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EXECUTIVE SUMMARY
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Interim Assessment of the Denatured ^{233}U Fuel Cycle: Feasibility and Nonproliferation Characteristics

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EXECUTIVE SUMMARY
OF
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INTERIM ASSESSMENT OF THE DENATURED ^{233}U FUEL CYCLE: FEASIBILITY AND NONPROLIFERATION CHARACTERISTICS

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ABSTRACT OF ORNL-5388

A fuel cycle that employs ^{233}U denatured with ^{238}U and mixed with thorium fertile material is examined with respect to its proliferation-resistance characteristics and its technical and economic feasibility. The rationale for considering the denatured ^{233}U fuel cycle is presented, and the impact of the denatured fuel on the performance of Light-Water Reactors, Spectral-Shift-Controlled Reactors, Gas-Cooled Reactors, Heavy-Water Reactors, and Fast Breeder Reactors is discussed. The scope of the R,D&D programs to commercialize these reactors and their associated fuel cycles is also summarized and the resource requirements and economics of denatured ^{233}U cycles are compared to those of the conventional Pu/U cycle. In addition, several nuclear power systems that employ denatured ^{233}U fuel and are based on the energy center concept are evaluated. Under this concept, dispersed power reactors fueled with denatured or low-enriched uranium fuel are supported by secure energy centers in which sensitive activities of the nuclear cycle are performed. These activities include ^{233}U production by Pu-fueled "transmuters" (thermal or fast reactors) and reprocessing. A summary chapter presents the most significant conclusions from the study and recommends areas for future work.

CONTENTS OF EXECUTIVE SUMMARY

Page

PROLOGUE 1

1.0. INTRODUCTION 1

2.0. ISOTOPICS OF DENATURED ^{233}U FUEL: NONPROLIFERATION AND FUEL CYCLE IMPLICATIONS 5

2.1. Nonproliferation Advantages 6

Isotopic Barrier of Fresh Fuel 6
Radiation Barrier of Fresh Fuel 6
Reduced Attractiveness of Spent Fuel 8
Reduced Accessibility of Isolated Fissile Isotope 8

2.2. Fuel Cycle Impact 8

2.3. Conclusions 10

3.0. STRUCTURE OF THE DENATURED ^{233}U FUEL CYCLE 11

3.1. Reactor Types Used in Denatured Fuel Cycle 12

3.2. Other Components of Fuel Cycle 13

3.3. Locations of Fuel Cycle Components 13

3.4. Symbiotic Character of Denatured Fuel Cycle 15

3.5. Conclusions 15

4.0. IMPACT OF ALTERNATE FUELS ON REACTOR PERFORMANCE 17

4.1. Alternate Fuel Types Considered 17

4.2. Reactor Designs Calculated 18

Light-Water Reactors (PWRs and BWRs) 18
Spectral-Shift-Controlled Reactors (SSCRs) 19
Heavy-Water Reactors (HWRs) 19
Gas-Cooled Reactors (HTGRs and PBRs) 19
Liquid-Metal Fast Breeder Reactors (LMFBRs) 20

4.3. Comparisons of Various Reactor-Fuel Combinations 20

Thermal Reactors 20
Fast Reactors 24

4.4. Conclusions 27

5.0. IMPLEMENTATION OF DENATURED FUEL CYCLE 28

5.1. Reactor Research and Development Requirements and Costs 29

Light-Water Reactors (LWRs) 29
Spectral-Shift-Controlled Reactors (SSCRs) 30
Heavy-Water Reactors (HWRs) 30
High-Temperature Gas-Cooled Reactors (HTGRs) 30
Fast Breeder Reactors (FBRs) 31
Summary of Reactor R,D&D Costs 31

5.2. Fuel Recycle Research and Development Requirements and Costs..... 32

Fuel Fabrication/Refabrication and Qualification..... 32
Fuel Reprocessing..... 33
Waste Treatment..... 33
Summary of Fuel Recycle R,D&D Costs..... 33

5.3. Possible Procedure for Implementing Denatured ^{233}U Fuel Cycle..... 35

5.4. Conclusions..... 36

6.0. ADEQUACY OF DENATURED POWER SYSTEMS FOR MEETING POWER DEMANDS..... 37

6.1. Systems with Price-Limited Uranium Supplies..... 40

Typical Results..... 40
Summary of Installed Nuclear Capacities, Energy Support Ratios and Ore/Enrichment Requirements..... 42

6.2. Systems with Unconstrained Uranium Supplies..... 45

6.3. Conclusions..... 47

7.0. CONCLUSIONS AND RECOMMENDATIONS..... 49

7.1. No-Recycle Options..... 50

7.2. Classical Reference Recycle Options..... 50

7.3. Denatured Recycle Options..... 52

7.4. Overall Conclusions and Recommendations..... 54

App. A. ^{232}U PRODUCTION AND DECAY PROCESSES..... 57

App. B. TIME-INTEGRATED DOSES DUE TO INHALED U AND Pu ISOTOPES..... 58

App. C. EFFECT OF IMPROVED LWR DESIGNS AND ENRICHMENT TECHNOLOGY..... 59

PROLOGUE

In a 1976 article published in the *Bulletin of the Atomic Scientists*[1] Feiveson and Taylor of Princeton University proposed that a thorium-based nuclear fuel cycle in which the ^{233}U fissile component is denatured with ^{238}U be considered as an alternative to the uranium-based plutonium cycle. Their thesis was that the denatured ^{233}U cycle could be made more proliferation resistant than the plutonium cycle, and, moreover, that it might even eliminate the necessity for fast reactors operating on and breeding plutonium. Soon thereafter a multi-institutional feasibility study of the denatured ^{233}U cycle was initiated by the Department of Energy, with Argonne National Laboratory, Brookhaven National Laboratory, Combustion Engineering, Inc., Hanford Engineering Development Laboratory, the Oak Ridge Gaseous Diffusion Plant, and Oak Ridge National Laboratory as the participants. ORNL was assigned the responsibility for compiling and editing the contributions to the study and publishing the final report, which was issued in December, 1978, as ORNL-5388 *(Interim Assessment of the Denatured ^{233}U Fuel Cycle: Feasibility and Nonproliferation Characteristics)*. An extended summary of the report is presented here.

1.0. INTRODUCTION

Ultimately the extent to which nuclear power can be used throughout the world will depend on the availability and cost of the fissile fuel supply. While today's generation of power reactors have remained competitive, even with the recent sharp increases in uranium ore prices, it is to be recognized that these reactors are essentially all ^{235}U burners, and ^{235}U, the only fissile isotope that occurs naturally, comprises less than 1% of natural uranium. As the known supplies of high-grade ores diminish, the costs for uranium exploration, mining and processing will increase. It is clear that if a widespread and long-term dependence on nuclear energy is to be realized, at some point it will be more economical to greatly reduce the use of ^{235}U and instead to produce and recycle an artificial fissile isotope. Of the many artificial isotopes that can be produced in quantity, only three can be classified as fissile isotopes: ^{239}Pu and ^{241}Pu, which are produced by the neutron bombardment of the fertile isotope ^{238}U; and ^{233}U, which is produced by the neutron bombardment of the fertile isotope ^{232}Th.

As the nuclear power industry has matured, the major trend has been toward the production and recycling of ^{239}Pu and ^{241}Pu, collectively referred to as Pu^f. This has been a natural development since in most power reactors the ^{235}U fuel is distributed in a matrix of fertile ^{238}U. For example, the heavy-water reactors (HWRs) developed by Canada (called CANDUs) are fueled with natural uranium, which is >99% ^{238}U, and the light-water reactors (LWRs)* developed by the United States are fueled with uranium enriched to only a slightly higher concentration of ^{235}U (3-4%). Thus Pu^f is a byproduct of all these reactor operations by virtue of the constant bombardment of the ^{238}U with reactor neutrons. Under the current U.S. policy of a "once-through" cycle, the Pu^f remains locked in the stored spent

*Commercialized as Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs).

fuel elements, but if the elements were reprocessed, it could be chemically separated and recycled, together with the unburned ^{235}U, in replacement elements. This does not mean, however, that the CANDUs and the LWRs could sustain themselves, since both are *thermal* reactors which for reasons discussed later cannot practically be developed into "breeders" (reactors that produce more fuel than they use). But recycled fuel could be used to supplement the supply of ^{235}U for thermal reactors. Moreover, it has been demonstrated that *fast* reactors* whose fissile cores are surrounded with ^{238}U "blankets" can over a period of time breed enough Pu^f to replace their own fuel and at the same time supplement the fuel supply of thermal reactors. It was, of course, the expectation that this would occur that prompted the U.S. industry's interest in the construction of commercial plants for reprocessing the LWR low-enriched uranium fuels (LEU fuel) and provided the incentive for the development of fast breeder reactors, particularly the Liquid-Metal Fast Breeder Reactor (LMFBR). The evolution of an LMFBR-type fuel cycle, with an increasing concentration of plutonium in the recycle elements, has been referred to as moving toward a "plutonium economy."

While the primary emphasis in the U.S. has been on the uranium-based plutonium cycle (also referred to as the Pu/U cycle), the development of a thorium-based cycle has also been pursued — to the extent that a prototype thermal High-Temperature Gas-Cooled Reactor (HTGR) that contains thorium in its core is already operating (the Fort St. Vrain plant located at Platteville, Colorado). Currently the reactor fuel in the HTGR cycle consists of highly enriched uranium ($\sim$93% ^{235}U in U) intermixed with ^{232}Th (referred to as an HEU/Th cycle), but if the cycle were closed so that the spent fuel elements could be reprocessed, the ^{233}U bred in the ^{232}Th could be extracted and used in new HTGR elements. Thus, the evolution of the closed HTGR cycle could result in an increasing concentration of ^{233}U in the recycle elements, although, again, the HTGR could not practically be developed into a completely self-sustaining reactor.

While our fissile fuel supply could be enhanced by deploying either the Pu/U cycle or the HEU/Th cycle, it has been argued by some groups that neither is as "proliferation resistant" as the currently used once-through LEU cycle. Their concern centers on the fact that in these fuel cycles weapons-usable fissile material is chemically extractable from the fresh fuel elements. The fear is that terrorist or nationalist groups might seize the fresh fuel elements as they are being transported to reactors, or even steal them from the reactor sites themselves, in order to extract fissile material from the elements and fabricate nuclear weapons, however crude. Fresh LEU fuel in the once-through cycle is not considered to be attractive for diversion because any ^{235}U chemically extracted from the fuel would be so diluted with ^{238}U that it would not be usable in weapons fabrication. The uranium would first have to undergo isotopic enrichment, which is technologically difficult and for which few facilities in the world currently exist. These arguments were obviously a major factor in the U.S. Administration's decision in April 1977 to defer commercialization of the Pu/U-fueled LMFBR in the United States.

*Contrary to popular misconception, the word "fast" is not meant to imply a fast breeding rate. Instead "thermal" and "fast" describe the relative energies (speeds) of the neutrons moving within the reactor cores.

Another proliferation concern that has been expressed is that the fissile material extracted when spent reactor fuel elements are reprocessed could be diverted to clandestine weapons-fabrication operations. This, along with other concerns, led the Administration to place a moratorium on reprocessing. As a result, only once-through cycles are used in U.S. commercial power reactors and when plutonium-containing spent elements are removed from the reactors they are stored on site, where they are protected from diversion both by institutional safeguards and by their high fission-product radioactivity. However, as the number of spent fuel elements increases and their radioactivity decreases, more permanent storage arrangements will have to be made.

In contrast to the once-through LEU cycle, both the Pu/U cycle and the HEU(233)/Th cycle mandate reprocessing. Thus there is a point in both cycles at which fissile material would be chemically isolated from all other materials in the spent fuel elements.* Also, the fresh fuel elements in both cycles would contain chemically separable fissile fuel (see Table 1.1). In the Pu/U cycle the fresh fuel would consist of a mixture of plutonium and uranium (Pu^f, plus ^{235}U diluted with ^{238}U) from which weapons-usable Pu^f could be extracted, and in the HEU/Th cycle the fresh fuel would consist of a mixture of uranium and thorium (^{233}U, ^{235}U, a small amount of ^{238}U, and ^{232}Th) from which weapons-usable ^{233}U and ^{235}U, with the ^{238}U, could be co-extracted. (As noted later, however, any fuel containing ^{233}U is radioactive, which would cause handling problems.)

Table 1.1. Comparison of Principal Fissile and Fertile Nuclides in Some Reactor Fuels

Fuel	Fresh Fuel Nuclides[a]	Spent Fuel Nuclides
LEU (no recycle)[b]	^{235}U, ^{238}U	^{235}U, Pu^f, ^{238}U
LEU (with recycle)	^{235}U, Pu^f, ^{238}U	^{235}U, Pu^f, ^{238}U
Pu/U (with recycle)	Pu^f, ^{238}U ((+^{235}U)[c]	Pu^f, ^{238}U (+^{235}U)[c]
HEU/Th (no recycle)[b]	^{235}U, ^{232}Th	^{233}U, ^{235}U, ^{232}Th
HEU/Th (with recycle)	^{233}U, ^{235}U, ^{232}Th	^{233}U, ^{235}U, ^{232}Th

[a] $Pu^f = {}^{239}Pu + {}^{241}Pu$.
[b] "Once-through" system.
[c] Until the cycle becomes self-sustaining, ^{235}U will be included.

With the above objections in mind, several groups have offered "alternative" nuclear fuel cycles which they view as being more proliferation resistant than either the Pu/U cycle or the HEU/Th cycle. For uranium-based cycles, these alternatives range from making the once-through LWRs more uranium efficient to implementing the Pu/U recycle mode with "full-scope" safeguards. For thorium-based cycles, the alternatives are aimed at making ^{233}U recycle more acceptable.

*It has been suggested that deliberately "spiking" or otherwise contaminating the fissile material with radioactive materials would discourage diversion.

One thorium-based alternative that has been proposed is the cycle now commonly referred to as the "denatured ^{233}U" cycle. Suggested in 1976 by Feiveson and Taylor,[1] a research team in the Program on Nuclear Policy Alternatives at Princeton University, the cycle would be structured so that its fresh fuel would have the same isotopic barrier that exists in LEU fresh fuel. That is, the ^{233}U would be mixed with ^{238}U, with the ^{233}U concentration kept at a level sufficiently low for the mixture to be unusable for weapons fabrication. The ^{238}U content would be limited to that required to "denature" the ^{233}U, the remainder of the fuel being comprised of ^{232}Th so that additional ^{233}U would be bred during reactor operations.

While Feiveson and Taylor outlined their concept of the full cycle, they made no attempt to detail a specific cycle nor to provide a technical assessment of the capabilities of power systems utilizing denatured ^{233}U fuel. However, with interest in thorium-based cycles increasing, the Department of Energy in 1977 initiated a multi-institutional study of the denatured ^{233}U cycle which concentrated on the following areas:

(1) The isotopics of denatured ^{233}U fuel, particularly as they provide inherent proliferation-resistance characteristics or impact the design of the fuel cycle;

(2) The structure of a denatured ^{233}U fuel cycle, i.e., the types of reactors and fuels that would be included, the support facilities that would be required, and the relative locations of the various components;

(3) The impact of denatured ^{233}U and other fuels in the cycle on the performance of the reactors;

(4) The technical and economic feasibility of commercially deploying the denatured ^{233}U fuel cycle; and

(5) The adequacy of postulated nuclear power systems utilizing denatured fuel for meeting power demands.

Because a nuclear data base for the denatured ^{233}U cycle was largely nonexistent, and the designs of the reactors in which denatured fuel would be used were also incomplete, the results of the study are necessarily preliminary. However, as will be apparent from the following summary, many institutional and technical requirements for implementing the denatured ^{233}U fuel cycle have been clarified and a broad view of its possibilities and limitations has been provided.

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[1]H.A. Feiveson and T.B. Taylor, "Security Implications of Alternative Fission Futures," *Bull. Atomic Scientists*, p. 14 (December 1976).

2.0. ISOTOPICS OF DENATURED ^{233}U FUEL: NONPROLIFERATION AND FUEL CYCLE IMPLICATIONS

As pointed out above, "fresh" denatured ^{233}U fuel* would consist of the fissile isotope ^{233}U diluted with the fertile isotope ^{238}U and mixed with the fertile isotope ^{232}Th. In order to minimize the production of plutonium and concomitantly to maximize the production of ^{233}U during reactor operations, the amount of ^{238}U denaturant used would be limited to that required to provide an effective isotopic barrier. That is, it would be determined by the allowable concentration of ^{233}U in ^{238}U. Just what the exact percentage of ^{233}U in the denatured uranium would be has not been firmly established, but estimates have been made on the basis of limits set for ^{235}U. Fast critical mass data for ^{235}U indicate that uranium metal containing less than 20% ^{235}U is unsuitable for weapons fabrication; therefore, this percentage has been set as the dividing line between low-enriched and high-enriched ^{235}U fuel.† Calculations that compare the infinite neutron multiplication factors of ^{233}U and ^{235}U, both as metals and as oxides, indicate that a comparable limit for ^{233}U would be between 11 and 12%. Thus in this study the upper limit for the enrichment of denatured fuel was set at 12%. At this enrichment the uranium would comprise approximately 22% of the fuel mix (2.6% ^{233}U and 19.4% ^{238}U), and ^{232}Th would comprise approximately 78%.

Fresh denatured ^{233}U fuel would also contain the very important isotope ^{232}U.§ While not contributing either to the energy production or to the fuel production, this isotope would be present because it is unavoidably produced along with the ^{233}U and can be isolated from the ^{233}U only by a difficult and costly isotopic separation process. Such a step would appear to be unwarranted since the ^{232}U would exist only in small concentrations and would not affect the operation of the reactor *per se*. Its importance stems from the fact that ^{232}U is an unstable isotope that emits radiation as it decays through ^{228}Th and its daughter products to stable ^{208}Pb, the most prominent emissions being 2.6-MeV gamma rays emitted in the decay of ^{208}Tl. Thus fresh denatured ^{233}U fuel would be radioactive. Moreover, as more and more of the ^{232}U decayed, the radioactivity of the fuel would increase in intensity for some time before peaking and eventually decreasing.

Spent denatured ^{233}U fuel would, of course, contain all of the isotopes included in the fresh fuel (^{233}U, ^{232}U, ^{238}U, and ^{232}Th) plus the Pu[f] produced in the ^{238}U. In addition, the spent fuel would contain several other isotopes that are present in the nuclide production chains of ^{238}U and ^{232}Th, one of particular interest being ^{233}Pa. Finally, the spent fuel would contain fission products, which because of their high radioactivity would generate gamma-ray fields orders of magnitude larger than those produced by the ^{232}U chain.

The isotopics of the denatured ^{233}U fuel cycle are unique in that they offer several inherent barriers to fuel diversion by terrorists or nationalist states. At the same time they introduce complications in the design of the cycle, as will be apparent from the following discussion.

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*As used here, "fresh" fuel is any fuel prepared for insertion into the reactor, regardless of the number of times the fissile material has been recycled.

†In general, however, "low enriched ^{235}U fuel" implies 3-4% enrichment and fuel enriched to about 20% is considered to be medium enriched uranium (MEU).

§The processes whereby ^{232}U is produced and subsequently decays are shown in Appendix A.

2.1. Nonproliferation Advantages

Isotopic Barrier of Fresh Fuel

The isotopic barrier provided by the ^{238}U denaturant is the primary nonproliferation feature of the denatured ^{233}U fuel cycle, the premise being that the isotope separation (enrichment) facilities required to upgrade the uranium to weapons material would not be available to subnational terrorist groups and, with some possible exceptions, probably would not be available to non-nuclear nationalist states.

While the isotopic barrier is formidable, it is not absolute, especially since enrichment technology is currently undergoing rapid development. Whereas 10 years ago all enrichment operations were performed at large gaseous diffusion plants, today the gas centrifugation technique is practical in small-scale plants and could be applied to both ^{233}U and ^{235}U fresh fuels. Relatively speaking, ^{233}U fuels would be easier to enrich than ^{235}U fuels. For example, because of its lower mass, ^{233}U would be more easily separated from ^{238}U than ^{235}U would be, assuming equal enrichments of the feed material. Also, because ^{233}U has a lower fast critical mass and thus a smaller amount would be needed, less enrichment capacity would be required to produce a weapons worth of ^{233}U from $^{233}U/^{238}U$ feed than to produce a weapons worth of ^{235}U from $^{235}U/^{238}U$ feed, again assuming equal enrichments of the feed material. And finally, less effort would be required to upgrade 12% ^{233}U material to 90% enrichment than would be required to upgrade 3-4% ^{235}U material (such as LWR-LEU fuel) to 90% enrichment.

In considering the enrichment of diverted fresh fuel, however, it is to be recognized that the current status of centrifuge technology is such that much advanced planning and long and undetected operations would be necessary to enrich a sufficient amount of weapons-grade material from any low-enriched fuel. Moreover, as the concentration of ^{233}U in the uranium increased, so also would the concentration of ^{232}U increase, resulting in a highly radioactive product unless the isotopic separation included the removal of ^{232}U. To completely remove the ^{232}U would require an increased centrifuge capacity, the total for ^{233}U recycle fuels approaching 50% to 90% of that required to enrich LEU fuel (3.2 wt% ^{235}U). Thus the problems encountered in enriching denatured ^{233}U fuel would appear to be sufficiently difficult to suggest that some other fuel might be a better choice. On the other hand, it also must be recognized that enrichment technologies that will exist in the next 20 to 25 years will be considerably advanced over the current technologies. Moreover, if means were available for handling the radioactive fuel — that is, if the ^{232}U contamination were acceptable — then the centrifuge capacity required to enrich the denatured ^{233}U fuel to weapons grade would be reduced to only 3% to 20% of the capacity required to enrich LEU fuel.

