reduced proliferation scenario), the ORNL ENFP was designed to produce ^{233}U . Thus, ^{232}Th metal is utilized as the base fertile material. In order to maximize ^{233}U production, the system has been designed to sacrificially burn Pu. Using Pu as the primary source of neutrons has three primary motivations. First, the value of ∇ is maximized by using Pu instead of ^{233}U as the fissile component. Second, the use of Pu as the primary neutron multiplier allows the net ^{233}U production to be increased since the bred ^{233}U does not have to be sacrificed to maintain the system's multiplication. Lastly, if the ENFP operates in conjunction with denatured reactors, Pu would be produced and hence stockpiles of Pu would increase if it were not consumed in some manner. In addition to the Pu and Th, the alloy would also contain ^{238}U . The purpose of including ^{238}U in the alloy is two-fold. First, the presence of ^{238}U reduces the effect on the reactivity that is due to the burnout of the original Pu. Second, it increases the fertile fast fission effect and thereby reduces the consumption rate of fissile material required to maintain a given multiplication, particularly in the hard spectrum characteristic of a metal-fueled system.

The use of a Th-based alloy (>78% Th) is also beneficial regarding thermal and irradiation performance. Due to high thermal conductivity of the metal, the fuel is capable of sustaining very high power densities while maintaining reasonable centerline temperatures. The irradiation behavior of Th-based alloys is considered in Reference 3.

Ideally, to minimize the parasitic losses and spectrum degradation, the fuel would be unclad metal. However, cladding is necessary to prevent fission-product contamination of the coolant, and this particular design utilizes V/Ti as the cladding material. V/Ti is specified for two reasons. First, it is compatible with the ternary metal fuel from a metallurgical sense (unlike stainless steel). Second, the neutronic properties affecting the neutron spectrum, principally the inelastic scattering cross sections, are markedly better than those of conventional fast reactor cladding materials.

An analogous argument applies to the choice of coolant. In the absence of non-neutronic considerations, the primary coolant would be gaseous (helium). However, considering the potential cooling requirements and the requirement for an accelerator beam window capable structurally of separating the very high pressure He from the accelerator vacuum, sodium has been selected as an alternate coolant. One advantage of the use of Na as a coolant (in addition to its excellent thermal properties) is that much of the liquid metal technology developed for the fast breeder program can be directly utilized.

The choice of target material is dictated primarily by its associated cooling requirement. The use of solid target material has a number of disadvantages, in addition to the problem of heat generation. The severe charged particle radiation damage, as well as the associated possible loss of structural integrity, indicates one of the potential problem areas. Thus, a liquid target has been utilized. Since high-energy transport results indicate little sensitivity to the target nuclear mass (see Reference 2), a Na target is employed, both for heat transfer reasons and for design compatibility with the blanket coolant material. The

use of the same coolant for both regions also eliminates the need for a structural component which would otherwise be required to separate the two regions and which would have a severely limited lifetime owing to radiation damage.

The net result of these considerations is a Na-cooled V/Ti-clad, ternary metal system -- the TMF-ENFP. Figure 2 depicts the calculational model for this system. A central sodium target (with a Na-cooled Th metal beam stop) is surrounded by a two-zone multiplier. The Pu enrichment is zoned to flatten the otherwise steep radial power gradient. The multiplier is surrounded by clad Th metal blanket regions to utilize the neutrons which are produced in but which leak out of the high flux multiplier.

2.3. ENFP Operating/Performance Characteristics

Some of the pertinent operating and performance characteristics of the proposed ternary metal fuel production facility are given in Table 2. It should be emphasized that the parameters are for the reference design depicted in Fig. 2. This reference design is intended to serve as a base case so that the effects of various future optimization changes and tradeoffs can be assessed. Thus, although the design does account for various specific physical limitations, it possibly does not represent the optimum design for the ternary metal concept.

The use of a centrally driven subcritical multiplying region does present a design problem regarding the radial power distribution. The midplane distribution for the reference design is shown in Fig. 3. As indicated, the peak-to-average ratio of the power density is ~ 2 . Although this value was regarded as acceptable for the preliminary evaluations, clearly additional effort aimed at reducing the peak-to-average ratio is required.

Fig. 2. Calculational Model of TMF-ENFP.