Radiation Barrier of Fresh Fuel

The ^{232}U-induced gamma activity in denatured ^{233}U fuel will constitute an effective radiation barrier against seizure of the fresh fuel. At the time this study was performed, the nuclear data required for calculating the concentrations of ^{232}U in denatured fuels (usually characterized as so many parts per million) were not sufficiently developed for accurate calculations to be performed; however, estimates have been made for some fuels with the data at hand. The results range from approximately 40 ppm ^{232}U in U for denatured HTGR fuel (after equilibrium recycle) to about 1600 ppm ^{232}U in U for recycled denatured LMFBR fuel.

In order to estimate the deterrence value of the ^{232}U content of the fresh fuel, it is necessary to correlate the ^{232}U concentrations with gamma-ray dose rates and the dose rates in turn with potential harm to would-be diverters. The highest level of deterrence, of course, would be provided by a gamma-ray dose that is immediately incapacitating (greater than 10,000 rem). While several factors must be considered, including the quantity of material being handled, the maximum dose rates that could be expected from fresh denatured fuel (fast-reactor bred material) would be on the order of 100 rem/hr, and thus disabling doses would not occur. However, doses in the range of 200 to 600 rem can cause eventual death, and the 100 rem/hr dose rate could discourage diversion by all except those individuals who were either disdainful of or ignorant of the fact that they were risking exposure to lethal doses.

If the fuel were successfully seized in spite of the radiation barrier, it would not be useful for weapons fabrication unless it was enriched, which would be difficult (as discussed above), especially if the ^{232}U were removed in the process. If the ^{232}U were not removed, the gamma activity of the enriched product would be proportionately higher, the fraction of ^{232}U in LMFBR-derived denatured fuels increasing to approximately 8000 ppm. Chemical processing might be employed after the enrichment process to remove the ^{232}U decay products that are the actual gamma-ray emitters; however, within 10 to 20 days further decay of the ^{232}U would provide a new population of ^{228}Th and its daughters. Thus, no advantage would be gained unless a highly accelerated schedule could be followed.

If the ^{232}U were not removed in the enrichment process, fabrication of a weapon with the resulting contaminated product presumably could be done by remote operation; however, construction and/or acquisition of the shielding, remote handling equipment, etc. would increase the risk of detection of a covert program before its completion. And while nonfissile material included in the weapon would provide some shielding during its delivery, additional shielding would be required to protect the operator of the delivery vehicle and to facilitate the loading operations. Thus, the radiation barrier would present problems throughout the entire diversion process. By contrast, fresh mixed oxide Pu/U fuel would present a much smaller radiation problem and the currently employed fresh LEU fuel would present essentially none at all.

In addition to discouraging diversion *per se*, the presence of the 2.6-MeV gamma ray emitted from ^{233}U-containing fuels would provide a useful handle for detecting material both during and after diversion. Adequate detection systems that could be adapted for this purpose are already available. In particular, a monitor system developed at Los Alamos Scientific Laboratory is capable of measuring a dose rate of about 2.5 mr/hr at a distance of 30 cm from a 20-g sample of PuO_2. Approximately the same dose rate would be measured for a similar sample of ^{233}U containing 100 ppm ^{232}U 12 days following chemical extraction of the daughter products, and, of course, the dose rate would increase manyfold as the daughter population built up again. Also, the efficiency of the detector could be improved if the detector window were set to cover only the strong 2.6-MeV gamma ray in the spectrum. Thus, with respect to detectability, the radioactivity of the fuel would be a definite advantage.

Reduced Attractiveness of Spent Fuel

Like all spent reactor fuel elements, spent denatured ^{233}U fuel elements would contain fissile material, some of which would be chemically separable (see Table 1.1); however, the elements would be protected from diversion, at least initially, by their high fission-product radioactivity.

In addition to unburned ^{233}U, which would not be chemically separable because of the ^{238}U denaturant, the spent fuel would contain Pu^f produced in the ^{238}U during reactor operations. If a terrorist or nationalist group could arrange to seize and process the fuel elements after they had decayed to a manageable radiation level, or if they could devise a processing system that could be operated remotely or semiremotely, then the plutonium could be chemically separated. However, the choice of denatured fuel for this purpose seems highly unlikely since the amount of ^{238}U included in the denatured elements would be only about one-fifth the amount included in LEU elements. Thus to extract a given amount of plutonium from spent denatured elements would require processing more elements than would be necessary to obtain an equivalent amount of plutonium from LEU elements.*

Spent denatured ^{233}U would also contain ^{233}Pa, which is an intermediate isotope in the ^{233}U production process initiated by the $^{232}Th(n,\gamma)$ reaction. ^{233}Pa decays to ^{233}U, and since it has a relatively long half-life (27.4 days), theoretically it could be chemically separated from the spent fuel and allowed to decay to ^{233}U after the separation. However, the chemical separation would have to be initiated shortly upon discharge of the elements from the reactor while the fission-product radioactivity is very intense, which is highly improbable. Moreover, the discharge concentration of ^{233}Pa is low (typically 5% of the discharge ^{233}U), which means a large quantity of heavy metal would have to be processed to recover a significant quantity of ^{233}Pa. Even then the amount recovered would be comparable to the amount of plutonium that could be recovered after the elements had cooled. Therefore, it would seem that if any diverter group were to seize spent denatured fuel for its fissile content, plutonium would be the choice, even though, as noted above, the plutonium content of the denatured fuel would be relatively small.

Reduced Accessibility of Isolated Fissile Isotope

The intense activity associated with ^{233}U would require that the fuel fabrication process be performed remotely behind several feet of concrete. While this would introduce complications in the fuel cycle development (see below), the remote nature of the process would provide an additional safeguard feature in that access to the fissile material would be severely restricted.

2.2. Fuel Cycle Impact

The presence of the gamma-emitting ^{232}U chain in denatured ^{233}U fuel would adversely affect the design of the fuel cycle in several important ways. First it would effectively preclude nondestructive assays (NDA) of the fuel because the gamma-ray signals from the fuel would be dominated by the ^{232}U decay gamma rays that could not be properly accounted for without a detailed history of the sample. Also, the desired signal would be reduced by the ^{238}U dilution.

*The number of elements varies with the type of reactor; for PWRs about three times as many denatured ^{233}U elements would be required.

More importantly, the requirement for remote fabrication of the fuel would necessitate significant modifications to the uranium oxide pellet fabrication process that is currently employed in fabricating LEU fuels and is planned for Pu/U fuels. It is possible that a continuous process, such as the sphere-pac process, in which liquids and microspheres are more easily handled remotely, would be required. In a continuous process, the usual accountability techniques, in which the control of fissile material is based on tracking individual "batches," could not be applied.

The remote operations would be required, of course, to provide protection for operating personnel against the gamma rays emitted by the ^{233}U fuel. Other potential radiological hazards are associated with the emission of alpha and beta particles by the fuel isotopes, and to the extent that the denatured fuel isotopes differ from those in other cycles, these hazards might also require special consideration.

To cause serious damage, alpha and beta emitters must be inhaled or ingested in the body, since the penetration ranges of the particles are so short that even if the radionuclides emitting them were deposited on the skin, the alpha particles would not penetrate the skin and the beta particles would give no more than skin doses. As a result, no serious hazard would exist from these particles unless the fuel were inadvertently dispersed in the environment. In that event, however, the radionuclides might be inhaled or ingested and subsequently migrate to critical organs where the *in situ* emission of the alpha and beta particles would cause damage of body tissue. While this is not expected, the potential toxicity of the various fuel isotopes must be considered.

The toxicity of any particular radionuclide is determined by several factors, including its specific activity (the number of disintegrations per second), the effective energy deposited in the organ per disintegration, the effective half life of the isotope,* and the critical body organ in which it is deposited. In general, the heavy metal isotopes in reactor fuels are bone seekers, and in the "worst case" scenarios, they would be inhaled. (The dose from inhaled heavy metal isotopes is orders of magnitude larger than the dose from ingested heavy metal isotopes.)

Estimates of the time-integrated doses (50-year doses) that can be expected to be delivered to the bone by the inhalation of the important fuel isotopes† show that in terms of dose per microgram of the isotope inhaled, ^{232}U has a higher toxicity than any other uranium or plutonium isotope except ^{238}Pu. ^{233}U also has a relatively high toxicity (higher than ^{235}U or ^{238}U), but it is considerably lower than the toxicity of any of the plutonium isotopes.

Obviously an important factor in estimating the potential danger posed by an environmentally dispersed fuel is the relative amount of each isotope in the fuel. For example, the fraction of the contaminant ^{232}U in denatured ^{233}U fuel would be much lower than the combined fraction of plutonium isotopes in Pu/U fuel — to the extent that the Pu/U fuel

*The half life of an isotope is the period of time required for one-half of a given quantity to disintegrate, that is, for one-half of the nuclei to change form.

†See Appendix B.

would be considerably more toxic. Also, since denatured ^{233}U fuel would be diluted with relatively nontoxic ^{238}U, it would contain proportionately less ^{232}U than highly enriched HTGR fuel and therefore would be somewhat less toxic than the HTGR fuel. On the other hand, it would be considerably more toxic than LEU fuel.

The toxicity of the fertile isotope ^{232}Th included in the ^{233}U-containing fuels is not considered because of the overriding importance of ^{232}U, which has a dose commitment to the bone that is more than four times greater. However, the radiological hazards associated with mining of U.S. thorium deposits are directly attributable to ^{232}Th and its decay products, and thus must be considered in the fuel cycle development.

2.3. Conclusions

In conclusion, the nonproliferation advantages and the impact on the fuel cycle design attributable to the isotopics of denatured ^{233}U fuel can be summarized as follows:

- Isotopically denaturing ^{233}U fuel with ^{238}U would provide a significant technical barrier to ^{233}U isolation (although not an absolute one) that would decrease with time at a rate that would be country-specific. Countries that have the technological expertise to develop isotope separation capabilities would have the technology required to circumvent this barrier; however, they probably would also have the option of utilizing natural uranium or low-enriched ^{235}U fuel as feed material for the enrichment process.
- Denatured ^{233}U fuel would have an inherent gamma radiation barrier due to ^{232}U daughter products in the fresh fuel that would significantly increase the effort required to obtain weapons-usable material from diverted fresh fuel. Moreover, the gamma rays emitted by the fuel would provide a useful handle for detecting the fuel during or after diversion.
- While the production of plutonium in the spent denatured fuel would represent a potential proliferation concern, the amount of plutonium in the denatured spent elements would be less than that in LEU spent elements. It therefore seems unlikely that spent denatured fuel would be diverted for its plutonium content. Other fissile constituents of the spent denatured elements would be even less attractive.
- The radioactivity of the denatured ^{233}U fuel would necessitate that all fuel cycle operations involving the fuel be designed for remote operation. This would provide an additional nonproliferation advantage by severely restricting accessibility to fissile material. However, it would complicate the development of the fuel cycle. In particular, the requirement for remote fuel fabrication might impose constraints on the use of uranium oxide pellets. Also, the radioactivity of the fuel would prevent the usual methods for nondestructive assays of the fuel from being applied.
- In designing the denatured ^{233}U fuel cycle, the toxicities of ^{232}U, ^{233}U, and ^{232}Th would have to be thoroughly established and taken into account.

3.0. STRUCTURE OF THE DENATURED ^{233}U FUEL CYCLE

The preceding discussion offers the denatured fuel cycle as a proliferation-resistant alternative to the Pu/U cycle for producing and recycling an artificial fissile isotope — ^{233}U — in power reactors. For thermal reactors, such as those that dominate today's nuclear power systems, ^{233}U is a particularly efficient fuel. When the fissioning neutrons are thermal neutrons, ^{233}U releases more energy per atom of fuel destroyed (by fission or transmutation) than either ^{239}Pu or ^{235}U, and it also produces more neutrons per atom of fuel destroyed than the other fissile isotopes. Thus for equivalent amounts of fuel, thermal reactors utilizing ^{233}U not only would generate more energy than thermal reactors operating on other fuels but also would have more excess neutrons available for breeding additional fuel.

^{233}U is less attractive for fast reactors. The impetus for developing fast reactors has always been due to their potential role as breeders, which requires a high production of excess neutrons, and when the fissioning neutrons are fast neutrons the neutron production of ^{233}U is well below that of ^{239}Pu. As a result, Pu-fueled fast reactors surrounded by fertile ^{238}U blankets have always been favored for breeder designs rather than ^{233}U-fueled fast reactors surrounded by fertile ^{232}Th blankets. While the latter are theoretically feasible, their production of excess fuel would be low or even marginal.

The relative ineffectiveness of ^{233}U-fueled fast reactors as ^{233}U breeders means, of course, that they could not be depended upon to produce an adequate supply of ^{233}U for the denatured fuel cycle. Nor could ^{233}U-fueled thermal reactors, since, as discussed earlier, state-of-the-art thermal reactors cannot practically be developed as breeders regardless of the fissile isotope they utilize.† On the other hand, a class of thermal reactors that are now being considered for development and are referred to as "advanced converter reactors (ACRs)" are expected to have breeding (conversion) ratios* that would reduce their requirements for an exogenous source of fissile fuel, which means that they would probably be good candidates for operation on denatured ^{233}U fuel. Still, some makeup ^{233}U would be required, and since none of the reactors appear to be capable of burning ^{233}U and simultaneously producing an excess of ^{233}U, reactors utilizing some other fuel would have to be developed as ^{233}U producers. Thus, implementation of the full denatured ^{233}U fuel cycle would require the deployment of different types of reactors operating on different types of fuels, and it could not be expected that all the fuels would have inherent proliferation-

†Although not considered in this study, current efforts are under way to develop a Light Water Breeder Reactor (LWBR) that utilizes and produces ^{233}U.

*The breeding ratio and the conversion ratio are both defined as the ratio at a specific point in time of the rate at which fissile material is produced in a reactor (by excess neutrons not required to sustain the fission process) to the rate at which fissile material is destroyed in the reactor. The term breeding ratio is used for those reactors for which the ratio is greater than 1 (as for fast breeders), and conversion ratio is used for those for which the ratio is less than 1. If the ratio is greater than 1, then, at least theoretically, the reactor is producing enough fuel to sustain itself.

resistance characteristics. As a result, technical and/or institutional barriers would be necessary to ensure that the fuels without an inherent resistance were not subject to diversion.

Similar barriers would also be required, of course, for the various steps in the fuel cycle at which fissile material would be isolated. Thus the structure of the denatured fuel cycle — that is, the locations of the various components in the cycle and their attendant technical and institutional barriers — would be a major consideration.

3.1. Reactor Types Used in Denatured Fuel Cycle

Just as the initial recycling of plutonium would be in LWRs already in operation, the initial use of denatured ^{233}U fuel would be in LWRs, followed by its introduction into other types of thermal reactors. In this study it has been assumed that the other types of thermal reactors would be the ACRs that are currently receiving attention as systems with a potential for significantly improved fuel utilization characteristics. These reactors are primarily based on three design concepts: the gas-cooled HTGR, which, as mentioned earlier, has already passed the prototype stage; the pressurized heavy-water CANDU, which has been commercialized by Canada; and the Spectral-Shift-Controlled Reactor (SSCR), which is basically a pressurized-water reactor (PWR) whose reactivity control system utilizes heavy water instead of soluble boron to compensate for long-term reactivity changes during the operating cycle.

While to date the ACRs have not yet been demonstrated on their own reference fuels, their feasibility appears assured (see Section 5) and their adaptation to denatured ^{233}U fuel would not require major alterations to their designs. Thus the addition of denatured ACRs to the denatured LWRs would no doubt improve the overall fuel efficiency of the cycle. However, since none of these reactors would operate in self-sustaining modes, they would each require an exogenous source of ^{233}U. For reasons stated above, the ^{233}U would undoubtedly be produced by a reactor that burned some fuel other than ^{233}U.

The obvious (and only) long-term choice for the fuel in a ^{233}U producer is the other artificial fuel — plutonium. And since ^{238}U would be included in the denatured fuel elements, plutonium would be available in the cycle. Feiveson and Taylor suggested that a plutonium-fueled fast reactor with a ^{232}Th blanket might be used as a ^{233}U-production device, and to distinguish this type of reactor from the classical fast breeder, the term "transmuter" was coined at ORNL. However, a thermal reactor that is fueled with plutonium and contains ^{232}Th within its core could also be a ^{233}U producer, and this type of reactor, whether based on an LWR design or one of the ACR designs, is also being referred to as transmuter. In any case, utilizing plutonium-fueled reactors in the system would provide a means for disposing of the plutonium produced in the fuel cycle, which would be a nonproliferation advantage.

The principal reactors in the denatured fuel cycle then would be thermal reactors operating on denatured ^{233}U fuel (LWRs and possibly ACRs) and thermal and/or fast reactors utilizing plutonium to produce ^{233}U (i.e., transmuters). Depending on the relative requirements for ^{233}U and plutonium, it might also be necessary for the cycle to include plutonium-fueled "breeder-transmuters," which would contain both ^{232}Th and ^{238}U in their

blankets. And for some energy scenarios the classical fast breeder might also be required, although a constraint in the overall system would be that the production of plutonium would not exceed the demand for it so that the net production of plutonium would be zero. Finally, depending on the demand for power, the cycle could also require fast reactors that would operate on denatured fuel and contain ^{232}Th in their blankets. As will be discussed more fully in Section 4, these "denatured fast breeder reactors" would produce more fissile fuel than they would consume, but the fuel they produced would consist of ^{233}U *plus* ^{239}Pu. The ^{239}Pu could not be recycled in the denatured reactor, and the amount of ^{233}U produced would not be sufficient to replace the ^{233}U consumed. Thus the denatured ^{233}U fast reactor would also require an exogenous source of ^{233}U. Even so, this type of "breeder" could be used advantageously.

3.2. Other Components of Fuel Cycle

With the requirement for reactors fueled both with plutonium and with denatured ^{233}U, the full denatured ^{233}U fuel cycle would include features of both the conventional plutonium cycle and the conventional thorium cycle. For example, facilities would have to be available for reprocessing uranium/plutonium fuels by the already well-developed Purex process. In addition, facilities would have to be available for reprocessing uranium/thorium fuels by the Thorex process, or, depending on the form of the fuel elements, with a modified version of the Thorex process. And, of course, fuel fabrication facilities for the two types of elements would be needed, as well as waste storage facilities.

3.3. Locations of Fuel Cycle Components

Because the plutonium-fueled reactors and several other fuel cycle components and/or operations would not have inherent proliferation-resistant characteristics, they would require special protection. The proposal by the Princeton team, and others before them,[2] is that the facilities falling in this category be centrally located in secure (guarded) energy parks. The reactors operating on denatured ^{233}U fuel would, of course, be dispersed outside the parks to locations where they were needed for producing power.

A schematic indicating the locations of the various components in a typical power system operating on the denatured ^{233}U fuel cycle is shown in Fig. 3.1. The transmuters and other sensitive support facilities are confined to the energy center and the fuel assemblies transported outside the center are limited to fresh denatured assemblies. All the plutonium produced in the dispersed reactors is returned to the center in the highly radioactive spent fuel elements. The plutonium is then extracted and burned in the transmuters inside the center to produce ^{233}U for the outside reactors. As a result, no "fresh" plutonium exists outside the energy center. The ^{233}U produced outside the center is similarly returned to the center in the radioactive spent fuel elements, or, if denatured fast breeder reactors are used, in radioactive blanket elements.

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[2]"A Report on the International Control of Atomic Energy," prepared for the Secretary of State's Committee on Atomic Energy by a Board of Consultants: Chester I. Barnard, Dr. J. R. Oppenheimer, Dr. Charles A. Thomas, Harry Winne, and David E. Lilienthal (Chairman), Washington, D. C., March 16, 1946, pp. 127-213, Department of State Publication 2493.

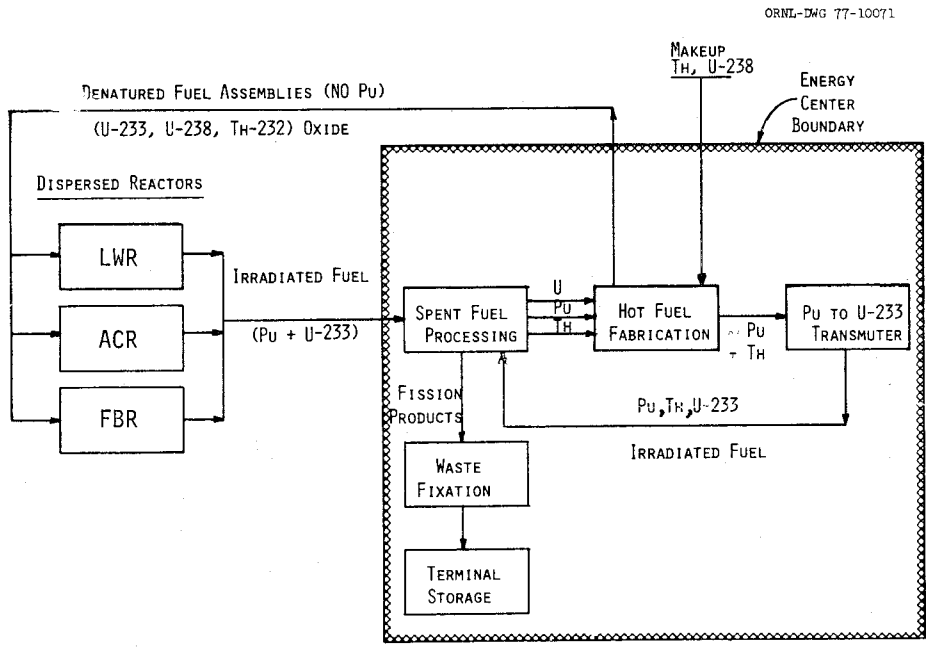

Fig. 3.1. Schematic of Nuclear Power System Consisting of an Energy Center and Dispersed Reactors Operating on Denatured ^{233}U Fuel.

While not indicated in Fig. 3.1, the power system would also include, an least initially, LWRs operating on the conventional "naturally denatured" LEU ^{235}U cycle, with the plutonium produced in the spent fuel elements being recycled within the center. Another possibility is that LWRs operating on MEU(235)/Th fuel might be included. In MEU(235)/Th fuel the ^{235}U enrichment would be increased above the 3 to 4% in LEU fuel but would remain below the 20% limit that has been set as the dividing line between low and high enrichment for ^{235}U. The increased enrichment would allow some of the ^{238}U to be replaced with ^{232}Th, compensating for the lower fission cross section of ^{232}Th relative to ^{238}U.* (Note: Even if the decision to utilize ^{233}U fuels is deferred, introducing thorium into currently operating LWRs would be a means for initiating a stockpile of ^{233}U and at the same time reducing the production of plutonium by such reactors. However, as discussed in Sections 4 and 5, this would entail a significant economic penalty.)

Also not indicated in Fig. 3.1 is the possibility that ACRs operating on some fuel other than denatured ^{233}U fuel might be included in the system. Those operating on low-enriched ^{235}U fuel could be used as dispersed reactors while those operating on highly enriched fuels or plutonium would be included in the energy centers. Finally, Fig. 3.1 does not indicate the possible inclusion of the classical breeders within the energy center.

*Neutrons produced by "fertile fissions" always contribute to the chain reaction, and thus the relative contributions by ^{238}U and ^{232}Th fissions must always be taken into account when ^{232}Th is substituted for ^{238}U in reactor cores.

3.4. Symbiotic Character of Denatured Fuel Cycle

The denatured ^{233}U fuel cycle thus would evolve into a system in which the reactors outside the energy center and those inside the center would be operating "in symbiosis." When the system reached maturity, no external source of fissile material would be supplied and the system would be self-contained.

Obviously, the inherent growth potential of a given power system would depend on the types of reactors it utilized, that is, whether or not they were net fissile consumers (thermal reactors) or net fissile producers (fast reactors). The greater the fraction of fast reactors, the greater the growth potential. In addition, however, the growth potential would be intimately tied with the ratio of power produced outside the center to the power produced inside the center, which is defined as the "energy support ratio."

Although any number of reactor mixes can be envisioned, three generic types of symbiotic systems are illustrative of the inter-relationship of the system growth potential and its energy support ratio. The generic systems can be described as (1) dispersed thermal reactors supported by energy-center thermal transmuters, (2) dispersed thermal reactors supported by energy-center fast transmuters, and (3) dispersed fast reactors supported by energy-center fast transmuters. System 1 would have no net fissile gain and thus its growth potential would be inherently negative and its installed nuclear capacity would decay as a function of time. System 2 would have a potential for growth because it includes fast reactors; however, a tradeoff between the support ratio and the growth rate clearly would exist for this system since maximizing the support ratio would mean that the thermal reactors would comprise the major fraction of the system and the growth rate would be detrimentally affected. System 3 would provide much more flexibility in terms of the allowable energy support ratio and inherent growth rate. Thus the design of a power system utilizing denatured ^{233}U fuel would be highly dependent not only on the power demand but also on the locations where the power is to be delivered.

3.5. Conclusions

The preceding discussion can be summarized by the following:

- Denatured ^{233}U fuel would be used initially in LWRs and subsequently could be introduced into advanced converter reactors (ACRs). Reactor types primarily being considered for development as ACRs are HTGRs, CANDUs, and SSCRs. FBRs might also be adapted for use with denatured ^{233}U fuel.
- Reactors operating on denatured ^{233}U fuel would be unable to sustain themselves and thus would require an exogenous source of ^{233}U. The most likely devices for ^{233}U production would be "transmuters" — reactors (thermal or fast) that burn plutonium to produce ^{233}U in fertile ^{232}Th. The plutonium produced in the ^{238}U denaturant could be used to fuel the transmuters.

- The denatured reactors and the transmuters would operate "in symbiosis," each producing the fuel needed by the other. In a mature system, no external fissile source would be required, although to reach this stage other types of reactors might have to be added (e.g. classical breeders).
- The Pu-fueled reactors and other components in the cycle lacking inherent proliferation-resistance characteristics would be constrained to a secure (guarded) energy center. Other components — particularly the reactors operating on denatured ^{233}U fuel — would be dispersed outside the energy center to locations where they were needed.
- The symbiotic nature of the denatured ^{233}U fuel cycle would mandate a tradeoff analysis of growth potential versus energy support ratio (ratio of power produced outside the energy center to the power produced inside the center). For thermal/thermal systems (thermal reactors inside center/thermal reactors outside center), the growth potential would be negative. Fast/thermal systems would permit some of the net fissile gain (i.e., growth potential) of the fast reactors to be sacrificed for a higher energy support ratio. Fast/fast systems would provide the highest growth potential.

4.0. IMPACT OF ALTERNATE FUELS ON REACTOR PERFORMANCE

The denatured ^{233}U fuel cycle has been described as a symbiotic system of reactors operating on several types of fuels — which may or may not include the fuels for which the reactors were originally conceived. Since each reactor type is specifically designed to operate on its own reference fuel, it can be assumed that most types would not perform equally well on "alternate" fuels. As a result, to produce the same amount of power, a reactor operating on an alternate fuel probably would require an increased fissile charge, which in turn would require increased U_3O_8 ore and separative work* (enrichment) units, either directly for its own fuel or indirectly to supply a supporting system. If the requirements are excessively high, then, of course, the use of that reactor-fuel combination in the denatured fuel cycle must be discounted.

Presumably the redesign of a reactor would improve its performance on an alternate fuel, but redesign without some preliminary indication of the feasibility of a particular reactor-fuel combination would be unrealistic. Therefore, first estimates of the impact of an alternate fuel on the performance of a reactor are made by performing "mass flow" calculations for the as-designed reactor operating on that fuel. The results are given in terms of fissile fuel charges and discharges, the U_3O_8 and enrichment requirements, conversion (or breeding) ratios, etc. Prior to the initiation of this study, such calculations were already being performed by various organizations, and pertinent results for reactor-fuel combinations of interest to the denatured ^{233}U fuel cycle were collected for this study. The rationale whereby specific combinations were then selected as components for postulated "denatured power systems" is described below.

4.1. Alternate Fuel Types Considered

In examining the performance of reactors operating on alternate fuels, it is useful to distinguish between two generic fuel cycle types: those in which the spent fuel is reprocessed concurrently (that is, *recycle systems*) and those in which the spent fuel is not reprocessed concurrently *(once-through systems)*.

All once-through systems must, of course, utilize the resource base since ^{235}U is the only naturally occurring fissile isotope and thus the only fissile isotope available without reprocessing. The most well-known once-through cycle is the LEU cycle (low enriched uranium) used in LWRs. Variations of this cycle are the natural-uranium cycle currently used in HWR-CANDUs and an SEU cycle (slightly enriched uranium) proposed for the advanced CANDU.

Another possible once-through cycle is the MEU(235)/Th cycle, sometimes referred to as the "denatured ^{235}U" cycle because the fuel composition is analogous to that of denatured ^{233}U fuel [i.e., MEU(233)/Th fuel]. As will be shown below, however, it probably would not be economic to introduce MEU(235)/Th fuel into reactors unless the fuel cycle was assumed to be in a *stowaway* mode — that is, unless it was planned to recover the ^{233}U

*Separative work units (SWUs) are dimensionless and are used to show the relative amounts of effort required to enrich various fuels.

produced in the spent fuel at some later date, in which case subsequent (but not concurrent) reprocessing would occur. By contrast, once-through cycles operating on a *throwaway* mode do not include plans for recovering the unburned uranium (^{235}U or ^{233}U) or plutonium.

Because the denatured ^{233}U fuel cycle mandates the recovery and recycling of the ^{233}U, it cannot be used in a once-through mode. Neither, of course, can any of the cycles in which the primary fuel, or even the "topping" fuel, is plutonium. Thus, whenever ^{233}U or plutonium is used in a reactor - thermal or fast - the system is a recycle system. And since plutonium is the only fuel seriously considered for fast reactors (although denatured ^{233}U might also be used), all fast-reactor systems are recycle systems. It follows, of course, that all once-through systems are thermal systems.

When fuels are being considered for use in proliferation-resistant power systems based on secure energy centers, they must also be classified as *dispersible* or *energy-center-constrained*, the dispersible fuels being those that have inherent proliferation-resistant characteristics. Dispersible fuels would include all fuels in which the ^{235}U enrichment is maintained below the 20% limit or the ^{233}U enrichment is maintained below the 12% limit. LEU fuel, the MEU(235)/Th and MEU(233)/Th fuels, SEU fuel and natural uranium would all be dispersible fuels. Conversely, fuels containing highly enriched ^{235}U or ^{233}U or plutonium would be energy-center-constrained fuels, e.g., the HEU(235)/Th and HEU(233)/Th fuels and the Pu/U and Pu/Th fuels. Viewed from this perspective, it is apparent that the energy support ratio of a power system, which has been defined as the ratio of the power produced outside the center to the power produced inside the center, is equivalent to the ratio of the power produced by reactors operating on dispersible fuels to the power produced by reactors operating on energy-center-constrained fuels.

4.2. Reactor Designs Calculated

The thermal reactors for which mass flow calculations were collected include the LWRs (both PWRs and BWRs), the three types of reactors under primary consideration as advanced converters (HTGRs, HWRs, and SSCRs), and a gas-cooled reactor identified as the Pebble Bed Reactor (PBR). The fast reactors are the standard LMFBR with its homogeneous core, plus an "advanced" LMFBR in which some blanket assemblies are intermixed with fuel assemblies (heterogeneous core).

In each case the reactor design used for the analysis was a current design optimized for the reactor's *reference* fuel, and thus reactor performance improvements that could result from redesign to accommodate the alternate fuels are not reflected. The assumptions included a 75% plant capacity factor, a 0.2 wt.% ^{235}U content in the uranium tails, a 0.5% loss in the conversion process ($U_3O_8 \rightarrow UF_6$), a 1% loss in the fuel fabrication process, a 1% loss in reprocessing, and no credit for the end-of-life fissile inventory.

Light-Water Reactors (PWRs and BWRs)

The analyses for PWRs were based on the Combustion Engineering System 80™ design for a 3800-MWt (1300-MWe) reactor, and most of the calculations were performed by Combustion

Engineering, with a few additional results provided by ORNL. The analyses for BWR-type LWRs were performed by General Electric and included calculations for "mixed lattices" in which MEU/Th pins (and also ThO_2 pins) were introduced within only a few of the LEU fuel assemblies rather than throughout the core.* (Note: In the "denatured power systems" considered later, only the PWR was used.)

Spectral-Shift-Controlled Reactors (SSCRs)

The mass flow calculations for SSCRs were also performed by Combustion Engineering, and since the SSCR design is based on a PWR, the CE PWR System 80™ design was again used. The advantage of the SSCR is that the heavy water used for long-term reactivity control shifts the neutron spectrum to higher energies at which they are preferentially absorbed in fertile materials. Thus the loss of neutrons to poisons is decreased with a concomitant increase in the amount of fissile material bred by the reactor.

Heavy-Water Reactors (HWRs)

The fuel requirements for HWR-CANDUs were calculated by Argonne National Laboratory. A current-generation 1200-MWe CANDU design was assumed for all cases except the natural-uranium fuel case, for which an older design rated at approximately 600 MWe was used. The CANDU utilizes D_2O as moderator and coolant in separate closed systems. It has a fuel management scheme which allows on-line refueling that minimizes downtime and promotes efficient use of the fuel by requiring less excess fuel to offset "parasitic" neutron absorption in fission products during long-term operations.

Gas-Cooled Reactors (HTGRs and PBRs)

The fuel-utilization characteristics of HTGRs were calculated by General Atomic, with some verification calculations carried out at ORNL. The assumed design was a 3360-MWt (1344-MWe) reactor with a core power density of 7.1 W_t/cm^3. The analyses for the PBR, which design concept was developed in West Germany and is represented by the 46-MWt Arbeitgemeinshaft Versuch Reacktor (AVR), were performed by a physics design group at KFA Julich, West Germany. The design was assumed to be a 3000-MWt (1000-MWe) reactor having a core power density of 5 MW/m^3.

Unlike any of the other reactors, the HTGR and PBR utilize a thorium-based reference fuel — currently a mixture of highly enriched uranium and thorium [i.e., HEU(235)/Th fuel which with recycle becomes HEU(233)/Th]; however, some consideration is being given to converting their designs to MEU/Th fuel. The principal difference between the HTGR and the PBR is that in the HTGR *prismatic* fuel elements are loaded into a graphite block whereas

*Concurrent with this study, investigations were made as part of NASAP (Nonproliferation Alternative Systems Assessment Program) to determine how much improvements in design and operating strategies would increase *in situ* utilization of bred fuel in LWRs operating on the once-through cycle. While such improvements were not considered as an integral part of this study, a brief calculation of the effects of an assumed 30% improvement in U_3O_8 utilization is discussed in Appendix A.

in the PBR small spherical elements of fissile and fertile material are introduced into a spherical core. An important feature of the PBR design is that it allows on-line refueling with spent fuel elements being removed from the bottom of the core. (Note: In the "denatured power systems" considered later, only the HTGR is used.)

Liquid-Metal Fast Breeder Reactors (LMFBRs)

Preliminary analyses of the impact of alternate fuels on LMFBRs were performed by Argonne National Laboratory, Hanford Engineering Development Laboratory, and Oak Ridge National Laboratory. Although differing in detail, each group selected as a reference design the "classical" LMFBR consisting of a Pu/U-oxide-fueled core surrounded by axial and radial blankets of fertile ^{238}U. The performance parameters of alternate fissile/fertile combinations were calculated by replacing the reference core and blanket materials with appropriate alternate materials. No attempt was made to optimize any of the designs to account for the different thermophysical properties of the alternate materials.

In addition to the calculations for these "homogeneous" LMFBRs (homogeneous cores), calculations were carried out at ORNL to determine the effect of intermixing fuel and blanket assemblies within the core. This reactor model is commonly referred to as the "heterogeneous" LMFBR. So that the results for the homogeneous and heterogeneous LMFBRs could be compared directly, only the ORNL-calculated fuel utilization and production data are included in this summary. In both sets of calculations, an oxide-based 1200-MWe plant was assumed, and the heterogeneity was accomplished by using alternating concentric fissile and fertile annuli in the core model.

4.3. Comparisons of Various Reactor-Fuel Combinations

Thermal Reactors

As discussed above, thermal reactors may operate on a once-through cycle or in a recycle mode. The once-through cycle, in turn, may be a throwaway cycle or a stowaway cycle, the economic feasibility of some fuels existing only with the latter. Table 4.1 shows, for example, that of the thermal reactors operating on once-through *throwaway* cycles, the HWR-CANDU utilizing SEU fuel would require the smallest U_3O_8 resource commitment. Ranking next would be the HTGR and PBR on LEU or MEU(235)/Th fuel and the HWR-CANDU on natural uranium fuel, with the HWR requiring little or no separative work.

Significantly, neither the HTGR nor the PBR requires a higher U_3O_8 commitment for the MEU(235)/Th once-through cycle than for the LEU case. This is primarily due to a high burnup design which allows most of the ^{233}U produced by these reactors to be burned *in situ* and contributes significantly to both the power and the conversion ratio. The unique design of the PBR would also permit recycle of the fertile elements without intervening reprocessing and thus would further reduce the ore (and SWU) requirements for the MEU(235)/Th cycle.

Table 4.1. 30-Year U_3O_8 and Separative Work Requirements of Thermal Reactors Operating on Resource-Based Fuels

Reactor/Fuel	U_3O_8 (ST/GWe)	Separative Work (10^3 kg SWU/GWe)
	Once-Through LEU Fuels	
HWR-CANDU/Nat. U	4,688	0
HWR-CANDU/SEU	3,563	922
PBR-LEU	4,500	-
HTGR/LEU(C/U = 400)	4,594	3,629
HTGR/LEU(C/U = 350)	4,860	3,781
SSCR/LEU	5,320	3,010
PWR/LEU	5,989	3,555
BWR/LEU	6,051	3,490
	Once-Through MEU(235)/Th Fuels	
HWR-CANDU/MEU(235)/Th	8,281	7,521[a]
PBR/MEU(235)/Th	4,184[b]	-
PBR/HEU(235)/Th	4,007[b]	-
HTGR/MEU(235)/Th (C/Th = 650)	4,515	4,143
HTGR/HEU(235)/Th	4,395	4,387
SSCR/MEU(235)/Th	7,920[a]	7,160[a]
PWR/MEU(235)/Th	8,360	7,595
BWR/MEU(235)/Th	8,680	7,763
	Recycle Fuels	
HWR-CANDU/MEU(235)/Th with U recycle	1,640	2,000
HTGR/MEU(235)/Th with ^{233}U recycle	3,666	3,361
HTGR/HEU(235)/Th with U recycle	2,280	2,278
SSCR/MEU(235)/Th with U recycle	3,220	3,077
PWR/LEU with U + Pu recycle	4,089	2,690
PWR/LEU with U recycle	4,946	3,452
PWR/MEU(235)/Th with U recycle	4,090	3,632
BWR/LEU with U + Pu recycle	3,869	1,980

[a]Estimated from other data provided on this reactor.

[b]Does not consider possible recycle of fertile elements without intervening reprocessing.

Ranking after the gas-cooled reactors and the HWR are the SSCR and LWRs on LEU fuel. The LWRs and the HWR-CANDU could not compete economically using MEU(235)/Th fuel on the once-through throwaway cycle. If, on the other hand, the once-through cycle were viewed as a *stowaway* cycle, in which case the fissile content of the spent fuel elements would be expected to be recovered at some future date, and especially if a stockpile of ^{233}U and/or Pu[f] were known to be required eventually, then the ranking of the reactors would

change. In this case, the PWR and the HWR would be the preferred reactors for MEU(235)/Th fuel because of their high ^{233}U production (see Table 4.2). Moreover, approximately one-third of the ^{235}U charge in each of these reactors would be recoverable. Similarly, the PWR and HWR, together with the SSCR, would rank highest on the LEU stowaway cycle if Pu^f recovery at some future date were anticipated.

It is to be remembered, however, that the spent fuel inventory is recoverable only when the spent fuel is reprocessed, whereas the U_3O_8 commitment is necessary throughout the operating lifetime of the reactor. Thus, on an economic basis, when MEU(235)/Th fuel is used, the expected future value of the recoverable fuel must offset the additional costs associated with using increasing amounts of the resource base.

Another aspect to consider in the stowaway cycle is the proliferation resistance of the stored spent fuel. For both the LEU and the MEU/Th once-through fuel cycles, the fissile uranium content of the spent fuel is denatured (diluted with ^{238}U) and hence is protected by the inherent isotopic barrier. Thus the plutonium in the fuel would be the fissile material most subject to diversion. Table 4.2 shows that employing the MEU/Th cycle in place of the LEU cycle sharply reduces the amount of plutonium produced (by 60-80%, depending on reactor type), and for both cycles the quantity of plutonium produced in the gas-cooled reactors is substantially less than that produced in the other reactor types. On the MEU(235)/Th cycle the HWR-CANDU also is a low plutonium producer.

Table 4.2. Estimated 30-Year Fissile Fuel Utilization and Production of Thermal Reactors Operating on Once-Through Cycles[a]

Reactor	^{235}U Charge (MT/GWe)	Fissile Discharge (MT/GWe)				Net Fissile Consumption (MT/GWe)
		^{235}U	^{233}U	Pu^f	Total	
			LEU Fuel			
HWR[b]	17.5	1.77	-	5.49	7.3	10.2
PBR[c]	18.1	2.79	-	1.89	4.7	13.4
HTGR	19.5	3.25	-	2.16	5.4	14.1
SSCR	22.3	5.46	-	5.88	11.3	11.0
PWR	24.7	6.45	-	5.22	11.7	13.0
			MEU(235)/Th Fuel			
HWR	32.6	10.08	14.28	0.75	25.1	7.5
PBR[c]	16.6	1.17	2.73	0.42	4.3	12.3
HTGR	18.0	1.35	2.31	0.69	4.4	13.6
PWR	33.8	11.52	7.80	2.13	21.4	12.4

[a]Calculated as initial charge for first year plus annual charge for equilibrium core times 29 years.

[b]SEU fuel.