Table 2. ENFP Design Basis

Multiplier Region:

Composition	Ternary Metal
Pin OD (in.)	0.250
Cladding Thickness (in.)	0.015
Fuel Rod Gap (in.)	0.00325
Spacers	Grid
Pins/Assembly	271
Pitch Diameter (in.)	1.33
Cladding Material	V-Ti alloy
Length (in.)	78.7
V_{Fuel}	0.35
V_{Na}	0.50
$V_{\text{Structure}}$	0.15

Radial Blanket Region:

Composition	Th Metal
Pin OD (in.)	0.484
Cladding Thickness (in.)	0.015
Rod Gap (in.)	0.00325
Spacers	Grid
Pins/Assembly	91
Pitch Diameter (in.)	1.16
Cladding Material	V-Ti alloy
Length (in.)	118.1
V_{Metal}	0.57
V_{Na}	0.30
$V_{\text{Structure}}$	0.13

ENFP Performance ParametersNuclear:

K_{eff} - BOC	0.84
K_{eff} - EOC	0.88
Multiplication - BOC	6.3
Multiplication - EOC	8.3
^{233}U Production Rate	5.8 kg/FPD
Pu Net Destruction Rate	2.9 kg/FPD
Fissile Loading	3400 kg
Residence Time	90 FPD
Average Burnup [Inner]	17 MWD/kg
[Outer]	9 MWD/kg

Thermal Hydraulic:

Thermal Power - BOC	3300 MW(Th)
Thermal Power - EOC (uncompensated)	4500 MW(Th)
Na Inlet Temperature	620° F
Maximum Na Outlet Temperature	995° F
Na Velocity	30 ft/sec
Peak Centerline Temperature	1382° F
Peak Cladding Temperature	1292° F
Maximum Smear Power Density	1100 W/cm ³
Maximum Radial Power Density	3150 W/cm ³

Fig. 3. Radial Power Distribution for Reference TMF-ENFP Concept.

Although the primary goal of the TMF-ENFP is to maximize the amount of fissile production, achievement of the goal presents a possible control problem. The attainment of a high fissile production rate typically causes the multiplication constant (and thus the power level) to increase with time. Possible methods of addressing this power swing are:

- (1) Design balance of plant to handle end-of-life power generation.
- (2) Adjust the fixed source level.
- (3) Introduce neutron poisons in the blanket to achieve a constant multiplication factor.

It should be noted that each of these three options represents a compromise between practical and optimum performance. In the first and second options,

high capital-cost facilities (either the heat removal equipment or the accelerator) are operated at below design capacity. In the last option, the presence of the poison will reduce the system's production rate.

One significant aspect of the operating characteristics of the TMF-ENFP does deserve amplification, however, and that is the effective multiplication factor range. As Fig. 1. clearly indicates, the production rates will be maximized by operating as close to critical as practicable. However, two factors dictate a reduced value for k . First, the tendency for k to increase with time must be accounted for. Second, the breeding of ^{233}U introduces a significant safety factor due to the relatively long half-life of ^{233}Pa . At the end of 90 full-power days, significant quantities of ^{233}Pa are present and thus any design must account for the possible criticality consequences associated with this potential increase in fissile mass after shutdown.

2.4. Alternate ENFP Design

Although the operating and performance characteristics of the ternary metal system appears reasonable, it is possible that unforeseen problems will make the concept impractical. Thus, it is prudent to consider an alternate design. Preliminary results indicate that an integral target/blanket of fluoride salts would have certain thermal and mechanical advantages over the ternary system. In particular, the elimination of the cladding and structural components markedly reduces the susceptibility of the system to radiation damage. Moreover, the fluid form of the target/blanket alleviates much of the problem of radial power distribution. Additionally, the on-line capability for ^{233}Pa extraction is an inherent advantage of molten salt systems. The molten salt concept does have certain inherent disadvantages, particularly in the production rate of fuel. This is estimated to be approximately 1.5 kg per full power day.

One significant problem is the inability of conventional salts to dissolve significant quantities of Pu, thereby eliminating the possibility of using a ternary salt. A further complication is the presence of ^6Li in the salt, which significantly affects the system's neutron economy. However, it is also clear that if a salt composition can be found in which trivalent solubility is greater than at present, the molten salt concept would significantly reduce the design problems inherent in the TMF-ENFP.