[c]Values for PBR estimated from equilibrium cycle.

The picture for MEU(235)/Th fuel changes again if the recovery of the spent fuel fissile material is performed concurrently with the reactor operation — that is, if reprocessing is permitted and the recovered fissile material is recycled in the reactors during their lifetimes. Table 4.1 shows that for MEU(235)/Th fuel with ^{233}U recycle, the demands for U_3O_8 and separative work units are greatly reduced for reasons apparent in Table 4.3. With ^{233}U being recycled in the reactors, substantially less ^{235}U charge is required — approximately 30% less for the PWR and the HTGR and approximately 70% less for the HWR. In most cases the net consumption of fissile material is also reduced. This is because, as pointed out earlier, ^{233}U is a more efficient fuel for thermal reactors than ^{235}U.

The greater efficiency of ^{233}U fuel is particularly obvious when the fissile fuel requirements of a given thermal reactor operating on MEU(235)/Th fuel with ^{233}U recycle are compared with those of the same reactor operating on MEU(233)/Th fuel with ^{233}U recycle (that is, the denatured ^{233}U cycle). As shown in Table 4.4, on MEU(233)/Th fuel the net

Table 4.3. Estimated 30-Year Fissile Fuel Utilization and Production of Thermal Reactors Operating on MEU(235)/Th Fuels with U Recycle[a]

Reactor	^{235}U Charge (MT/GWe)	Fissile Discharge (MT/GWe)		Net ^{235}U Consumption (MT/GWe)
		^{235}U	Pu[f]	
HWR	9.4	3.0	0.9	6.4
HTGR[b]	20.7	1.9	0.8	18.8
SSCR	19.9	6.8	1.9	13.1
PWR	22.4	8.4	1.9	14.0

[a]Calculated as initial charge for first year plus annual charge for equilibrium core times 29 years.

[b]Data for HTGRs deduced from Table 6.1-3 of main report.

Table 4.4. Estimated 30-Year Fissile Fuel Utilization and Production of Thermal Reactors Operating on MEU(233)/Th Fuels[a] with U Recycle[b]

Reactor	^{233}U Charge (MT/GWe)	Fissile Discharge (kg/GWe)		Net ^{233}U Consumption (MT/GWe)
		^{233}U	Pu[f]	
HWR	25.7	21.9	1.0	3.8
HTGR[c]	12.3	3.3	0.8	9.0
SSCR	21.5	13.5	2.2	8.0
PWR	23.4	13.4	1.9	10.0

[a]Denatured ^{233}U fuel.

[b]Calculated as initial charge for first year plus annual charge for equilibrium core times 29 years.

[c]Data for HTGRs deduced from Table 6.1-3 of main report.

fissile fuel requirements of a given reactor are considerably reduced over those for the same reactor on MEU(235)/Th fuel. It is to be re-emphasized, however, that while the ^{235}U required for the MEU(235)/Th cycle can be obtained from the resource base, the ^{233}U required for the MEU(233)/Th cycle cannot. Thus any ^{233}U required by a reactor in excess of what it produces itself must be "manufactured" by another reactor, and the manufacturing process will make demands on the resource base — indirectly if not directly — until such time as a self-sustaining symbiotic system of reactors has evolved.

As discussed previously, one possible technique for manufacturing ^{233}U is to use thermal transmuters that operate on plutonium and produce ^{233}U via neutron absorption in the thorium included in their cores. A comparison of the various thermal reactors operating on Pu/Th fuel (Table 4.5) indicates that the HWR would be more efficient in this role than either the PWR or the SSCR. In all cases, of course, the reactors would utilize more fissile fuel than they would produce, which means that a power system consisting solely of thermal transmuters and denatured thermal reactors would not be self-sustaining. This is made more obvious by comparing Tables 4.4 and 4.5. Table 4.4 shows, for example, that over its lifetime a PWR operating on denatured ^{233}U [that is, on the MEU(233)/Th cycle] would require ~10 MT ^{233}U per GWe, while Table 4.5 shows that the only thermal transmuter that could produce this amount would be the HWR-CANDU. But the HWR transmuter, in turn, would require approximately 19.9 MT of Pu^f and none of the reactors operating on denatured ^{233}U even approach an adequate Pu^f production. Under its current design, the HTGR would not be an efficient transmuter, largely because it consumes much of the ^{233}U it breeds *in situ*.

Table 4.5. Estimated 30-Year Fissile Fuel Utilization and Production of Thermal Transmuters (Pu/Th Fuel)

Reactor	Pu^f Charge (MT/GWe)	Fissile Discharge (MT/GWe)		Net Pu^f Consumption (MT/GWe)	Transmutation Efficiency[a]
		Pu^f	^{233}U		
HWR-CANDU[b]	26.9	7.0	11.8	19.9	0.59
HTGR[b]	19.1	3.8	2.8	15.3	0.18
SSCR	47.9	23.4	8.2	24.5	0.33
PWR	42.6	20.9	8.2	21.7	0.38

[a]Tons of plutonium "transmuted" into tons of ^{233}U.

[b]Based on annual mass flow data in Table 6.1-3 of main report.

Fast Reactors

The fast reactors were considered as possible candidates for two roles: as power reactors operating on denatured ^{233}U fuel; and as transmuters burning plutonium to produce

^{233}U. With an LMFBR used as the model, the denatured FBRs were analyzed for a range of $^{233}U/U$ enrichments to parameterize the impact of the fuel on the reactor performance, and the transmuter FBRs were analyzed both for a $Pu/^{238}U$ core driving a ThO_2 blanket and for a Pu/Th system in which the thorium was included both in the core and in the blanket.

In denatured fast reactors the specified $^{233}U/U$ enrichment is a crucial parameter. Increasing the allowable enrichment of ^{233}U in U permits more thorium to be used in the fuel material and hence allows the reactor to be more self-sufficient (i.e., reduces the required ^{233}U makeup). Increasing the ^{233}U enrichment also reduces the amount of fissile plutonium contained in the discharge fuel, which is obviously desirable from a safeguards viewpoint. However, increasing the ^{233}U fraction also increases the vulnerability of the denatured fuel to isotopic enrichment, effectively forcing a compromise between proliferation concerns regarding the fresh fuel versus proliferation concerns regarding the spent fuel. These trends are apparent in Table 4.6. At a 12% enrichment, which is the lowest enrichment feasible for a denatured fast system, the overall breeding ratio is 1.12, but the breeding ratio for ^{233}U is only 0.41. Thus a significant amount of ^{233}U makeup is required. Still the system is a net fissile producer because of the plutonium it breeds. If the enrichment is increased to 20%, or preferably to 40%, the self-sufficiency of the system greatly increases. It would increase even more so if Th blanket elements were introduced inside the core (that is, if a heterogeneous core were employed). However, except for the 12% case, all these enrichments exceed the criterion set for this study. On the other hand, the ratio of ^{233}U produced to Pu^f produced is very sensitive to the specified degree of denaturing in the range of 12-20% $^{233}U/U$. This suggests that significant performance improvements may be possible for relatively small increases above the 12% limit. Of course, the overall "breeding" ratio of the denatured LMFBR will always be considerably degraded below that for the reference $Pu/^{238}U$ cycle.

Because of the superior breeding potential of a ^{239}Pu-fueled system relative to a ^{233}U-fueled system in a fast neutron spectrum, the fast reactor is ideally suited to the role of a plutonium-fueled transmuter (or breeder-transmuter). Moreover, in contrast to

Table 4.6. Denatured LMFBR Mid-Equilibrium Cycle "Breeding" Ratio Components*

Fuel†	^{233}U Component	Pu Component	Overall Breeding Ratio
^{233}U(12%)/U	0.41	0.71	1.12
^{233}U(20%)/U/Th	0.70	0.39	1.09
^{233}U(40%)/U/Th	0.90	0.15	1.05
^{233}(100%)/Th	1.02	-	1.02

*A separate, more recent study [Proliferation Resistant Large Core Design Study (PRLCDS)] indicates that with design modifications substantial improvements in the FBR performance is possible.

†For homogeneous cores; i.e., no blanket elements introduced inside core.

the thermal transmuters, the fast reactors result in a net overall fissile material gain. Table 4.7 shows that such reactors can have a net fissile production that is comparable to that of classical fast breeders, both in the homogeneous-core configuration and in the heterogeneous-core configuration (the latter with internal blanket elements).

In the case of the homogeneous-core breeder-transmuter, the reactor both sustains itself (though only marginally) and produces ^{233}U, its overall "breeding" ratio (i.e., net fissile production per GWe) being approximately the same as that for the reference Pu/U cycle. When a homogeneous Pu/Th core is used (next to last entry in Table 4.7), the ^{233}U production increases almost a factor of 4, but this is achieved by the "sacrificial" consumption of plutonium. Thus, these two reactor types represent a tradeoff between maximum ^{233}U production and maximum total fissile production on homogeneous-core reactors. Converting to heterogeneous cores would increase the net fissile production, but again the gain would be at the expense of an increased plutonium consumption. On the other hand, numerous advanced LMFBR concepts currently under study could significantly impact the performance parameters of such systems.

Table 4.7. Estimated 30-Year Fissile Fuel Utilization and Production of LMFBRs Operating on Pu^f Fuel in Combination with ^{238}U and/or ^{232}Th

Fuel	Blanket (A/R/I)[a]	Pu^f Charge (MT/GWe)	Net Fissile Production (MT/GWe)		
			Pu^f	^{233}U	Total
		Classical Breeders			
Pu/U (Ref.)	U/U	24.1	+ 5.6	0	+ 5.6
Pu/U	U/U/U	35.2	+10.3	0	+10.3
		Breeder-Transmuter			
Pu/U	U/Th	24.1	+ 0.9	+4.6	+ 5.5
		Transmuters			
Pu/U	U/Th/Th	37.5	- 7.1	16.1	+ 9.0
Pu/Th	Th/Th	27.6	-14.8	17.5	+ 2.7
Pu/Th	Th/Th/Th	40.9	-16.7	24.0	+ 7.3

[a] Axial/Radial/Internal (if any).

[b] Editor's Note: These comparisons of net fissile production are not meant to imply that a heterogeneous LMFBR core design is superior to a homogeneous design since many other factors enter into the comparisons of the overall systems.

4.4. Conclusions

Since optimization of the various reactors for the particular fuels considered was beyond the scope of this study, the results presented above are subject to several uncertainties. Nevertheless, certain general conclusions on the impact of the various fuel cycles on reactor performance are believed to be valid:

- For once-through *throwaway* systems, the various systems studied are ranked in order of optimum resource utilization as follows: the HWR on the LEU cycle or on natural uranium; the HTGR and PBR on either the LEU cycle or on the MEU/Th cycle; and the SSCR and PWR on the LEU cycle. On the MEU/Th cycle the SSCR and PWR require more uranium than they do on the LEU cycle and hence do not merit further consideration for once-through operation.

- For once-through *stowaway* systems, in which the fissile material in the spent fuel is expected to be recovered at some future date, the relative ranking of the systems would depend on the ultimate destination of the fissile material. If future nuclear power systems are to be thermal recycle systems, then early emphasis should be placed on reactors and fuel cycles that have a high ^{233}U discharge. If the future systems are to be fast recycle systems, then emphasis should be placed on reactors and fuel cycles that will provide a plutonium inventory.

- For thermal recycle systems, the preferred basic fissile material is ^{233}U. However, implementation of a ^{233}U fuel cycle will require an exogenous source of the fissile material; therefore, it is likely that the MEU(235)/Th cycle would be implemented first to initiate the production of ^{233}U. Both the unburned ^{235}U and the ^{233}U would be recycled; thus the system would evolve towards the MEU(233)/Th cycle, which is the denatured ^{233}U cycle as defined in this study. However, it is to be emphasized that these reactors will not produce enough ^{233}U to sustain themselves and separate ^{233}U production facilities must be operated. A Pu/Th-fueled reactor has been considered as a ^{233}U production facility.

- For fast recycle systems, the preferred basic fissile material is Pu^f. Using ^{233}U as the primary fissile material or placing thorium in the core sharply reduces the breeding performance of fast reactors. However, fast reactors using plutonium fuel and thorium blankets would be efficient ^{233}U production facilities.

5.0. IMPLEMENTATION OF DENATURED FUEL CYCLE

As defined in this study, a nuclear power system based on the denatured ^{233}U fuel cycle would consist of LWRs and possibly ACRs (SSCRs, HWRs, or HTGRs) and FBRs operating on denatured ^{233}U fuel, plus all the support facilities required to sustain those operations. The support facilities would include transmuters, which could be any of these same reactors operating on Pu/Th fuel, and they might also include breeder-transmuters or even the classical breeders. In addition, the support facilities would include the necessary facilities for fresh fuel fabrication, spent fuel reprocessing, and waste treatment (see Fig. 3.1).

As of this date, neither the required reactors nor the supporting fuel cycle facilities are fully developed for operation in such a power system. The LWRs, of course, are commercialized for operation on their reference LEU cycle and the ACRs have reached advanced stages of development for their own reference fuels; however, only preliminary consideration has been given to adapting any of these reactors to denatured ^{233}U or Pu/Th fuel. The FBRs, which for this study were assumed to be of the LMFBR type, are similarly well advanced for their own reference fuel but not for an alternate fuel. Thus prior to the implementation of a denatured fuel cycle, the reactor-fuel combinations to be included in the cycle must be selected and the research and development required for implementing and sustaining the resulting systems must be carried out.

While it was too early for detailed R&D programs to be described in this study, it was possible to make subjective evaluations of the status of the individual reactor and fuel recycle technologies and to project estimated schedules for their completion, together with order-of-magnitude costs. Such estimates were made and are reported below. It should be emphasized, however, that the schedules assumed for deployment of the reactors and fuels are based solely on the minimum time estimated to be *required to solve technical problems.* That is, no allowances were made for impediments to commercialization, such as licensing difficulties, interrupted construction schedules, etc. Under these conditions, the introduction dates for specific reactors operating on specific fuels were determined as follows, the introduction date being defined as the date of startup of the first unit, with additional reactors introduced at a *maximum* rate of 1,2,4,...units each biennium:

1987 - LWRs operating on "denatured ^{235}U" fuel [i.e., MEU(235)/Th fuel].

1991 - LWRs operating on denatured ^{233}U [i.e., MEU(233)/Th], Pu/U and Pu/Th fuels.

- SSCRs operating on LEU, denatured ^{233}U, and Pu/Th fuels.

1995 - HWRs operating on any of several proposed fuels.

- HTGRs operating on any of several proposed fuels.

2001 - FBRs operating on Pu/U, Pu/Th, and denatured ^{233}U fuels.

Obviously such an accelerated schedule could not be met without substantial initiatives and strong financial support from the U.S. Government. And because the first several commercial units would have to be ordered before the operation of the initial demonstration plant, partial government support would have to extend through these units.

5.1. Reactor Research and Development Requirements and Costs*

In planning for the introduction of the various reactors into the denatured ^{233}U fuel cycle, it was assumed that the development of each reactor on its own reference fuel would be completed first, and that conversion to an alternate fuel would follow. In general, the R&D required to bring a reactor concept to the point of commercialization on its reference fuel is divided into three areas:

(1) Proof of principle (operating a test reactor of small size);
(2) Design, construction, and operation of a prototype plant (intermediate size);
(3) Design, construction, and operation of a commercial-size demonstration plant (about 1000 MWe).

Similarly, the R&D required to convert a reactor to an alternate fuel would be divided into three areas:

(1) Data base development (providing physics verification and fuel performance information necessary for the design and licensing of reactors operating on the alternate fuel);
(2) Reactor components development needed to accommodate the alternate fuel;
(3) Demonstration of the reactor on the alternate fuel cycle.

Of the reactors considered in this study, only the LWRs have been fully developed for their reference cycle. Thus, the R&D on LWRs could be immediately directed to that necessary to convert LWRs to alternate fuels. The three ACRs have progressed past the proof-of-principle step, and moreover, their concepts have been sufficiently developed that the prototype plant stage either has been completed or could be bypassed. As a result, the remaining R&D for the ACRs is that essential to the operation of demonstration plants on their reference cycles, to be followed, of course, by the R&D required to convert them to alternate fuels. The LMFBR is similarly at the demonstration stage on its reference fuel.

Light-Water Reactors (LWRs)

Preliminary evaluations of design and safety-related considerations indicated that LWRs could operate on thorium-based fuels [MEU(235)/Th or MEU(233)/Th] with little modification, and an essentially current-generation LWR could serve as a demonstration plant. However, prior to any such demonstration, R&D programs would be required to verify the physics data base for thorium-based systems and to develop and test as-yet unidentified reactor components. Also, some core design changes would be necessary to accommodate the new fuel, and safety analyses of the new core would be required for licensing. Since the private sector would have little incentive to convert to MEU/Th fuels, especially without assurances that the ^{233}U in the spent fuel would eventually be recycled, all these programs would require government subsidy to support the R&D program. In addition, the demonstration itself probably would require a subsidy to insure the sponsoring utility against the potential for decreased reactor availability. The government subsidy could total between $100M and $200M. Presumably the costs of converting LWRs to Pu/Th fuels would be the same order of magnitude.

*See Section 4.2 for additional discussion of reactors.

Spectral-Shift-Controlled Reactors (SSCRs)

The proof-of-principle of the SSCR was accomplished by the operation of the BR3 reactor in Belgium. A prototype plant may be unnecessary since various components required for heavy-water handling and reconcentration are well established by heavy-water reactor operating experience and the SSCR itself is a modification of the already commercialized PWR. Thus the next step for an SSCR is a demonstration of the reactor on its reference fuel (LEU).

The demonstration plant could be designed so that it operated in either the conventional poison control mode or the spectral-shift mode. Then the capital risk would be limited to that required for spectral-shift control plus heavy water costs. Because the proposed schedule for commercialization of the SSCR is more rapid than for any of the other ACRs, it was assumed that the government would cover all the component R&D costs and licensing costs for the demonstration plant and that it would also purchase all the extra equipment required for the first five units. Additional government support to mitigate the effect of probable lower capacity factors for the first experimental unit would be anticipated, as well as carrying charges on the D_2O inventory. The total government subsidy would range between \$300M and \$350M. Incremental costs for then converting the SSCR to MEU/Th fuel would be \$10M to \$60M. In the long-term, a major impediment to commercialization of the SSCR could be the availability of D_2O, and government incentives to ensure D_2O production would be necessary. (The costs for converting the SSCR to Pu/Th fuel were not considered in this study.)

Heavy-Water Reactors (HWRs)

It is assumed here that the U.S. HWR would be based on the CANDU and deployed under Canadian license and with Canadian cooperation. Thus, the anticipated R&D and the associated costs would be those required to adapt the present CANDU design to SEU fuel, to extend the design to a larger plant (1000 MWe), and to acquire U.S. licensing.

As with the SSCR, government support would be required, the total costs for transferring the Canadian technology to the U.S. and upgrading the HWR to meet U.S. licensing criteria probably reaching the range of \$200M to \$400M. Also, a substantial government subsidy would be required for the first unit and further government support could be necessary for the next four units, especially if an accelerated schedule is mandated. The costs of converting the HWR to a denatured fuel were estimated to be approximately the same as converting an LWR to denatured fuel. Again, as with the SSCR, the D_2O supply would be a crucial factor. (The costs associated with the use of Pu/Th fuel in HWRs were not considered.)

High-Temperature Gas-Cooled Reactors (HTGRs)

The HTGR status in the U.S. is considered to be at the prototype stage with the 330-MWe Fort St. Vrain HTGR plant, and the basic reactor development still required is that associated with the demonstration of a large plant design. The costs for component R&D and licensing requirements would be in the range of \$200M to \$300M. As was the case for the HWR, it is assumed that a substantial subsidy would be required for the first demonstration plant and a partial subsidy would be necessary for each of the next four units, especially in the advent of an accelerated schedule.

Since unlike the other reactors considered, the HTGR has a thorium-based reference fuel, it is possible that a denatured cycle could be designated as the reference cycle so that the demonstration plant would be a denatured system. If this were done, the additional costs required to convert the HTGR to a denatured fuel might be smaller than those required to convert LWRs to thorium-based fuels. (As with the other reactors, the costs of converting the HTGR to a Pu/Th system were not considered.)

Fast Breeder Reactors (FBRs)

FBR R,D&D requirements and costs were not included in this study since updated data for the LMFBR both on its reference cycle and on the denatured ^{233}U and Pu/Th cycles were being developed under other programs.

Summary of Reactor R,D&D Costs

The estimated costs for reactor R,D&D are summarized in Table 5.1.

Table 5.1. Estimated Cost Ranges for Development and Commercialization of LWRs on MEU/Th Fuels and ACRs on Reference Fuels

	Costs ($M)	Comments
LWR; MEU/Th Fuels		
Research, design, component development, licensing	50-150	Government-subsidized.
Large-scale demonstration	50-200	Demonstration in current-generation LWR; 25% government-subsidized.
	100-350	
SSCR; LEU Fuel		
Research, design, component development, licensing	50-100	Government-subsidized.
Large-scale demonstration		
First unit	150	Demonstration designed for either poison control or spectral-shift control; requirements for latter (i.e., ~$150M) government-subsidized.
Next four units (commercial)	100	Government subsidy for extra (spectral shift) components.
	300-350[a]	Could probably be converted to MEU/Th fuel for additional $10M - $60M if LWRs already converted.
HWR; SEU Fuel		
Research, design, component development, licensing	200-400	Government-subsidized.
Large-scale demonstration		
First unit	800	50% government-subsidized.
Next four units (commercial)	2,800	25% government-subsidized.
	~4,000[a,b]	Additional incremental cost to convert to MEU/Th fuels approximately equal to that for LWR conversion.
HTGR; HEU/Th Fuel		
Research, design, component development, licensing	200-300	Government-subsidized.
Large-scale demonstration		
First unit	800	50% government-subsidized.
Next four units (commercial)	2,800	25% government-subsidized.
	~4,000[b]	If MEU/Th fuel selected as reference fuel, additional incremental cost probably less than cost of converting LWR to MEU/Th fuels.