3. FUEL RESOURCE UTILIZATION AND FUEL CYCLE AND POWER COSTS FOR SYSTEMS CONSISTING OF A TMF-ENFP AND THERMAL CONVERTER REACTORS

As has been stated in Section 2, the principal objective of the proposed TMF-ENFP is the production of ^{233}U which will be the fissile component of denatured fuel used in thermal converter reactors. The TMF-ENFP achieves this purpose by sacrificially consuming Pu as a neutron multiplier, the Pu being produced as a byproduct in the ^{238}U used as the denaturant in the reactor fuel. In this section we examine the use of fuel resources and the corresponding fuel cycle and power costs when the TMF-ENFP is combined with two types of converters: current-generation PWRs (modified somewhat to burn denatured fuel) and denatured CANDUs. The denatured PWRs used in the analysis represent near-term converters and the denatured CANDUs represent possible advanced converters.

3.1. Fuel Resource Utilization

Table 3 shows the utilization of fuel resources by the TMF-ENFP, the net production of ^{233}U being 5.90 kg per FPD (full power day) or 1,613 kg per year and the net burnup of plutonium being 2.78 kg per FPD or 760 kg per year.

Table 3. Fuel Resource Utilization of the TMF-ENFP

	$^{233}\text{U}^a$	^{238}U	Fissile Pu	^{232}Th	Total ^b
<u>Inner and Outer Core (Zones 1 and 8)</u>					
Loading, kg/yr	0	14,130	10,270	84,290	108,690
Discharge, kg/yr	1,077	13,935	9,510	83,090	107,655
Net requirement, kg/yr	-1,077 ^c	195	760	1,200	
Net requirement, kg/FPD	-3.930		2.78		
<u>Axial Thorium Blanket (Zone 4)</u>					
Loading, kg/yr				54,030	54,030
Discharge, kg/yr	167			53,860	54,027
Net requirement, kg/yr	-167			170	
Net requirement, kg/FPD	-0.610				
<u>Radial Thorium Blanket (Zone 5)</u>					
Loading, kg/yr				42,485	42,485
Discharge, kg/yr	340			42,130	42,471
Net requirement, kg/yr	-340			355	
Net requirement, kg/FPD	-1.240				
<u>Thorium Beam Stop (Zone 2)</u>					
Loading, kg/yr				1,530	1,530
Discharge, kg/yr	29			1,500	1,529
Net requirement, kg/yr	-29			30	
Net requirement, kg/FPD	-0.105				
<u>Total (All Zones)</u>					
Loading, kg/yr	0	14,130	10,270	182,335	206,735
Discharge, kg/yr	1,613	13,935	9,510	180,580	205,683
Net requirement, kg/yr	-1,613	195	760	1,755	
Net requirement, kg/FPD	-5.90		2.78		

^a $^{233}\text{U} + ^{233}\text{Pa}$ ($^{233}\text{Pa} \rightarrow ^{233}\text{U} = 0.972$).

^bTotal includes ^{234}U , ^{235}U , ^{236}U , etc.

^cThe minus sign indicates net production.

In this analysis it was assumed that the capacity factor of the TMF-ENFP was 75%, so that one year of operation would be equivalent to 273 FPDs. It was also assumed that the residence time for the heavy metal loadings would be as follows: 91 FPDs in zones 1, 8, and 4; 273 FPDs in zone 5, and 182 FPDs in zone 2. The zone numbers in the table correspond to those shown in Fig. 2.

Table 4 shows the utilization of fuel resources by the PWR and CANDU converter reactors operating on a denatured ^{233}U fuel cycle with SGRU (self-generated recycled uranium). For comparison, it also gives comparable figures

Table 4. Fuel Resource Utilization of Various Converter Reactor Systems
(1000 MWe; 30-yr Lifetime; 71.4% Average Capacity Factor)