[a]Excludes costs of heavy-water plant facilities.

[b]This cost assumes an accelerated development schedule; if only the first unit of the four commercial units required a subsidy, the cost would be $2,000M to $2,500M.

5.2. Fuel Recycle Research and Development Requirements and Costs

Since the denatured ^{233}U fuel cycle would consist of reactors operating on different fuels in symbiosis, it is possible that an integrated recycling technology could be developed to handle the various fuels required in the system. However, until a U.S. strategy is developed to establish what reactors would be used and what regulatory requirements would be imposed at the various points of the fuel cycle, as well as the size of the commercial industry, it would be inadvisable even to attempt to describe an integrated system, much less to project its R&D requirements and costs. Thus, at this point, we can only describe the status of the various areas of recycle technology for the different types of fuel and make very preliminary estimates of the requirements to complete the individual cycles. The technological areas included are as follows:

(1) Fuel fabrication/refabrication (fuel material preparation, rod fabrication, and element assembly).

(2) Fuel qualification (irradiation performance testing and evaluation).

(3) Fuel reprocessing (headend treatment, solvent extraction, product conversion, and off-gas treatment).

(4) Waste treatment (concentration, calcination, vitrification, and radioactive-gas treatment).

Fuel Fabrication/Refabrication and Qualification

The technology for fabricating uranium-based metal-clad oxide pellet fuels such as those used in LWRs and HWRs is complete. While Pu/U oxide pellet fuels have also been fabricated, the work has been at pilot plant scale and a significant amount of R&D is still required to commercialize Pu/U fuel. In particular, a pelletizing process must be demonstrated, and methods must be developed for verifying and controlling the characteristics of Pu/U fuels, recovering contaminated scrap, and performing nondestructive assays of powders, fuel rods, and wastes. In addition, the remote operation of a large-scale fabrication plant must be demonstrated, and the irradiation performance of Pu/U fuels produced in commercial-scale processes and equipment must be shown to be satisfactory.

The technology for fabricating thorium-based metal-clad oxide pellet fuels is less complete than that for Pu/U fuels, requiring significantly more effort in all the aforementioned areas. Also, the developmental effort will be complicated by the need for remote operations and maintenance at all points in the cycle where radioactive ^{232}U is present.

The technology for fabricating uranium oxide or uranium carbide microspheres embedded in a graphite fuel element, such as are used in the HTGR, is well advanced; however, additional R&D prior to construction of a hot demonstration facility is needed in the areas of refabrication equipment scaleup, recycle of scrap material, control of effluents, assay of fuel-containing materials, and qualification of recycle fuel. Further work will also be required to fabricate denatured uranium fuels for HTGRs because of a higher uranium content of the fissile particle and an increased production of plutonium during irradiation.

Fuel Reprocessing

The well-established Purex process contains the basic technology for reprocessing U and Pu/U metal-clad oxide pellet fuels with low burnup; however, a commercial reprocessing plant that conforms to current U.S. federal and state requirements has not been operated. Neither has an integrated plant for fuels with high burnup been demonstrated. Specific areas still needing attention are: operation and maintenance of the mechanical headend equipment; methods for handling highly radioactive residues following dissolution of high-burnup fuel; techniques for reducing radioactive off-gas releases to conform to anticipated regulations; and conversion processes for Pu from power reactor fuels.

The Thorex process for reprocessing Th-based oxide pellet fuels is not as advanced as the Purex process. While the Thorex process has been used to process oxide fuels of low burnup in limited quantities, it is still to be tested in a large plant with high-burnup fuel.

In the Thorex process the headend treatment will differ for the metal-clad fuel and the graphite-based HTGR fuel. For the metal-clad fuels additional R&D will be required to develop techniques for removing the zirconium metal cladding, and if fluoride is required, significant waste-handling problems may be encountered. For the HTGR-type fuel, development work is needed in the crushing, burning, and particle separation operations and in the treatment of ^{14}C-containing off-gases.

Developmental work is also needed in several areas of the solvent extraction process for thorium-containing fuels. These include: fuel dissolution, feed adjustment and clarification; techniques for containing ^{220}Rn and other radioactive gases; recovery of fully irradiated thorium in large-scale facilities, partitioning of fuel solutions containing U, Pu, and Th; recovery and handling of highly radioactive product streams; and process and equipment design integration.

While commercialization of these processes could require from 12 to 25 years, depending on their current status, *commercial-scale* reprocessing will not necessarily be required on the same time scale as the introduction of the recycle fuels, since initially the demand for recycle fuel could be met by pilot or prototype plants. In fact, commercial reprocessing would not even be feasible until a sufficient backlog of spent fuel had accumulated.

Waste Treatment

To treat the wastes in all cycles requires the development of processes for concentration, calcination, and vitrification of high-level and intermediate-level solid and gaseous wastes. The requirements will be similar for all cycles but somewhat more complex for thorium-based cycles if fluorides are present. No examination of these costs was included in this study.

Summary of Fuel Recycle R,D&D Costs

Estimated cost ranges for the research, development, and commercialization (demonstration) of typical new fabrication/refabrication and reprocessing technologies are given in Tables 5.2 and 5.3 respectively. To these must be added the cost for waste

Table 5.2. Estimated Cost Range for Development and Commercialization of New Refabrication Technology[a]

	Unescalated Billions of Dollars
Base technology	0.1 - 0.3
Cold component testing	0.2 - 0.4
Irradiation performance testing	0.1 - 0.4
Total[b]	0.4 - 1.1
Large-scale demonstration[c]	0.7 - 1.4

[a]Time requirements: 8 - 10 years if similar to established technology; 15 years for new technology.

[b]Estimated government subsidy.

[c]Commercial facility; extent of government participation difficult to define at this time.

Table 5.3. Estimated Cost Range for Development and Commercialization of New Reprocessing Technology[a]

	Unescalated Billions of Dollars
Base technology R&D	0.1 - 0.5
Hot pilot plant testing	0.5 - 1.0
Subtotal	0.6 - 1.5
Large-scale cold prototype testing	0.2 - 0.5
Total[b]	0.8 - 2.0
Large-scale demonstration plant[c]	1.0 - 3.0

[a]Time requirements: ∿12 years for established technology to possibly 25 years for new technology.

[b]Estimated government subsidy.

[c]Commercial facility; extent of government participation difficult to define at this time.

Table 5.4. Estimated Range of Fuel Recycle R&D Costs*

Reactor Type	Billions of Dollars			
	Pu/U	Pu/Th	MEU/Th	HEU/Th
Water Reactors	1.3-2.3	1.6-3.0	1.8-3.3	1.6-2.9
HTGRs	1.4-2.6	1.6-3.0	1.8-3.3	1.6-2.9
FBRs	1.6-3.0	1.8-3.2	2.0-3.6	1.7-3.1

*Includes costs for developing reprocessing and refabrication technologies and a portion of the waste treatment technology development costs; excludes large-scale demonstration plant.

treatment technology development. Traditionally the costs borne by the government are those covering all steps up to demonstration. Part of the initial demonstration may also be government supported, but since the demonstration plant will be a commercial facility the costs could be recovered.

Tables 5.2 and 5.3 show that the major costs associated with commercialization of fuel cycles lie at the far end of the R&D progression. Thus the costs will be least during the base technology phase, which may require 2 to 6 years. They will increase consistently during the engineering phase (5 to 12 years) and rapidly during the facility design and construction (8 to 12 years). However, considerable overlap of these phases and the associated costs can be expected. The thorium-based cycles are expected to require the longest developmental times.

Table 5.4 presents the R&D cost ranges in terms of reactor types and fuel recycle systems. Although a large uncertainty is associated with each case, the trends are apparent. For all reactors, the Pu/U fuel cycle would be the least expensive and the MEU/Th cycle (denatured uranium cycle) would be the most expensive. Thus, if a decision is made to utilize denatured ^{233}U fuel, the costs for developing the fuel recycle facilities will be significantly greater than for any other cycle and will be compounded by the requirement for the simultaneous operation of reactors on other fuels.

5.3. Possible Procedure for Implementing Denatured ^{233}U Fuel Cycle

From the preceding discussion, it is obvious that the only reactors that could operate on denatured ^{233}U fuel in the near term (by 1991) would be LWRs and SSCRs, and presumably LWRs would be used first. Two possibilities exist for obtaining the required ^{233}U prior to the introduction of commercial fuel reprocessing. In the first, an MEU(235)/Th core would be introduced in LWRs to initiate the production of ^{233}U. Unfortunately, this scheme suffers from very high fissile inventory requirements (see Section 4.3). In the second, non-fueled ThO_2 rods would be introduced in certain lattice locations and/or MEU(235)/Th fuel would be used in only a fraction of the fuel rods, the remaining fuel rods being conventional LEU fuel rods. This second option significantly reduces the fissile inventory penalty associated with full thorium loadings in LWRs and for BWRs may even offer operational benefits. Also, it would allow experience with thorium-based fuels to be gained on a stepwise basis.

Although a reprocessing capability would be necessary to recover the bred ^{233}U, such a capability would not be required for the qualification and demonstration of the initial MEU(235)/Th fuel. Fabrication of MEU(235)/Th fuel could probably be accomplished with existing LEU facilities within 2 or 3 years, and qualification and/or demonstration could be completed within an additional 5 to 7 years. Thus, the operation of LWRs with MEU(235)/Th fuel or with partial thorium loadings could be accomplished during the next decade, and the spent fuel could be stored in repositories in secure fuel storage centers. A stockpile of ^{233}U and plutonium would then be initiated. The additional fuel cycle service facilities, such as isotopic separation, reprocessing, fuel refabrication and possibly waste isolation, could be introduced into these centers later as the need develops, initially as pilot-plant-scale facilities followed by larger prototypes and then commercial-scale plants.

With the deployment of the pilot-scale reprocessing and refabrication facilities, recovery of Pu and U from spent fuel and the subsequent refabrication of Pu/Th and denatured ^{233}U fuels could be demonstrated within the center. Dispersed LWRs fueled with denatured ^{233}U or LEU and energy-center LWRs fueled with Pu/Th could then begin operation. At this point the first phase of a nuclear power system that includes reactors operating both in energy centers and at dispersed locations outside the centers would be in effect. During this phase, which is represented in Fig. 5.1a, the research and development that will be required to deploy Pu-fueled FBR transmuters with thorium blankets in the energy centers could be pursued.

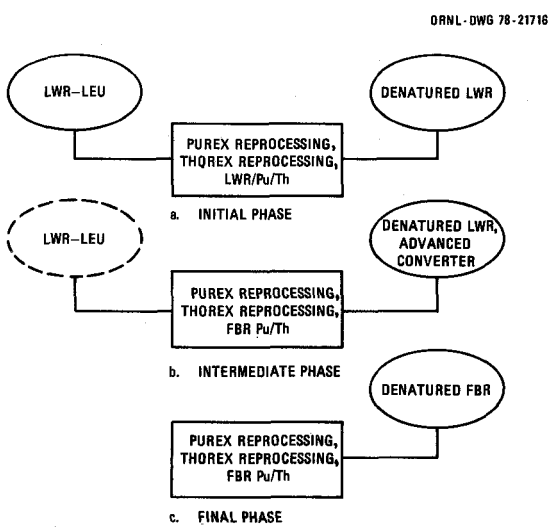

Fig. 5.1. Three Phases for an Evolving Energy Center.

With these advance preparations having been made, by the time conventional LEU fueling in LWRs begins to phase out (due

to increasing depletion of an economic resource base), the power system would evolve into a fast/thermal combination (see Fig. 5.1b). Such a system could provide adequate capacity expansion for modest energy demand growth; however, if the energy demand is such that the fast/thermal system is inadequate, an all-fast system including denatured FBRs could be substituted as shown in Fig. 5.1c. The necessity of the third phase of the energy center is uncertain at this time, depending on both the supply of economically recoverable U_3O_8 and the energy demand.

Variations of all three of these phases of an energy center operating in concert with dispersed reactors are represented in the denatured power system scenarios analyzed in Section 6.

5.4. Conclusions

The preceding discussion can be summarized as follows:

- The rapid introduction of advanced reactor concepts operating on MEU/Th or Pu/Th fuels would require very large government support for R&D, for demonstration facilities, and for lead commercial plants.

- The initial production of ^{233}U for the denatured ^{233}U fuel cycle could be accomplished by introducing thorium into the LWRs currently operating on the LEU once-through cycle. However, no economic incentive would exist within the private sector to convert LWRs to thorium-based fuels because of the increased costs associated with the concomitant higher fissile loadings. If government incentives were provided, initial production of ^{233}U for later recycle could begin by the mid-1980's. Recycle of ^{233}U on a commercial scale would not be feasible prior to the year 2000, however.

- A fuel recycle R&D program for MEU/Th and Pu/Th fuels should be initiated at the same time a decision is made to fabricate thorium-containing fuel for large-scale irradiation in existing LWRs. Pilot-scale recycle facilities could be required within a few years after the initiation of a thorium irradiation program.

- Fuel service/energy centers whose ultimate purpose is to utilize plutonium both for energy production and for ^{233}U production would progress through various phases. Initially these centers would be fuel storage facilities. With the introduction of reprocessing and refabrication in the center, LWRs located at dispersed sites would be fueled with denatured ^{233}U. Concurrently, Pu-fueled thermal transmuters would be deployed within the center. Later, ACRs operating on denatured fuel would be added at dispersed locations and FBRs operating on Pu/Th fuel would be added in the energy center. Ultimately, to meet long-term energy demands, an all-fast system could evolve, with FBR transmuters in the energy center and denatured FBRs at dispersed locations.

6.0. ADEQUACY OF DENATURED POWER SYSTEMS FOR MEETING POWER DEMANDS

The final step in this interim study was an analysis to determine the adequacy of several "denatured power systems" based on the secure energy center and dispersed reactor concept. A power system was considered to be adequate if it could meet projected U.S. nuclear power demands of 350 GWe in the year 2000, which is consistent with the current construction plans of utilities through the 1980s, and provide a net increase of 15 GWe/year thereafter up to the year 2050. The analysis procedure can be described as follows: Given a specified U_3O_8 ore supply and a specified set of reactor options, calculate (1) the potential role of nuclear power, (2) the resources required to achieve that role, and (3) the amount, composition, and movement of the fissile material through each step of the fuel cycle. Step 3 was included because such information must be available in order to assess the diversion resistance of a power system.

The denatured power systems were divided into two major categories: those consisting of thermal reactors only and those consisting of both thermal and fast reactors. In each case the power system was initiated with LWRs operating on the LEU cycle. ACRs operating on LEU and/or other types of fuel, including denatured ^{233}U fuel, were then added as they became available. In those cases that included fast reactors, FBRs operating on denatured ^{233}U, Pu/Th, and/or Pu/U fuel were also added. The times that specific reactor and fuel combinations could be introduced were consistent with the schedule given in Section 5.

Three "nuclear policy options" were examined under each of the two major categories of denatured systems, the individual options differing primarily in the extent to which plutonium existed in the system. In turn, four cases were studied under each option, the individual cases being distinguished by the type of ACR utilized in the system — LWR,* SSCR, HWR, or HTGR. In addition to the denatured options, three nondenatured options were studied for comparison purposes: one utilizing LEU fuel on a throwaway/stowaway cycle and two utilizing Pu/U recycle fuel.

All the cases analyzed are listed in Tables 6.1a, 1b, and 1c. It will be noted that the second LWR listed in Table 6.1a is an "extended discharge" LWR which is an advanced converter that is assumed to have U_3O_8 requirements 6% lower than those of the standard LWR and to become available in 1981. Although not considered in this original analysis, LWRs can probably be optimized so that their U_3O_8 requirements on the once-through cycle are reduced as much as 30%. A subsequent analysis that assumed this improvement is discussed in Appendix C.

Two different U_3O_8 ore supply models were employed, one representing a conservative estimate of the amount of ore that is recoverable in the U.S. at a reasonable cost and the other an optimistic estimate. Because preliminary calculations had shown that nuclear power plants could not compete with coal power plants at long-run marginal costs greater than $160 per pound of U_3O_8, these estimates are given in terms of the amount of uranium ore available at less than $160 per pound. The conservative estimate is 3 million ST U_3O_8 and the optimistic estimate is 6 million ST. The marginal costs corresponding to these two supply models, which are referred to as "High-Cost U_3O_8 Supply" and "Intermediate-Cost U_3O_8 Supply"

*That is, an LWR operating on some fuel other than LEU fuel.

Table 6.1a. Descriptions of Nondenatured Nuclear Power Systems

Case	Dispersed Reactors	Energy-Center Reactors
Option 1: Throwaway/Stowaway Option		
(Spent fuel returned to energy center for ultimate disposal.)		
1L	LWR-LEU-S[a] LWR-LEU-E[b]	
1S	LWR-LEU SSCR-LEU	
1H	LWR-LEU HWR-Nat.U HWR-SEU	
1G	LWR-LEU HTGR-LEU-T[c]	
Option 2: Pu/U Recycle Option with ACRs		
(Pu recycled in energy center only.)		
2L	LWR-LEU	LWR-Pu/U
2S	LWR-LEU SSCR-LEU	LWR-Pu/U
2H	LWR-LEU HWR-Nat.U HWR-SEU	HWR-Pu/U
2G	LWR-LEU HTGR-LEU	HTGR-HEU(235)/Th HTGR-HEU(233)/Th HTGR-Pu/Th
Option 3: Pu/U Recycle Option with ACRs/FBRs		
(Pu recycled in energy center only.)		
3L	LWR-LEU	LWR-Pu/U FBR-Pu/U/U[d]
3S	LWR-LEU SSCR-LEU	LWR-Pu/U FBR-Pu/U/U
3H	LWR-LEU HWR-Nat.U HWR-SEU	HWR-Pu/U FBR-Pu/U/U
3G	LWR-LEU HTGR-LEU	HTGR-HEU(235)/Th HTGR-HEU(233)/Th HTGR-Pu/Th FBR-Pu/U/U

[a]Standard LWR; this reactor used in all other cases.
[b]Extended discharge LWR.
[c]Optimized for throwaway.
[d]FBR with Pu/U core, U blanket.

Table 6.1b. Descriptions of Denatured Nuclear Power Systems Using Thermal Reactors Only

Case	Dispersed Reactors	Energy-Center Reactors
Option 4: Denatured ACRs, Pu Throwaway		
(U only recycled; Pu stored in energy center for ultimate disposal or future use.)		
4L	LWR-LEU LWR-MEU(235)/Th LWR-MEU(233)/Th[a]	
4S	LWR-LEU LWR-MEU(235)/Th SSCR-LEU SSCR-MEU(233)/Th	
4H	LWR-LEU HWR-Nat.U HWR-SEU HWR-MEU(235)/Th HWR-MEU(233)/Th	
4G	LWR-LEU HTGR-LEU HTGR-MEU(235)/Th HTGR-MEU(233)/Th	
Option 5U: Denatured ACRs, Pu Minimization		
(Pu recycled in energy center; goal is to minimize amount of Pu produced and to "transmute" all Pu into ^{233}U.)		
5UL	LWR-LEU LWR-MEU(235)/Th LWR-MEU(233)/Th	LWR-Pu/Th
5US	LWR-LEU LWR-MEU(235)/Th SSCR-LEU SSCR-MEU(233)/Th	SSCR-Pu/Th
5UH	LWR-LEU HWR-Nat.U HWR-SEU HWR-MEU(235)/Th HWR-MEU(233)/Th	HWR-Pu/Th
5UG	LWR-LEU HTGR-LEU HTGR-MEU(235)/Th HTGR-MEU(233)/Th	HTGR-Pu/Th
Option 5T: 5U Minus MEU(235)/Th Reactor		
(Pu recycled in energy center; no attempt made to minimize amount of Pu produced, but all Pu produced is "transmuted" to ^{233}U.)		
5TL	LWR-LEU LWR-MEU(233)/Th	LWR-Pu/Th
5TS	LWR-LEU SSCR-LEU SSCR-MEU(233)/Th	SSCR-Pu/Th
5TH	LWR-LEU HWR-Nat.U HWR-SEU HWR-MEU(233)/Th	HWR-Pu/Th
5TG	LWR-LEU HTGR-LEU HTGR-MEU(233)/Th	HTGR-Pu/Th

[a]Denatured ^{233}U fuel.

respectively, are plotted in Fig. 6.1. For most of the analysis, the available U_3O_8 supply in either model was restricted to that costing less than $160 per pound; however, in order to determine how much ore would be required for all the options to meet the projected nuclear demand, additional calculations were performed in which the price constraint was removed from both models.