	Loading (kg)		Discharge (kg)		Lifetime (kg)	
	1st Cycle	Equilibrium Cycle (10th)	1st Cycle	Equilibrium Cycle (10th)	Total	Total ^a (Net)
DU ²³³ -PWR-SGRU						
²³³ U	1,658	720	340	421	9,410	9,540
²³⁸ U	12,132	5,050	2,865	5,105	10,195	11,665
Fissile Pu	0	0	20.8	74	-1,860	-1,840
²³² Th ^d	58,140	16,800	17,875	16,460	32,055	36,820
DU ²³³ -CANDU-SGRU						
²³³ U	1,648	797	616	700	2,827	3,045
²³⁸ U	12,060	5,745	4,600	5,665	2,340	4,090
Fissile Pu	0	0	8.2	30.5	-885	-875
²³² Th ^d	99,510	47,085	37,855	46,330	22,245	36,530
LEU-PWR with No Recycle ^b						
²³⁵ U	1,410	760	170	205		23,460
²³⁸ U	77,655	24,540	22,695	23,955		787,100
Fissile Pu	0	0	86	167		5,040
U ₃ O ₈ (0.200% tails) Sep. Work	319 S.T. 140 MSWU	181 S.T. 110 MSWU				5,555 S.T. 3,315 MSWU
DU ²³⁵ -CANDU-SGRU ^c						
²³³ U	0	635	123	646		-1,935
²³⁵ U	2,120	250	630	100		6,265
²³⁸ U	8,485	5,350	3,235	5,210		1,705
Fissile Pu			6.5	29		825
²³² Th ^d	102,450	47,325	39,080	46,670		21,410
U ₃ O ₈ (0.200% tails) Sep. Work	534 S.T. 485 MSWU	37.5 S.T. 36 MSWU				1,695 S.T. 1,600 MSWU
DU ²³⁵ -CANDU-U ²³³ -STOWAWAY ^e						
²³³ U	0	0	368	455		-13,740
²³⁵ U	2,120	1,010	260	325		20,695
²³⁸ U	8,485	4,040	3,200	3,980		1,875
Fissile Pu			20	24		-735
²³² Th ^d	102,450	48,800	36,685	48,060		21,990
U ₃ O ₈ (0.200% tails) Sep. Work	534 S.T. 485 MSWU	255 S.T. 230 MSWU				7,855 S.T. 7,135 MSWU

^aIncludes 1% process losses.

^bWith 3.7 million short tons of U₃O₈, 666 reactors could be operated, for a cumulative power production of 20 TWe-yr.

^cWith 3.7 million short tons of U₃O₈, 2,182 reactors could be operated, for a cumulative power production of 65 TWe-yr.

^dTh is recycled only after 30 years of storage.

^eWith 3.7 million short tons of U₃O₈, 470 reactors could be operated, for a cumulative power production of 14 TWe-yr; with the ²³³U generated by these reactors, 2,290 DU²³³ CANDU-SGRU reactors could be operated, for a cumulative power production of 69 TWe-yr.

for a current-generation PWR operating on low-enriched uranium (no recycle) and for two CANDUs operating on denatured ^{235}U fuel, one with SGRU and the other with ^{233}U stowaway.* The large differences in fuel utilization between the denatured PWR and the denatured CANDU are due to the reduced ^{233}U makeup requirements of the CANDU system, as is shown in Fig. 4.

With the values shown in Tables 3 and 4, the power support capability of the TMF-ENFP for the denatured PWR and CANDU systems can be estimated. The results, shown in Tables 5 and 6, indicate that one TMF-ENFP, operated over its design lifetime of 30 years, will support 15 denatured CANDUs or five denatured PWRs introduced into the power system at various times after the startup of the TMF-ENFP, each reactor rated at 1 GWe. With the TMF-CANDU system a total on-line power of 467.4 GWe-yr will be produced, and with the TMF-PWR system a total of 167.4 GWe-yr will be produced. These data presume that the TMF-ENFP is not replaced at the end of its lifetime; however, with the ^{233}U it produces, the lifetime of the TMF-CANDU and TMF-PWR systems will be extended to 81 years and 41 years respectively. (It should be noted that the envisioned systems are not self-sufficient in terms of Pu, it being presumed that such makeup Pu is available either from stockpiles or from reprocessed LEU fuel.)

The power generation in each year is shown in Figs. 5 and 6 for the TMF-CANDU and TMF-PWR systems respectively. The TMF itself contributes a net average electrical power production (exclusive of its own power requirements) of approximately 580 MWe, but, of course, this power is unavailable in each system after year 30.