In addition to the limitations on the resource base per se, it was assumed that it would be difficult for the U.S. to mine and mill more than 60,000 ST of U_3O_8 per year in the 1990s. Although the combined maximum capability of a coalition of states could be greater, there is

Table 6.1c. Descriptions of Denatured Nuclear Power Systems Using Both Thermal and Fast Reactors

Case	Dispersed Reactors	Energy-Center Reactors
Option 6: Denatured ACRs with FBR Transmuters		
(Light Pu-to-^{233}U transmutation rate realized.)		
6L	LWR-LEU LWR-MEU(235)/Th LWR-MEU(233)/Th	LWR-Pu/Th FBR-Pu/U/Th[a]
6S	LWR-LEU LWR-MEU(235)/Th SSCR-LEU SSCR-MEU(233)/Th	SSCR-Pu/Th FBR-Pu/U/Th
6H	LWR-LEU HWR-Nat.U HWR-SEU HWR-MEU(235)/Th HWR-MEU(233)/Th	HWR-Pu/Th FBR-Pu/U/Th
6G	LWR-LEU HTGR-LEU HTGR-MEU(235)/Th HTGR-MEU(233)/Th	HTGR-Pu/Th FBR-Pu/U/Th
Option 7: Denatured ACRs/FBRs with FBR Transmuters		
(Light Pu-to-^{233}U transmutation rate realized.)		
7L	LWR-LEU LWR-MEU(235)/Th LWR-MEU(233)/Th FBR-MEU(233)/Th	LWR-Pu/Th FBR-Pu/U/Th
7S	LWR-LEU LWR-MEU(235)/Th SSCR-LEU SSCR-MEU(233)/Th FBR-MEU(233)/Th	SSCR-Pu/Th FBR-Pu/U/Th
7H	LWR-LEU HWR-Nat.U HWR-SEU HWR-MEU(235)/Th HWR-MEU(233)/Th FBR-MEU(233)/Th	HWR-Pu/Th FBR-Pu/U/Th
7G	LWR-LEU HTGR-LEU HTGR-MEU(235)/Th HTGR-MEU(233)/Th FBR-MEU(233)/Th	HTGR-Pu/Th FBR-Pu/U/Th
Option 8: Denatured ACRs/FBRs with FBR Transmuters Containing Th in Their Cores		
(Heavy Pu-to-^{233}U transmutation rate realized.)		
8L	LWR-LEU LWR-MEU(235)/Th LWR-MEU(233)/Th FBR-MEU(233)/Th	LWR-Pu/Th FBR-Pu/Th/Th[b]
8S	LWR-LEU LWR-MEU(235)/Th SSCR-LEU SSCR-MEU(233)/Th FBR-MEU(233)/Th	SSCR-Pu/Th FBR-Pu/Th/Th
8H	LWR-LEU HWR-Nat.U HWR-SEU HWR-MEU(235)/Th HWR-MEU(233)/Th FBR-MEU(233)/Th	HWR-Pu/Th FBR-Pu/Th/Th
8G	LWR-LEU HTGR-LEU HTGR-MEU(235)/Th HTGR-MEU(233)/Th FBR-MEU(233)/Th	HTGR-Pu/Th FBR-Pu/Th/Th

[a]FBR with Pu/U core, Th blanket.

[b]FBR with Pu/Th core, Th blanket.

no real basis for specifying an upper limit. Recognizing this, the nuclear policy options analyzed were considered to be more feasible if their annual mining and milling rate was less than 60,000 ST of U_3O_8 per year, especially through the 1990s. (Note: The "effective supply" could be increased by assuming a decrease in the ^{235}U content of the uranium tails from the enrichment process. In the standard enrichment process the assay fraction of the tails is 0.0020. The effect of decreasing this assay fraction is discussed in Appendix C.)

Several other assumptions are inherent in the analytical method. For example, for those options that assumed reprocessing, the fuel was stored after discharge until reprocessing and refabrication capabilities were available, and a reactor that required Pu or ^{233}U could not be constructed unless the projected supply of fissile material was sufficient throughout the reactor's lifetime. In addition, a nuclear plant design

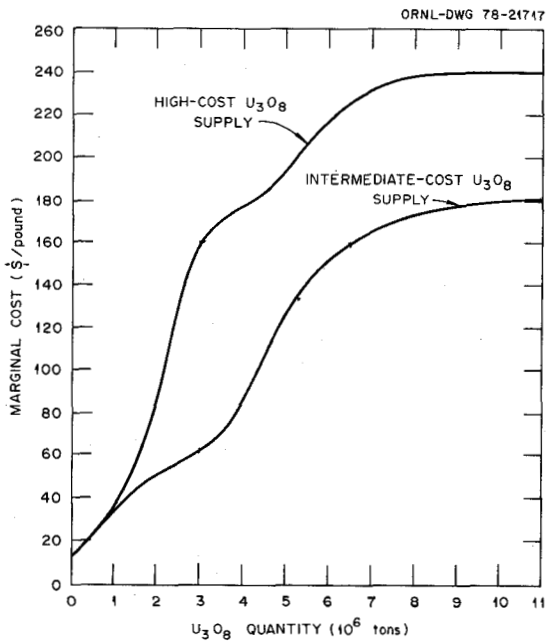

Fig. 6.1. Marginal Costs for High- and Intermediate-Cost U_3O_8 Supply Curves. The increasing costs as a function of the cumulative amount mined is indicative of a continuous transition from higher grade to lower grade resources.

that differed from established technology could be introduced only at a limited rate (typically at a rate of 1, 2, 4, 8, etc. during successive bienniums). And once the manufacturing capability to produce a particular reactor type was established, the rate at which that reactor could lose its share of the new construction market was limited to a specified fraction per year (typically 10% per year).

The analyses also provided detailed cost data, including the total power cost of each case and the total power cost of each reactor type in each case, the latter in turn being subdivided. Unfortunately, however, the combined uncertainties on the costs were so large that any conclusions about the various nuclear policy options based on economics were tenuous at best; therefore, the cost of a nuclear unit usually did not enter into the decision of whether it could be constructed or not. In general, a unit was selected if (1) it was available in the option and (2) it had a lower U_3O_8 consumption rate than other units under the same option. However, for those cases in which the U_3O_8 supply was assumed to be sufficiently large so as not to limit the growth of the nuclear system over the planning horizon, the unit was chosen on the basis of its total power cost.

6.1. Systems with Price-Limited Uranium Supplies

Typical Results

Typical results from the calculations for price-constrained uranium supplies are presented in Figs. 6.2 and 6.3. Figure 6.2 shows that with the currently used LWR-LEU system operating on the high-cost uranium supply (Case 1L), the installed nuclear capacity would peak somewhat above 400 GWe around year 2010. If, however, plutonium recycle were allowed in some of the LWRs (Case 2L), the peak would go above 600 GWe and would be delayed until about year 2020. While both these cases would meet the criterion of 350 GWe in the year 2000, neither could sustain the required growth rate through year 2050.

Adding an LWR that operates on denatured ^{233}U fuel (Case 5TL) would not significantly increase the total installed nuclear capacity above that of Case 2L, nor would it delay the peaking date. However, a distinct difference between Case 2L and Case 5TL is apparent: in Case 2L the system's peak capacity is realized by a sharp increase in the installed capacity within the energy center, the ratio of the outside capacity to the inside capacity (that is, the energy support ratio) being approximately 2.5. By contrast, when the peak capacity occurs in Case 5TL, the energy support ratio is greater than 4 and remains above 4 until after year 2035. By this time the support ratio for Case 2L has decreased significantly. Thus with Case 5TL a much larger fraction of the reactors can be dispersed outside the center. On the other hand, in the near term (to year 2015), there is no noticeable difference in the energy support ratios of the two systems, and if this period were the only period of concern, then Case 2L would clearly be the choice. This statement is based on the fact that to deploy Case 5TL would require a nuclear industry that is capable of reprocessing significant amounts of fuel containing thorium and fabricating significant amounts of fuel containing gamma-emitting ^{232}U, in addition to a Pu recycling capability, whereas Case 2L would require only the plutonium recycling capability.

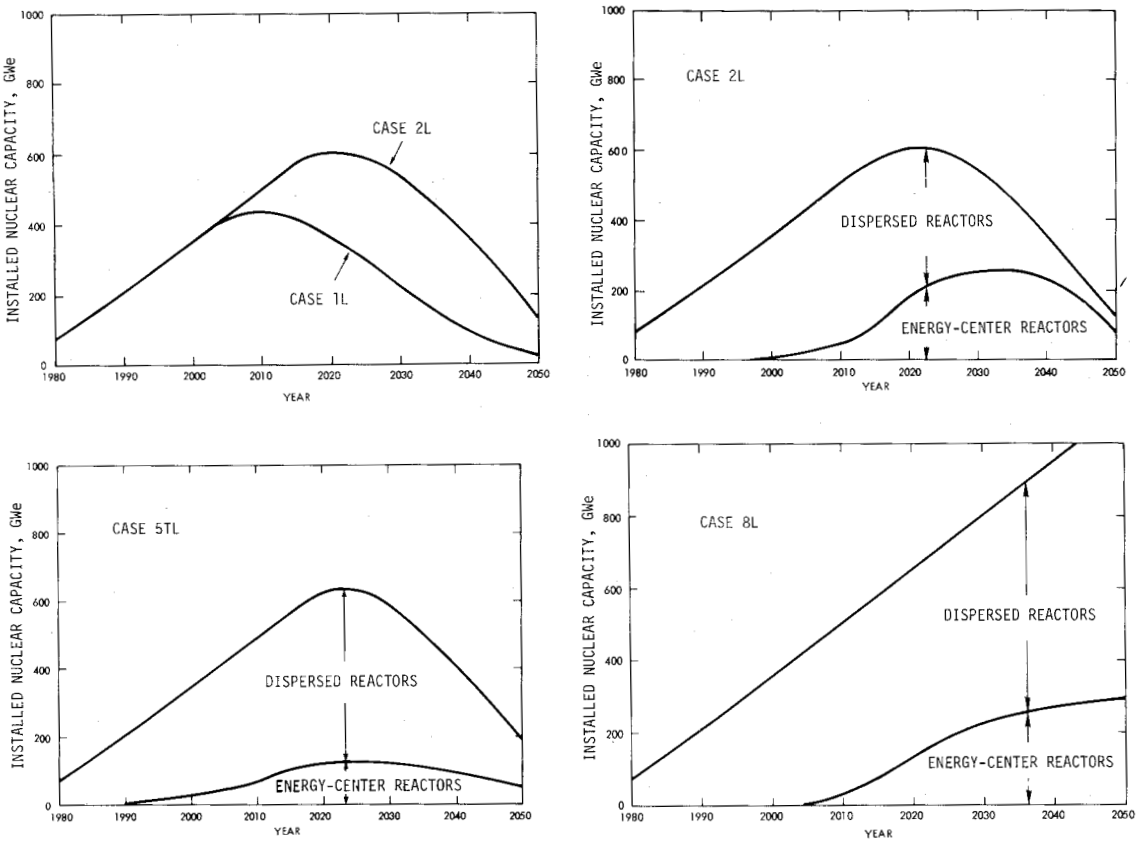

Fig. 6.2. Installed Nuclear Capacities Attainable by Typical Nuclear Power Systems Operating on High-Cost U_3O_8 Supply: Relative Contributions of Dispersed Reactors and Energy-Center-Constrained Reactors.

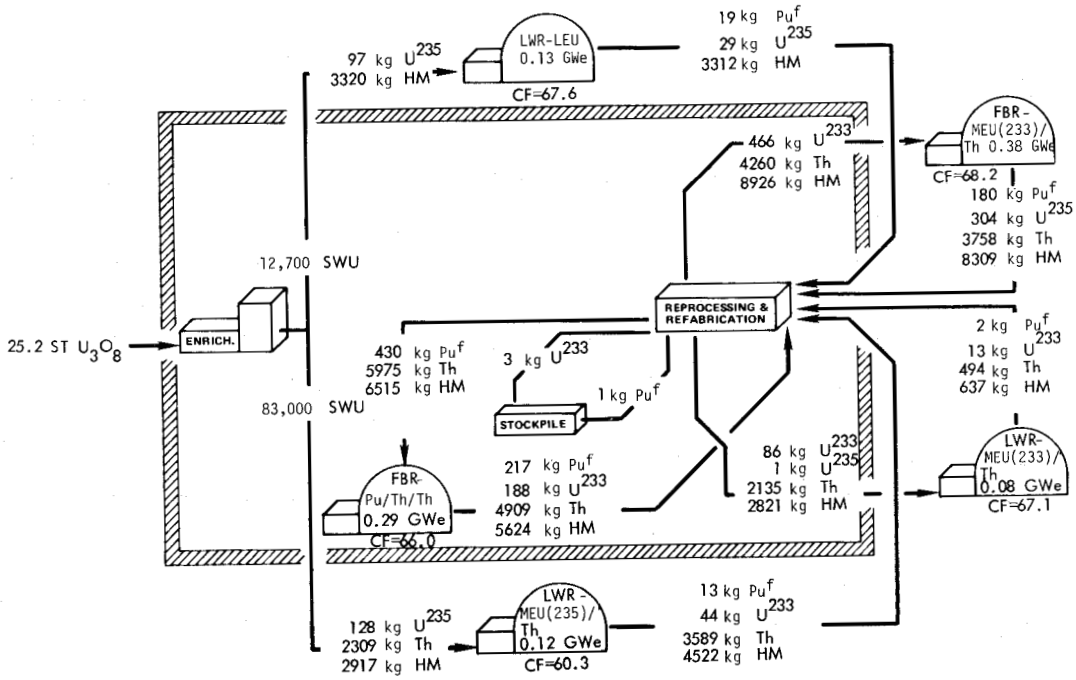

Fig. 6.3. Utilization and Movement of Fissile Material in Typical Power System. (Case 8L, High-Cost U_3O_8 Supply, Year 2035.)

The addition of the FBRs in Case 8L makes a decided impact on the potential of the power system. The nuclear power demand in year 2050 (1100 GWe) is effectively met and the installed nuclear capacity of the system is rapidly rising. Moreover, at year 2050 the energy support ratio of the system is approximately 3 and is increasing.

If these systems were to operate on the intermediate-cost U_3O_8 supply, in which case 6 million ST U_3O_8 would be available to them at costs less than $160 per pound, their installed nuclear capacities would, of course, be much greater. Case 2L would not peak until about year 2040, at which time it would be approaching an installed capacity of 1000 GWe. Case 5TL would peak a few years later with a capacity of about 1000 GWe. The effect on the LEU-LWR throwaway system (Case 1L) would be to increase the peak installed capacity to about 730 GWe and to delay the peak about 20 years.

Since Case 8L is less dependent on the U_3O_8 supply, its installed nuclear capacity for the intermediate-cost fuel supply would not differ greatly from that shown for the high-cost supply; however, the availability of a more abundant economic fuel supply would change the character of the system — to the extent that approximately 35% more U_3O_8 would be used by the system by year 2050 and the energy support ratio in year 2050 would increase from 3 to 5.5.

Figure 6.3 represents a "snapshot in time" for Case 8L, the snapshot occurring during the system's 55th year of operation on the high-cost fuel supply. During the 55th year (year 2035), System 8L would require 25.2 ST of U_3O_8 and 21,000 SWU of enrichment per GWe, and 71% of the installed nuclear capacity would be outside the energy center. The sketch also indicates the flow of fissile material and heavy metal through the system during the year.

Summary of Installed Nuclear Capacities, Energy Support Ratios and Ore/Enrichment Requirements

Summary data for all the options calculated for price-constrained uranium supplies are presented in Tables 6.2 and 6.3. The maximum nuclear capacities that could be attained by all the systems considered are presented in Table 6.2, together with the years in which the maximums would occur. Table 6.3 shows what the energy support ratios of the different systems would be in year 2050.

The effect of varying the fuel cycle can be seen by reading across Table 6.2, and the effect of changing the ACR option within a fuel cycle option can be deduced by reading down a column. For the high-cost U_3O_8 supply, it is obvious that introducing ACRs has little effect on the maximum attainable nuclear capacity of the throwaway/stowaway cycle (Option 1). This is directly due to the introduction dates assumed for the ACRs. By the time the ACRs become predominant in the system, a very significant fraction of the U_3O_8 supply has already been committed to the standard LWR. It follows that with the larger supply of U_3O_8, the ACRs would have a greater impact. For example, with the intermediate-cost uranium supply, an approximately 17% greater installed capacity is attainable when HWRs are added, whereas with the high-cost supply only a 3% greater capacity is attainable.

Table 6.2. Maximum Nuclear Capacity of Various Nuclear Power Options Limited to $160 per pound U_3O_8 and Year in Which Maximum Occurs

(Note: A capacity of 1100 GWe in year 2049 meets demand.)

ACR	Maximum Installed Nuclear Capacity (GWe)/*Year Maximum Occurs*								
	LEU,	Pu/U		Denatured with ACRs			Denatured with ACRs/FBRs		
	1	2	3	4	5U	5T	6	7	8
				With High-Cost U_3O_8 Supply					
LWR (L)	433	611	1100	585	716	637	1100	1100	1087
	2009	*2021*	*2049*	*2019*	*2027*	*2021*	*2049*	*2049*	*2049*
SSCR (S)	440	661	1100	660	820	764	1100	1100	1084
	2009	*2023*	*2049*	*2023*	*2033*	*2029*	*2049*	*2049*	*2049*
HWR (H)	444	630	1100	756	915	856	1100	1100	1100
	2011	*2021*	*2049*	*2031*	*2041*	*2035*	*2049*	*2049*	*2049*
HTGR (G)	437	818	1100	545	671	638	1091	1100	958
	2009	*2033*	*2049*	*2019*	*2023*	*2021*	*2049*	*2049*	*2041*
				With Intermediate-Cost U_3O_8 Supply					
LWR (L)	729	968	1100	1002	1062	1012	1100	1100	1097
	2027	*2041*	*2049*	*2047*	*2049*	*2047*	*2049*	*2049*	*2049*
SSCR (S)	763	1078	1100	1084	1100	1100	1100	1100	1100
	2029	*2049*	*2049*	*2049*	*2049*	*2049*	*2049*	*2049*	*2049*
HWR (H)	852	1062	1100	1084	1100	1100	1100	1100	1100
	2035	*2049*	*2049*	*2049*	*2049*	*2049*	*2049*	*2049*	*2049*
HTGR (G)	783	1100	1100	971	1065	996	1100	1100	1100
	2031	*2049*	*2049*	*2041*	*2049*	*2045*	*2049*	*2049*	*2049*

Several other effects are apparent from Table 6.2. As noted earlier, Pu/U recycle in an all-thermal system (Option 2) significantly increases the attainable peak power over that of the throwaway system and delays the time the peak would be reached. Moreover, although not indicated in the table, this would be accomplished with lower maximum annual U_3O_8 ore requirements. The HTGR case (2G) would provide the greatest level of installed capacity for both uranium supplies, largely because HEU/Th-fueled HTGRs are included and no other reactors used in the study employ HEU fuel. With the intermediate-cost U_3O_8 supply, all cases using ACRs would effectively meet the demand.

As would be expected, with FBRs added to the Pu/U recycle case (Option 3), the system would fully meet the power demands, again (in most cases) with reduced maxiumum annual ore

Table 6.3. Energy Support Ratios in Year 2050 for Various Nuclear Policy Options (Support Ratio = Installed Nuclear Capacity Outside Energy Center/Installed Nuclear Capacity Inside Energy Center)

ACR	Support Ratio								
	LEU,	Pu/U		Denatured with ACRs			Denatured with ACRs/FBRs		
	1	2	3	4	5U	5T	6	7	8
				High-Cost U_3O_8 Supply					
LWR (L)	∞	1.54	0.72	∞	5.69	3.74	1.27	1.46	3.09
SSCR (S)	∞	1.47	0.76	∞	6.33	3.86	2.13	2.13	3.27
HWR (H)	∞	0.49	0.92	∞	5.79	3.07	1.07	1.06	2.89
HTGR (G)	∞	0.24	0.24	∞	4.02	2.50	1.26	1.28	3.11
				Intermediate-Cost U_3O_8 Supply					
LWR (L)	∞	2.42	1.65	∞	5.06	5.05	5.37	5.37	5.49
SSCR (S)	∞	2.10	1.65	∞	4.78	4.78	4.78	4.78	4.78
HWR (H)	∞	1.85	0.94	∞	4.03	3.84	1.03	1.04	3.07
HTGR (G)	∞	1.77	1.82	∞	3.30	3.20	2.74	2.74	3.62

and enrichment requirements. However, because the Pu/U recycle options are so dependent on Pu-fueled reactors that must be confined to energy centers, the energy support ratios for all cases in Options 2 and 3 are relatively low. (Note: It is to be emphasized that the energy ratio of a given system changes with time. The ratios listed in Table 6.3 were selected as "end points" of the analysis, some of them being higher in earlier years and others being lower, as is apparent, for example, in Fig. 6.2.)

The all-thermal denatured systems (Options 4, 5U, and 5T) all have denatured ^{233}U ACRs operating outside the energy center and Options 4 and 5U also have the "denatured ^{235}U" ACRs operating outside the center. The primary difference in the systems is that Option 4 does not utilize Pu whereas Options 5U and 5T include Pu/Th-fueled ACRs in the center. In the case of the high-cost U_3O_8 supply, not recycling Pu causes Option 4 to peak at lower installed capacities and at earlier dates in all cases compared to Options 5U and 5T. And, except for the HWR (Case 4H), the maximum installed capacities and peaking dates for Option 4 do not exceed or are lower than those for Option 2 — the HTGR case much lower. In comparing the ACRs in these cases, it should be noted that the relatively good performance of the HWR (not only in Option 4, but also in Options 5U and 5T) is directly dependent on the denatured ^{233}U HWR, and of all the reactor designs, the design of alternate-fueled HWRs has received the least amount of analysis. Therefore, at this point large uncertainties are associated with the projected performance of the HWR on such fuels. It should also be noted that the relatively poor performance of the HTGR in these options can probably be attributed to the fact that the nonoptimization of the reactors for alternate fuels would have a greater impact on the HTGR than on the other reactors. (This disadvantage is compensated for in Options 2 and 3, each of which includes HTGRs operating on their reference HEU/Th cycle.)

The absence of the "denatured ^{235}U" ACR in Option 5T (compared to Option 5U) reduces the peak installed nuclear capacities with the high-cost U_3O_8 supply in all the cases and causes them to be realized earlier. It also significantly reduces the energy support ratio in year 2050.

With the intermediate-cost U_3O_8 supply, all cases in Options 4, 5U, and 5T are improved, with several meeting the power demand. However, in so doing, these cases will require more than 60,000 ST of U_3O_8 per year in some years, which could limit them, especially if these large demands occurred while 60,000 ST per year was still considered to be the maximum mining and milling rate. On the other hand, if the mining and milling rates could be met, then with the intermediate-cost uranium supply the all-thermal denatured power systems (especially those including plutonium utilization in secure energy centers) could satisfy or almost satisfy the postulated nuclear energy demand through year 2050 at competitive costs.

The denatured power systems utilizing FBRs (Options 6, 7, and 8) would have greater potential nuclear growth rates than the all-thermal denatured systems. With both assumed U_3O_8 supplies, the projected power demand is met in essentially all cases, and those falling short would no doubt meet the demand if slightly improved FBR designs were used. The Th-containing FBRs supporting dispersed denatured ACRs perform as well as the analogous Pu/U cycles within the framework of this analysis. Of these, the FBR with a Pu/U core and Th blanket is particularly resource-efficient.

The denatured power systems utilizing FBRs also have lower U_3O_8 and enrichment requirements than any of the other options. However, with the high-cost uranium supply, and in some cases with the intermediate-cost supply, the energy support ratios in year 2050 are generally lower than those of the all-thermal denatured systems.

6.2. Systems with Unconstrained Uranium Supplies

In order to determine what the ore and enrichment requirements of the various power systems would be under the condition that the projected demand for nuclear power be met, irrespective of the cost, all the cases were recalculated with no cost constraint on the available ore. That is, it was assumed that the power systems could afford any amount of resources they required to meet the demand. Again, however, the calculations were performed for the two different U_3O_8 marginal cost models shown in Fig. 6.1, and for most cases the ore requirements for the two models differed owing to different reactor mixes associated with the two different price structures. High-cost U_3O_8 favored the choice of fuel-efficient (but high capital cost) reactors, whereas lower-cost U_3O_8 favored the continued use of LWRs.

The results from these calculations are summarized in Tables 6.4 and 6.5. It is to be noted that for the LEU-LWR throwaway/stowaway option (1L), a cumulative consumption of 7.1 million ST of ore would be required through year 2049 and the maximum annual consumption would be 183,000 ST U_3O_8. Introducing ACRs on the throwaway cycle would reduce both these requirements, most noticeably with an HWR, and allowing Pu recycle could reduce the cumulative U_3O_8 consumption down to 4 million ST and the maximum annual consumption down to 82,000 ST (Case 2G).

Table 6.4. Cumulative U_3O_8 Consumption of Various Nuclear Policy Options Fully Meeting Projected Nuclear Power Demands

(Restriction to $160 per pound U_3O_8 removed.)

ACR	Cumulative U_3O_8 Consumption (millions of tons) Through year 2025/*Through Year 2049*								
	LEU,	Pu/U		Denatured with ACRs			Denatured with ACRs/FBRs		
	1	2	3	4	5U	5T	6	7	8
				With High-Cost U_3O_8 Supply					
LWR (L)	3.41	2.39	2.14	2.87	2.36	2.36	2.18	2.14	2.29
	7.05	*5.23*	*2.73*	*5.41*	*4.83*	*4.94*	*2.82*	*2.83*	*2.86*
SSCR (S)	3.26	2.23	1.99	2.70	2.35	2.14	1.93	1.93	2.07
	6.52	*4.35*	*2.70*	*4.65*	*3.86*	*3.86*	*2.69*	*2.69*	*2.83*
HWR (H)	3.10	2.72	2.29	2.50	2.16	2.14	2.25	2.21	2.29
	5.58	*4.64*	*2.70*	*4.36*	*3.27*	*3.77*	*2.61*	*2.55*	*2.87*
HTGR (G)	3.23	2.19	1.97	2.58	2.32	2.34	2.15	2.12	2.32
	6.26	*4.04*	*2.75*	*5.13*	*4.43*	*4.94*	*2.70*	*2.68*	*3.18*
				With Intermediate-Cost U_3O_8 Supply					
LWR (L)	3.41	2.39	2.28	2.87	2.36	2.36	2.37	2.37	2.37
	7.05	*5.23*	*4.40*	*5.41*	*4.91*	*4.94*	*4.38*	*4.38*	*4.48*
SSCR (S)	3.26	2.23	2.20	2.70	2.14	2.14	2.14	2.14	2.14
	6.52	*4.35*	*4.14*	*4.65*	*3.86*	*3.86*	*3.86*	*3.86*	*3.86*
HWR (H)	3.10	2.72	2.31	2.94	2.52	2.51	2.32	2.30	2.38
	5.58	*4.64*	*2.71*	*5.40*	*4.32*	*4.37*	*3.66*	*2.70*	*3.37*
HTGR (G)	3.23	2.32	2.30	2.58	3.32	2.34	2.23	2.23	2.26
	6.26	*4.23*	*4.22*	*5.13*	*4.43*	*4.94*	*4.19*	*4.19*	*4.24*

The denatured thermal options with Pu recycle (5U and 5T) would reduce the cumulative U_3O_8 consumption even further (but not always the maximum annual consumption). When the high-cost uranium supply is assumed, the preferred ACR is again the HWR, but when the intermediate-cost supply is assumed, the preferred ACR is the SSCR. When Pu is recycled in the denatured thermal options, the total resource requirements are generally less than those for thermal systems operating solely on the Pu/U recycle mode (the HTGR cases are the exception).

Table 6.5. Maximum Annual U_3O_8 Requirements of Various Nuclear Policy Options Fully Meeting Projected Nuclear Power Demands

(Restriction to $160 per pound U_3O_8 removed.)

ACR	Maximum U_3O_8 Consumption (thousands of tons per year)								
	LEU,	Pu/U		Denatured with ACRs			Denatured with ACRs/FBRs		
	1	2	3	4	5U	5T	6	7	8
				With High-Cost U_3O_8 Supply					
LWR (L)	183	120	60	111	115	115	62	60	68
SSCR (S)	160	115	52	83	83	83	50	50	55
HWR (H)	120	83	66	78	62	69	64	63	65
HTGR (G)	140	82	53	105	96	115	61	60	65
				With Intermediate-Cost U_3O_8 Supply					
LWR (L)	183	120	92	111	117	115	86	86	92
SSCR (S)	160	115	93	83	83	83	83	83	83
HWR (H)	120	83	66	110	89	90	66	66	66
HTGR (G)	140	86	86	105	96	115	87	87	87

The power systems that include FBRs (Options 3, 6, 7, and 8) have considerably more flexibility and reduced cumulative ore requirements and consumption rates. When the U_3O_8 costs are high, these options reduce their U_3O_8 requirements by increasing the fraction of FBRs in their reactor mix.

In summary, to completely satisfy the projected demand for nuclear power through year 2050, LEU throwaway systems would require 5.6 to 7.1 million ST U_3O_8, thermal recycle systems would require 3.3 to 5.4 million ST, and FBR-containing systems would require 2.6 to 4.4 million ST, the systems including denatured ^{233}U reactors requiring approximately the same cumulative amount of U_3O_8 as their Pu/U counterparts. These results qualitatively support those obtained from the earlier cost-constrained cases, although lower U_3O_8 supplies were available for the earlier calculations.

6.3. Conclusions

Under the assumptions that the projected nuclear power demand is 350 GWe in the year 2000 with an increase of 15 GWe/year thereafter to the year 2050, and that nuclear power would not be competitive at U_3O_8 prices exceeding $160/lb, the adequacy of several pos-

tulated denatured and nondenatured nuclear power systems based on the secure energy center and dispersed reactor concept were analyzed. The results showed the following:

- If nuclear power systems were limited to the once-through LEU cycle and to 3 million ST U_3O_8 below \$160/lb, the U.S. nuclear power capacity would peak around the year 2010. If 6 million ST U_3O_8 below \$160/lb were available, the peak would be higher and would be delayed 20 to 25 years.
- With the limitation of 3 million ST U_3O_8 below \$160/lb, all once-through LEU systems, regardless of the reactor types employed, would result in approximately the same maximum installed nuclear capacity (about 440 GWe). With 6 million ST U_3O_8 below \$160/lb available, adding ACRs to the cycle (i.e., SSCRs, HWRs, or HTGRs) would increase the maximum installed nuclear capacity above that of an all-LWR system.
- Thermal Pu/U recycle systems have the capability of increasing the maximum installed nuclear capacity over the once-through cycle. Under the limitation of 3 million ST U_3O_8 below \$160/lb, the best thermal Pu/U recycle system could support twice the maximum installed capacity of the once-through cycle. With 6 million ST U_3O_8 below \$160/lb, some thermal Pu/U recycle systems could support the nuclear demand.
- With fast breeders added to the Pu/U recycle system, the nuclear power demand could be fully met under both ore supply assumptions. However, all Pu/U recycle systems would have relatively low energy support ratios because all Pu-fueled reactors would be restricted to the energy center.
- Thermal recycle systems that include denatured ^{233}U reactors would have the capability of supporting more installed nuclear capacity than thermal Pu/U recycle systems; however, achieving this capability would require Pu utilization.
- Thermal recycle systems that include denatured ^{233}U reactors and utilize Pu could attain relatively high energy support ratios, especially if the systems also included "denatured ^{235}U" reactors.
- Essentially all systems that use fast transmuters to produce ^{233}U for denatured thermal reactors could fully meet the projected nuclear power demand under both U_3O_8 supply assumptions. However, because of their dependency on Pu utilization, the systems limited to 3 million ST U_3O_8 below \$160/lb would have relatively low energy support ratios.
- To completely satisfy the projected nuclear power demand, LEU throw-away/stowaway systems would require 5.6 to 7.1 million ST U_3O_8, thermal recycle systems (both denatured and nondenatured) would require 3.3 to 5.4 million ST, and FBR-containing systems would require 2.6 to 4.4 million ST.

7.0. CONCLUSIONS AND RECOMMENDATIONS

Depending on the degree to which the proliferation concern is addressed, various nuclear power strategies could be developed between the current no-reprocessing option (and hence no recycle) and options that would permit the unconstrained recycle of plutonium. A strategy based on the denatured ^{233}U fuel cycle would retain important advantages of both these extremes: It would employ recycled fissile material, and thus would extend the effectiveness of the resource base. At the same time it would include reactors whose fresh fuel would have an inherent isotopic barrier, plus a radiation barrier, that would be difficult to circumvent. The cycle would also have an added advantage in that extracting weapons-usable material from the spent denatured fuel would be considerably more difficult than extracting an equivalent amount of fissile material from spent LEU fuel. And while some components and facilities in the denatured ^{233}U cycle would not have inherent protection factors, they could be restricted to secure (guarded) energy centers.

Before any proposed new fuel cycle can be implemented, however, it must be examined in the light of practical considerations, such as when the necessary reactors and fuels could be on line, whether an economic supply of U_3O_8 would be available, and how the technical or institutional barriers required to ensure nonproliferation could be practically implemented. In this interim study of the denatured cycle all these factors were considered insofar as possible within the constraints of the study, and several postulated nuclear power systems utilizing denatured ^{233}U fuel were analyzed to determine whether they could meet a projected U.S. nuclear power growth demand of 350 GWe in the year 2000 followed by a net increase of 15 GWe/year up to the year 2050. Two different U_3O_8 supply models were employed: one that assumed that only 3 million ST U_3O_8 would be available at an economically competitive cost of $160/lb, and another that assumed that 6 million ST U_3O_8 would be available. So that the denatured cycle could be compared with other fuel cycles, similar analyses were performed for a number of nondenatured power systems.

From the perspective of an overview, the various nuclear power systems can be classified under three major categories: (a) no-recycle options, (b) classical recycle options, and (c) denatured recycle options. These, in turn, can be subdivided into options that utilize LWRs only and those that utilize LWRs in combination with ACRs and/or FBRs. An integrated assessment of these various options is presented in matrix form in Table 7.1, with each option characterized on the basis of the following criteria:

(1) Nuclear proliferation resistance relative to other systems.

(2) Potential for commercialization of the reactor/fuel cycle components.

(3) Technical feasibility on a reasonable schedule (and at reasonable costs) for research, development, and demonstration of the reactor/fuel cycle components.

(4) Capability of the system for meeting long-term nuclear energy demands.

(5) Economic feasibility.

7.1. No-Recycle Options

Case A in Table 7.1 is the currently employed once-through low-enriched uranium cycle in LWRs, which represents the only significant commercial possibility in the near term. At current ore and separative work prices, this cycle is economically competitive with other energy sources, and it has favorable proliferation-resistance characteristics: its fresh fuel contains an inherent isotopic barrier; and while its spent fuel contains plutonium, the fuel is contaminated with highly radioactive fission products and thus has a radiation barrier. The principal drawback of the cycle is that to satisfy the nuclear demand postulated in this study would require the consumption of 5.6 to 7.1 million tons of U_3O_8. Also it would require that 90,000 to 130,000 tons of U_3O_8 be mined and milled annually, which under current capabilities seems unfeasible. Possibilities exist for reducing these requirements (see Appendix C), but even with improvements, the cycle would be limited by the availability and producibility of U_3O_8 in the next century.

In Case B, ACRs (i.e., HWRs, HTGRs, or SSCRs) operating on once-through LEU or MEU(235)/Th fuel would be added to the LEU-LWRs already commercialized. When operated on the once-through LEU cycle, all the ACRs considered in this study would utilize less U_3O_8 than LWRs (see Table 4.1), particularly the HWR, for which the uranium would be only slightly enriched. When operated on the MEU(235)/Th once-through cycle, the HTGR also would use less fuel than the LWR on the LEU once-through cycle. Thus the substitution of ACRs on LEU fuel and possibly HTGRs on MEU(235)/Th fuel would be resource efficient. However, as made clear in Section 5, considerable effort and expenditures would be required to commercialize the ACRs, and if MEU(235)/Th fuel were to be used, additional fuel R,D&D would be necessary. And even then the generic drawback of once-through cycles would remain — that is, the uncertainty in the size of the economically recoverable resource base. On the other hand, as costs for extracting the resource base increase (to above \$100/lb U_3O_8, for example), commercialization of the ACRs would become more attractive. [The implementation of MEU(235)/Th in LWRs would be uneconomic because of high fissile loading requirements and would not be considered except to initiate a stockpile of ^{233}U.]

If either the continuation of Case A or the implementation of Case B is adopted as a long-term policy, plans should be included to provide centralized and secure regions within which spent fuel could be stored and enrichment facilities could be operated. As time passes, safeguarding the spent fuel discharged from operating reactors will assume greater importance since all the elements will contain Pu^f that will become increasingly accessible as the fission-product radioactivity of the spent fuel decays. [The MEU(235)/Th elements, however, would have much less Pu than the LEU elements.] And in the event that recycling were eventually decided upon, such centers would be ready sources of Pu^f and ^{233}U, as well as forerunners of the fuel cycle energy centers for recycle-based options.

7.2. Classical Reference Recycle Options

If growth of nuclear-based electrical generation is to be sustained indefinitely, the breeding and recycling of artificial fissile material will be mandatory. With the growth of the nuclear industry, it has been assumed that the Pu now being produced in the ^{238}U contained in the LWR fuel elements eventually would be chemically extracted from the spent

Table 7.1. Integrated Assessment of Various Nuclear Policy Options for Meeting Projected U.S. Nuclear Power Growth Demand

Reactor/Fuel Cycle Combination	Proliferation Resistance	Implementation/Commercialization	R,D&D Cost and Time of Commercial Introduction	Ability to Meet Power Demands	Economics
			No-Recycle Options		
A LWRs on LEU cycle	• Probably best to the extent that non-nuclear weapons states continue to forego national fuel recycle • Fresh fuel has isotopic barrier; spent fuel contains radioactive fission products • Spent fuel stockpile containing Pu is a risk; requires institutional barriers	• In wide commercial use • Concern exists about fuel supply • Emphasis on improved LWRs and U_3O_8 resource development needed	• Low cost • Gradual improvements introduced from year 1980 to year 2000	• Least resource efficient • Peaks out between years 2010 and 2030 and declines thereafter unless large amounts of low-grade U_3O_8 are exploited • Peak could be increased and delayed 10 to 15 years with reactor improvements and reduced tails assay	• Economics closely linked to U_3O_8 price • Very favorable at current U_3O_8 prices
B LEU-LWRs followed by advanced converters on LEU (SEU) cycle or on MEU(235)/Th cycle	• Similar to above • HTGRs on MEU/Th cycle would reduce Pu production by factor of 5 over LEU-LWRs but fresh fuel would have higher ^{235}U content (20%) • HWRs on SEU cycle about equal to LWRs on LEU cycle in Pu production	• Little commercial incentive to introduce advanced converter • Known to be technically feasible • Concern exists about long-term fuel supply	• Up to $2 billion for advanced converter R,D&D • Advanced converters introduced in 1990's	• Advanced converters could extend usefulness of once-through cycle up to 10 years over standard LWRs	• Uncertain capital costs cloud near-term interest • Advanced converters favored at high U_3O_8 prices (>$100/lb)
			Classical Reference Recycle Options		
C Once-through LEU-LWRs followed by LWRs with Pu recycle	• Recycled Pu in fresh fuel chemically separable; probably acceptable if Pu can be limited to nuclear weapons states and to secure international fuel service centers • Option requires technical and institutional barriers for Pu-fueled reactors (∿30%) • Spent fuel contains radioactive fission products	• Acceptable to private sector • Requires completion of Generic Environmental Impact Statement on Mixed Oxide Fuel	• Over $1 billion, mainly for fuel cycle R&D • Introduction in late 1980's	• Gains 10-15 years relative to Case A; somewhat less relative to improved A	• Preferred over Case A at high U_3O_8 (>$100/lb)
D Once-through LEU-LWRs followed by LWRs and FBRs with Pu recycle	• Increased risk over Case C because system tends to become Pu dominated • Leads to significant Pu inventories and requires extensive Pu transportation for dispersed reactors • Requires technical and institutional barriers	• Preferred by private sector • FBR licensing and commercialization may be difficult	• FBR R,R&D up to $10 billion • Fuel cycle R,D&D $1.6 to $3 billion • FBRs not available before 2000	• Superior ability to respond to power growth greater than that considered in this study • Divorce from mining possible	• Economics uncertain because of FBR costs, but probably acceptable
			Denatured Recycle Options		
E Dispersed LWRs operating on LEU and denatured ^{233}U fuel with U recycle; energy-center thermal transmuters (LWRs) with Pu recycle	• "Fresh" denatured fuel has isotopic and radioactive barriers; spent fuel contains radioactive fission products • Spent denatured fuel contains less Pu than spent LEU fuel (factor of 2.5 less) • Requires technical and institutional barriers to limit Pu to secure energy centers • Reduces Pu-fueled reactors by factor of 2 compared with Case C	• Fuel cycle somewhat more complex than Pu/U cycle, but functionally equivalent • Requires government incentive	• Up to $0.5 billion, PWRs and BWRs • Fuel cycle R,D&D $1.8 to $3.3 billion • Introduction in 1990's	• Somewhat better than Case C due to superiority of ^{233}U as thermal reactor fuel	• Close to Case C
F Dispersed LWRs and advanced converters operating on LEU and denatured ^{233}U fuel with U recycle; energy-center thermal transmuters (LWRs and advanced converters) with Pu recycle	• Fresh and spent denatured fuel advantages same as for Case E • Requires technical and institutional barriers • Use of HWRs or HTGRs substantially reduces Pu production relative to Cases C and E • Pu produced in denatured HWRs and HTGRs may be discarded with minor loss of fuel efficiency	• Same as Case E • Advanced converters likely to to be attractive if FBRs are unavailable	• Up to $2.5 billion for advanced converters • Fuel cycle same as in Case E • Introduction in late 1990's	• Can fully satisfy assumed demand through year 2050 for plentiful U_3O_8 supply; especially true if HWR converters used	• Possibly lowest cost for U_3O_8 price range of $100-$200/lb, especially for HTGR converter
G Dispersed LWRs and advanced converters operating on LEU and denatured ^{233}U fuel U recycle; energy-center fast transmuters with Pu recycle	• Very similar to Case E except that 15 to 50% of reactors may be Pu-fueled FBRs, depending on choice of cycles	• Same as Case E • Private sector likely to accept government mandate • Should be structured for maximum thermal-to-fast reactor ratio to allow siting flexibility	• Up to $10 billion for FBRs • Converter R,D&D as in Cases E and F • Fuel cycle $2 to $3.6 billion • Introduction after year 2000	• As good as Case D above for assumed power demand • Divorce from U mining less likely than for Case D above	• Economics similar to Case D above • If FBR costs are high, can compensate by reducing the fraction of FBRs in the mix and increasing the mining rate

elements and used in the fabrication of replacement LWR cores, and as the Pu/U cycle matured, FBRs would be added. This progression is represented by Cases C and D in Table 7.1.