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*The TWe-yr used in Table 4 and in subsequent tables is equivalent to 10^3 GWe-yr or 10^6 MWe-yr.

Fig. 4. Cumulative ^{233}U Requirements of PWR and CANDU Denatured Cycles (1000 MWe, 30-Yr Lifetime, 71.4% Average Capacity Factor).

Fig. 5. On-Line Electrical Power Generated by a System Consisting of 1 TMF-ENFP and 15 DU²³³ CANDUs.

Fig. 6. On-Line Electrical Power Generated by a System Consisting of 1 TMF-ENFP and 5 DU²³³ PWRs.

Table 5. Fuel Resource Utilization of a TMF-ENFP/DU²³³-CANDU-SGRU System

Power Rating:

Cumulative on-line power: 467.4 GWe/yr
Maximum on-line power: 11.58 GWe

Ratio:

1 TMF-ENFP (0.58 GWe average) supports 15 DU²³³ CANDUs (1 GWe each).
Assuming the TMF-ENFP starts up in year 1, the 15 CANDUs start up in
years 4, 5, 6, 8, 10, 12, 14, 17, 20, 23, 28, 36, 40, 49, 50.
System lifetime: 81 years.

	Lifetime Resource Requirements							
	DU-CANDU's (15)			TMF-ENFP (1)			System	
	Total (kg)	Processing Losses (kg)	Total Net (kg)	Total (kg)	Processing Losses (kg)	Total Net (kg)	Total Net (kg)	Total Net (kg/30 GWe-yr)
²³³ U	42405	3255	45660	-48390	485	-47905	-2245	-144
²³⁸ U	35100	26250	61350	5850	4180	10030	71380	4580
Fissile Pu	-13275	135	-13140	22800	2855	25655	12515	805
²³² Th	333675	214275	547950	52650	54175	106825	554775	42025

Note: If it is assumed that 3.7 million short tons of U₃O₈ and an unlimited supply of Pu are available, 39,810 systems (each at 467.4 GWe-yr = 18,607 TWe-yr) could be operated and would consume 28.6 million short tons of thorium.

Table 6. Fuel Resource Utilization of a TMF-ENFP/DU²³³-PWR-SGRU System

Power Rating:

Cumulative on-line power: 167.4 GWe/yr
Maximum on-line power: 5.58 GWe

Ratio:

1 TMF-ENFP (0.58 GWe average) supports 5 DU²³³ PWRs (1 GWe each).
Assuming the TMF-ENFP starts up in year 1, the 5 PWRs start up in
years 4, 5, 7, 10, 11.
System lifetime: 41 years.

	Lifetime Resource Requirements							
	DU-PWRs (5)			TMF-ENFP (1)			System	
	Total Total (kg)	Processing Losses (kg)	Total Net (kg)	Total (kg)	Processing Losses (kg)	Total Net (kg)	Total Net (kg)	Total Net (kg/30 GWe-yr)
²³³ U	47050	653	47705	-48390	485	-47905	-200	-35
²³⁸ U	50975	7350	58325	5850	4180	10030	68355	12250
Pu-fissile	-9300	100	-9200	22800	2855	25655	16455	2950
²³² Th	160275	23815	184090	52650	54175	106825	290915	52135

Note: If it is assumed that 3.7 million short tons of U₃O₈ and an unlimited supply of Pu are available, 41,571 systems (each at 167.4 GWe-yr = 6660 TWe-yr) could be operated and would consume 13.3 million short tons of thorium.

3.2. Fuel Cycle and Power Costs

The primary assumptions employed in the economic analysis are presented in Table 7. It is to be noted that the fixed capital cost of the TMF totals \$1.5 billion [based on a cost of \$1,092,000,000 for the TMF "reactor" (780 kWe x 1400 MWe) plus \$400,000,000 for the accelerator], and that the fixed charge rate is 15% per year compounded.