While recycling Pu in LWRs alone would extend the usefulness of the uranium resource base, Case C, like other all-thermal systems, would be inherently limited in that the amount of fissile material produced (Pu^f) would always be less than the amount of fissile material burned up (^{235}U). The overall effect would be to gain several years over Case A and a few years over Case B. However, this option has the advantage that it could be implemented earlier than the Case B option since it requires only that the already well-advanced Pu/U fuel cycle R&D be completed and commercialized. The perceived disadvantage of the system is that for it to be proliferation resistant the Pu-fueled reactors (on the order of 30%) and essentially all the other components of the fuel cycle would have to be located in secure energy centers. As a result, the system's energy support ratio (ratio of power produced outside the center to the power produced inside the center) would be relatively low. And since energy centers could not be sited as conveniently as single reactors, long-distance electric power transmission could both decrease the efficiency of the system and increase the costs. Still, if U_3O_8 prices were to increase to more than $100/lb, Case C could be preferred over Case A.

With fast breeders included in the Pu/U cycle (Case D), the projected power demands could be fully met and a positive growth rate could be anticipated far into the future. The system would become self-sustaining and eventually could be divorced from the uranium resource base, thus eliminating the necessity for further mining. However, commercialization of the FBR probably could not be accomplished before the year 2000, and large costs to complete the reactor and fuel cycle R,D&D could be expected. Also, this option would tend to have an even lower energy support ratio than Case C.

7.3. Denatured Recycle Options

The three denatured recycle options, Cases E, F, and G in Table 7.1, are all basically the same, differing primarily in the reactor mix utilized. In each case the system is structured with proliferation resistance as a primary criterion and it relies heavily on the energy-center and dispersed-reactor concept. A large fraction of the reactors (up to 85%) utilize denatured ^{233}U or LEU fuel and thus can be dispersed outside the energy center to locations where they are most needed (see Fig. 3.1). Components located in the energy center would include Pu-fueled thermal or fast transmuters dedicated to the production of ^{233}U plus all the facilities required for fuel fabrication, reprocessing, etc. Thus the Pu would be restricted to and destroyed within the energy center and all the fresh fuel outside the center would have isotopic barriers that would preclude isolation of the ^{233}U or ^{235}U through chemical processing. In addition, the fresh denatured ^{233}U fuel would have a radioactive barrier due to the decay daughters of the ^{232}U impurity that is unavoidably produced along with the ^{233}U. Although the spent denatured fuel would contain Pu, the amount would be less than one-half that in spent LEU fuel. Moreover, it would have the usual protection of fission-product radioactivity while it was being returned to the center.

Case E is a denatured recycle option that would utilize LWRs only. In this respect it is comparable to Case C; however, in Case C the LWRs would be using only LEU and Pu/U fuel, whereas in Case E they would be using LEU, MEU(233)/Th (i.e., denatured ^{233}U), and Pu/Th fuel. As a result, the fuel cycle would be more complex and its R,D&D costs would be higher, partially because of the necessity for developing remote operations to handle the radioactive ^{233}U (+ ^{232}U) fuel. Also, in order for the LWRs to accommodate MEU(233)/Th and Pu/Th fuel, additional LWR R&D would be required. An advantage that this case has over Case C is that ^{233}U is superior to either ^{235}U or Pu as a fuel for thermal reactors and thus the system probably would better meet the projected power demand. However, like Case C, this option is inherently limited by the absence of FBRs. Also, with the R,D&D still required, the option would require approximately 10 more years than Case C for deployment, and then only with strong government incentives and support.

Case F differs from Case E in that some of the LWRs both inside and outside the energy center would be replaced with ACRs; thus, before this option could be made available, the R,D&D of the ACRs would have to be completed. This would considerably increase the costs of the system, as well as the requirements for government support, but at the same time the projected power demand could probably be fully met. In fact, if the U_3O_8 price were to increase to $100 to $200 per pound, this option would be economically attractive compared to the preceding options. Still it would be an all-thermal system that would be using more fuel than it produced, and its long-range feasibility would be intimately tied to the recovery costs of the uranium resource base.

In Case G the energy center would utilize fast transmuters rather than thermal transmuters. This system could fully meet the projected power demand, and thus in energy production it would be equivalent to the classical FBR Pu/U option (Case D), although independence from the resource base would not be as probable as it would be for Case D. Because this system would use essentially all the reactor and fuel types considered in this study, its deployment would require that all the reactor R,D&D and all the fuel cycle R,D&D mentioned for the other cycles be carried out. As a result, its costs would be higher than those for Case D and its implementation would require a strong government mandate. If it were implemented, however, its proliferation-resistance characteristics would allow a large fraction of the power-producing reactors to be dispersed outside the energy center to locations where they were most needed, whereas in Case D most of the reactors would be restricted to the center.

As was stated in Chapter 6, it was not possible in this study to evaluate the denatured ^{233}U fuel cycle or any of the other cycles in detail on the basis of economics due to the uncertainties in unit cost factors. However, the economics of the denatured cycle appear to be equivalent to, or slightly better than, those of the classical Pu/U cycle for moderate growth-rate scenarios (that is, scenarios that would require the use of fast and thermal reactors in combination). While the R,D&D costs and fuel cycle unit costs of the denatured cycle were assumed to be higher than those of the Pu/U cycle, power systems utilizing denatured ^{233}U fuel typically would allow a larger fraction of the reactors to be thermal reactors (LWRs or ACRs), which would have lower capital costs than fast reactors. This is directly due to the fact that ^{233}U can be used in thermal reactors more efficiently than in fast reactors.

Neither did this study single out any one ACR as an obvious selection for further development and conversion to alternate fuels (MEU/Th and Pu/Th fuels). Both the HWR and the HTGR, particularly the HWR, appear to have certain superior fuel utilization characteristics relative to the SSCR or LWRs (see Section 4). But the SSCR could be deployed faster and with significantly lower R,D&D costs, the more so if the PWR on which the SSCR design is based had already been converted, which it undoubtedly would be. As discussed in Section 5, developing an alternate-fueled PWR would be much less difficult than developing an ACR due to the backlog of LWR experience and the reduced risk associated with a previously demonstrated reactor system. And the capital cost of an alternate-fueled LWR would be somewhat lower than the capital cost of an ACR. Thus, the improved performance of an ACR must be weighed against the increased R,D&D and capital costs and the delay in introduction.

The reactor data in Section 4 and the system analyses in Section 6 indicate that fast transmuters would have more favorable resource characteristics as ^{233}U producers than would thermal transmuters. The logical transmuter candidate would be a Pu-fueled fast reactor with a thorium blanket. It should be noted, however, that a more rapid growth in energy demand could dictate that classical Pu/U breeders also be included in the system or even that fast reactors operating on denatured ^{233}U be used. In these cases the nuclear power capacity could grow independently of the resource base.

In summary, the denatured cycle appears to possess advantages relative to the Pu/U cycle, but several important areas require further study. In particular, the refinement of the denatured ACR characterization is of prime importance, both to evaluate various reactor options and to study the overall use of ACRs as opposed to LWRs. Also, system interaction studies for the dispersed denatured reactors and centralized transmuters require refinement based on improved reactor designs and updated mass balances. Finally, the question of implementing the energy-center concept, together with the use of specially designed transmuters as a source of denatured fuel, deserves more detailed study. Characterizations of improved fast transmuters, improved LWRs, reoptimized ACRs and LMFBRs, as well as a characterization of the Light Water Breeder Reactor (LWBR), have been developed under the Nonproliferation Alternative Systems Assessment Program (NASAP) and a DOE Proliferation Resistant Large Core Design Study (PRLCDS) and should be utilized in any further studies that are performed.

7.4. Overall Conclusions and Recommendations

The denatured ^{233}U fuel cycle emerges from this assessment as a potential alternative to the conventional Pu/U cycle, with advantages that can be characterized as follows:

- The denatured ^{233}U cycle offers proliferation-resistance advantages relative to the Pu/U cycle in that: the fresh denatured fuel would have an isotopic barrier that would preclude isolation of the ^{233}U through chemical processing; the fresh fuel would have a radioactivity barrier due to the daughter products of its ^{232}U impurity; and the spent fuel would contain relatively small amounts of Pu. By contrast, the Pu/U cycle, especially when including fast breeder reactors, would tend

toward an equilibrium in which all the fresh fuel would contain chemically extractable Pu and the spent fuel would contain increasing amounts of Pu.

- Because ^{233}U is a more efficient fuel for thermal reactors than either ^{235}U or Pu, power systems employing denatured ^{233}U fuel could meet moderate growth-rate demands with a larger fraction of thermal reactors than power systems based on the Pu/U cycle. This would tend to minimize the overall capital costs of the power system since thermal reactors have significantly lower capital costs than fast reactors.
- If denatured power systems were to include ACRs as well as LWRs, the dependence on fast reactors could be further minimized due to the improved resource utilization of ACRs compared to LWRs. The degree of economy would depend, of course, on the reactor mix since the ACRs would have higher capital costs than the LWRs.
- Under the mandate of a proliferation-resistant system based on the secure energy-center and dispersed-reactor concept, denatured power systems could be divorced from the resource base and still support dispersed reactors whereas power systems operating on the Pu/U cycle alone could not.

The disadvantages of the denatured ^{233}U cycle are the following:

- The cycle would be more complex than the Pu/U cycle, and since the required ^{233}U must first be produced in transmuters, the rate at which reactors fueled with denatured ^{233}U could be introduced would be inherently limited. The Pu/U cycle is closer to commercialization and Pu is already being produced in currently operating reactors.
- Because the Pu/U cycle technology is well advanced, it is the preferred cycle both of the U.S. industry and foreign governments; therefore, their reluctance to embrace an alternative which is less developed and is considered primarily on the basis of its nonproliferation advantages would have to be overcome.
- The R,D&D costs for developing the denatured ^{233}U fuel cycle would be significantly higher than those for the Pu/U cycle. If ACRs were also required, even higher costs would be incurred.

Other important conclusions from this study are as follows:

- The LWR-LEU once-through cycle is likely to dominate nuclear power production through the year 2000, which should provide time to develop either the denatured cycle or the Pu/U cycle for the recycle mode.
- Denatured ^{233}U fuel can be used in LWRs, SSCRs, HWRs, HTGRs, and FBRs without major changes from their present conceptual designs.

- After the necessary R,D&D is completed, the denatured ^{233}U fuel cycle appears to be economically competitive with the Pu/U fuel cycle.
- With the fuel resources assumed, the nuclear power demand postulated in this study (350 GWe in the year 2000 and a net increase of 15 GWe/yr thereafter) can be met as well by power systems operating on the denatured fuel cycle as it can by power systems using the Pu/U cycle. However, the Pu/U cycle with FBRs has an inherent ability to grow at a faster rate than the other cycles.

On the basis of this study, it is recommended that:

- Optimized designs of improved LWRs, ACRs, and fast reactors operating on alternate fuels (specifically denatured ^{233}U fuel and Pu/Th fuel) be examined to refine the characteristics of the denatured cycle relative to fuel utilization, economics, and energy-support ratio. The study should also be expanded to include LWBRs and the fast breeder designs developed by DOE in the Proliferation Resistant Large Core Design Study (PRLCDS). More detailed assessments of the proliferation risks and the economics of the denatured cycles compared to other recycle options (Pu/U and HEU/Th) should also be pursued.

These further studies could provide guidance for the following R&D programs:

- Thorium fuel cycle R&D to investigate the use of MEU(235)/Th, MEU(233)/Th (denatured ^{233}U), and Pu/Th fuels in LWRs and HWRs (the latter in cooperation with Canada). This program might also include the LWBR fuel cycle.
- Studies to consider denatured ^{233}U or ^{235}U fuels as candidates for the HTGR reference fuel cycle.
- Thorium technology studies, particularly for blanket assemblies, as an integral part of the LMFBR program and the GCFBR program (Gas Cooled Fast Breeder Reactor).
- Exploratory work with utilities and PWR and BWR vendors for qualification and use of MEU/Th and Th fuel rods in commercial reactors. An example of the beneficial use of Th would be in corner rods of the BWR fuel assembly.

APPENDIX A. ^{232}U PRODUCTION AND DECAY PROCESSES

The production of ^{233}U from thorium results in the concomitant production of ^{232}U (see Fig. A.1) which probably would not be isotopically separated from the ^{233}U before the fresh denatured fuel was fabricated. As the ^{232}U decays through ^{228}Th and its daughter products to stable ^{208}Pb (see Fig. A.2), numerous gamma rays would be emitted, the most prominent being a 2.6-MeV gamma ray associated with the decay of ^{208}Tl. Thus the fresh denatured ^{233}U fuel would be radioactive, and would be increasingly more radioactive with the passing of time. This characteristic of the fuel would have several ramifications, both with respect to proliferation and to the development of the fuel cycle, as has been discussed in Section 2.

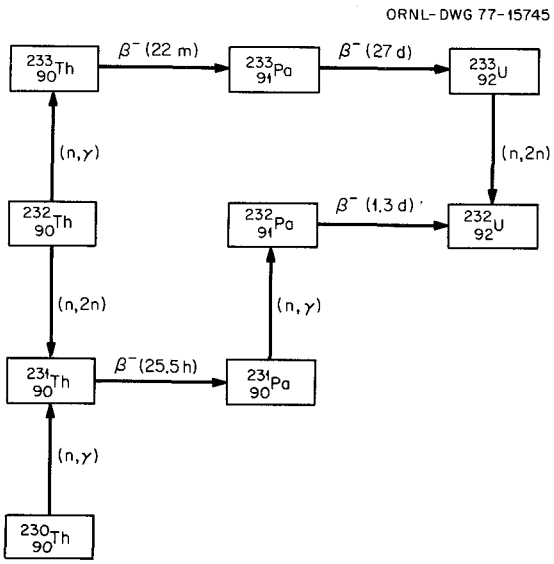

Fig. A.1. Important Reaction Chains Leading to the Production of ^{232}U and ^{233}U.

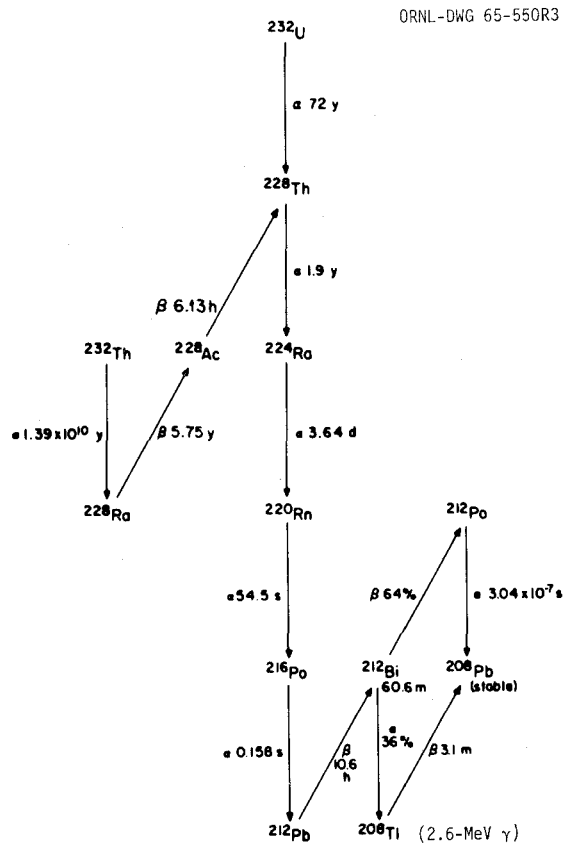

Fig. A.2. Decay of ^{232}U and ^{232}Th.

APPENDIX B. TIME-INTEGRATED DOSES DUE TO INHALED U AND Pu ISOTOPES

Estimates of the time-integrated doses (50-yr doses) that can be expected to be delivered to the bone by the inhalation of the important fuel isotopes are compared in Table B.1. Although the values given for ^{232}U and ^{233}U are based on limited experimental data, it is apparent that, in terms of dose per μg inhaled, the toxicity of ^{232}U is higher than that of any of the other isotopes listed except ^{238}Pu. When compared with ^{235}U or ^{238}U, ^{233}U also has a relatively high toxicity, but one that is considerably lower than the toxicity of any of the plutonium isotopes. It should be pointed out, however, that these toxicities are based on the assumption that all bone-seeking radionuclides are five times more effective in inducing bone tumors than is ^{226}Ra, and some evidence exists that ^{232}U and ^{233}U are not that damaging.

Table B.1. Time-Integrated (50-yr) Dose Deliveries to Bone by Inhaled Uranium and Plutonium Isotopes

Isotope	Specific Activity[a] (Ci/g)	Effective Half Life in Bone[b] (days)	Dose to Bone (rems/μg inhaled)
^{232}U	21.42×10^{0}	3.0×10^{2}	2.4×10^{3}
^{233}U	9.48×10^{-3}	3.0×10^{2}	2.1×10^{-1}
^{235}U	2.14×10^{-6}	3.0×10^{2}	4.3×10^{-5}
^{238}U	3.33×10^{-7}	3.0×10^{2}	6.3×10^{-6}
^{238}Pu	17.4×10^{0}	2.3×10^{4}	9.9×10^{4}
^{239}Pu	6.13×10^{-2}	7.2×10^{4}	4.0×10^{2}
^{240}Pu	2.27×10^{-1}	7.1×10^{4}	1.5×10^{3}

[a] 1 Ci = 3.70×10^{10} disintegrations per second.

[b] Time required for one-half of a given quantity to disintegrate, that is, for one-half of the nuclei to change form.

The fraction of the contaminant ^{232}U in denatured ^{233}U fuel would, of course, be much lower than the combined fraction of plutonium isotopes in Pu/U fuel. While no calculations of dose commitments have been performed specifically for denatured ^{233}U fuel, an upper limit can be estimated from calculations for HTGR fuel containing 93% ^{233}U in U. As recycle progresses, the ^{232}U content of HTGR fuel could increase to a maximum of perhaps 1000 ppm ^{232}U in U, in which case the dose commitment to the bone resulting from the inhalation of 10^{-12} g of the fuel would be about 4×10^{-3} mrem if inhaled immediately after the fuel has been processed. Because of the ingrowth of ^{232}U daughters, however, the potential dose commitment would increase for a period of approximately 10 years (to about 3×10^{-2} mrem/μg inhaled) after which it would decrease. Since denatured ^{233}U fuel is diluted with relatively nontoxic ^{238}U, it would contain proportionately less ^{232}U and would be somewhat less toxic than highly enriched HTGR fuel. By contrast, Pu/U fuel would be significantly more hazardous and LEU fuel would be significantly less hazardous.

APPENDIX C. EFFECT OF IMPROVED LWR DESIGNS AND ENRICHMENT TECHNOLOGY

While not considered in the system analyses discussed in Section 6, it is possible to improve LWR designs to greatly enhance their utilization of U_3O_8 per unit of energy produced — possibly as much as 30% on the once-through cycle. In order to estimate the effect that such improvements could have, a series of calculations was run to determine what the U_3O_8 requirements of LWRs would be at points in the future if their designs were gradually improved. At the same time the effect of a gradual decrease in the ^{235}U content of the uranium enrichment tails was considered.

In these calculations it was assumed that the LWR U_3O_8 utilization would be improved in sequential increments of 10%. Reactors starting up between 1981 and 1991 were assumed to need 90% of the U_3O_8 required by the standard LWR, those starting up between 1991 and 2001 would require 80%, and those starting up after 2001 would require 70%. It was also assumed that in those same decades the improvements would be retrofitted in all operating LWRs (with no downtime considered). The reduced tails schedule began with the standard ^{235}U fraction of 0.0020 in 1980 and gradually decreased to 0.0005 by 2010 and remained constant thereafter.

The results of the calculations, summarized in Table C.1, indicate that with improved LWR designs alone, the U_3O_8 consumption level would be reduced 25% by year 2029. If, in addition, the decreased tails enrichment were realized, the total U_3O_8 consumption could be reduced by 36%.* The U_3O_8 consumption of LWRs on once-through cycles would then be comparable to that of the standard LWR operating on the Pu/U recycle mode (or on denatured ^{233}U fuel).

Table C.1. Comparison of U_3O_8 Utilization of Standard and Improved LWRs Operating on Throwaway/Stowaway Option With and Without Improved Tails

	ST U_3O_8/GWe			
	Standard LWR Technology		Improved LWR Technology	
Year	Normal Tails	Improved Tails	Normal Tails	Improved Tails
1989	5236	4759	4649	4224
2009	5236	4508	4079	3560
2029	5236	4398	3923	3346

*Normal tails assume 0.2 w/o ^{235}U in ^{238}U; improved tails assumed 0.05 w/o ^{235}U in ^{238}U; 75% capacity factor.

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* This would require a large increase in SWU requirements, however.

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