Table 8 shows that, using the assumptions of Table 7 and \$0/gram for Pu, the TMF-PWR system will have a power cost of 34.5 mills/kWe-hr and a fuel cycle cost of 11.1 mills kWe-hr, while the TMF-CANDU system will have a power cost of 38 mills/kWe-hr and a fuel cycle cost of 10.8 mills/kWe-hr. However, if it is assumed that the fixed capital costs of the TMF-ENFP can be reduced from \$1.5 billion to \$1 billion, which may be realistic since the costs for the TMF reactor are based on FBR estimates and the containment and safety systems for the TMF reactor should not be nearly as expensive, and further if it is assumed that the fixed charge rate is 10% rather than 15%, then the TMF-PWR power and fuel cycle costs are reduced to 23.1 mills/kWe-hr and 9.4 mills/kWe-hr, respectively, and the TMF-CANDU power and fuel cycle costs are reduced to 26.5 mills/kWe-hr and 9.1 mills/kWe-hr respectively. These costs are comparable with those associated with current-generation LEU-PWRs operating from a \$80/lb U_3O_8 resource base.

Table 7. Economic and Unit Cost Assumptions Used in Analysis

Initial Capital Cost (including interest during construction):

LWR:	\$625/kWe
CANDU:	\$755 + \$160 (initial D ₂ O) = \$915/kWe
ENFP Reactor:	\$780/kWe (same as FBR)
ENFP Accelerator:	\$400,000,000

Operations, Maintenance and Nuclear Liability Cost:

LWR (1GWe at 71.4% capacity factor):	M\$5.54/yr
CANDU (1GWe at 71.4% capacity factor):	M\$10.3/yr
ENFP Reactor (1.4 GWe at 75% capacity factor):	M\$8.15/yr
ENFP Accelerator:	M\$4/yr

Fabrication of fresh fuel and refabrication of fuel with recycled material:

LEU LWR (²³⁵ U - ²³⁸ U) ₂ :	\$100/kg HM
DE LWR (²³⁵ U - ²³⁸ U - Th) ₂ :	\$140/kg HM
DE LWR (²³³ U - ²³⁸ U - Th) ₂ :	\$550/kg HM
Nat & SE CANDU (²³⁵ U - ²³⁸ U) ₂ :	\$60/kg HM
HE CANDU (²³⁵ U - ²³⁸ U - Th) ₂ :	\$85/kg HM
DE CANDU (²³⁵ U - ²³⁸ U - Th) ₂ :	\$85/kg HM
DE CANDU (²³³ U - ²³⁸ U - Th) ₂ :	\$350/kg HM
ENFP (Pu - ²³⁸ U - core):	\$700/kg HM
ENFP (Th - Axial Blanket):	\$25/kg HM
ENFP (Th - Radial Blanket):	\$150/kg HM

Reprocessing and Spent Fuel Shipping Costs:

LWR:	\$165/kg HM
CANDU:	\$160/kg HM
FBR:	\$280/kg HM
ENFP:	\$165/kg HM

Waste Storage and Shipping Costs:

LWR:	\$55/kg HM
CANDU:	\$20/kg HM
FBR:	\$165/kg HM
ENFP:	\$55/kg HM

Throwaway Costs: (includes spent fuel shipping costs and permanent storage)

LWR:	\$115/kg HM
CANDU:	\$110/kg HM
FBR:	\$180/kg HM

²³⁵U Enrichment Cost:

\$75/SWU

Thorium Cost:

\$29/kg HM

The thorium cost is calculated on the assumption that irradiated thorium material is not recycled back

U₃O₈ (Natural) Cost:\$50/lb U₃O₈parametric variation \$0/lb, U₃O₈, \$25, \$50, \$75, \$100, \$125External ²³³U Cost:

parametric variation

External Pu-fissile Cost:

parametric variation

Total Equivalent Cost of Money (Fixed Rate)

15% per year compounded

Table 8. Summary of TMF-Reactor System Costs

System	Costs (mills/kWe-hr)	
	Power Costs	Fuel Cycle Costs
\$1.5 billion TMF, 15% fixed charge rate, \$0/gram Pu		
TMF-PWR (5 PWRs at 1 GWe each; cumulative power is 167.4 GWe-yr)	34.5	11.1
TMF-CANDU (15 CANDUs at 1 GWe; cumulative power is 467.4 GWe-yr)	38	10.8
\$1 billion TMF, 10% fixed charge rate, \$0/gram Pu		
TMF-PWR	23.1	9.4
TMF-CANDU	26.5	9.1

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