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TO: ORNL-5388 Distribution

SUBJECT: "Interim Assessment of the Denatured ^{233}U Fuel
Cycle: Feasibility and Nonproliferation Char-
acteristics," Edited by L. S. Abbott, D. E.
Bartine, and T. J. Burns (December, 1978)

Please see that the changes indicated on the enclosed errata
are made in your copy of the subject report.

A handwritten signature in cursive script that reads "Lorraine S. Abbott".

Lorraine S. Abbott

ORNL-5388 ERRATA

- Page 3-7, line 1: Change "10% ^{233}U -enriched uranium" to "13% ^{233}U -enriched uranium"
- Page 3-14, paragraph 2, line 2: Change " ^{232}U decays to ^{232}Th " to " ^{232}U decays to ^{228}Th "
- Page 3-23, line 1: Change "spectrum from a ^{232}U sample" to "spectrum from a ^{233}U sample"
- Page 3-25, Table 3.3-2 footnote: Change description of fuel type D to "20 wt% ^{235}U in U plus Th"
- Pages 3-30 through 3-33: Substitute attached new (gummed-back) pages
- Page 4-15, Table 4.1-1: In Case D, change the numbers in the third and fourth columns to the following:
- | | |
|----------------------|----------------------|
| 348 ^{233}U | 351 ^{233}U |
| 685 ^{235}U | 281 ^{235}U |
| | 62 Pu^f |
- Page 4-33, Table 4.3-2: Under the first column change the sixth entry from "LEU, U+Pu recycle" to "Pu/U, Pu recycle"
- Page 4-55: Opposite Case 9 under column titled "Fissile Doubling Time" change "112.3" to "118.1"
- Page 4-56: In next to last line, change "Pu/Th" to "Pu/U"
- Page 5-9: In table title, change "20% $^{233}\text{U}/^{238}\text{U}$ -Th" to "12% $^{233}\text{U}/^{238}\text{U}$ -Th"
- Page 5-19: Under second column of table, change "215" to "185" (opposite LWR) and change "750" to "650" (opposite HTGR)
- Page 6-8: In column 10 of table, change footnote "b" to footnote "c"
- Pages 6-12 and 6-15: Under Options 1 and 3, in the phrase "Uranium is recycled in all reactors and plutonium is recycled in energy-center reactors" delete "is recycled in all reactors"
- Page 7-11, Table 7.2-1: Under second column opposite PBR, change "4289" to "4500"
- Page 7-13: Substitute attached new (gummed back) page
- Page 7-14: Substitute attached new (gummed back) page
- Pages 7-36 and 7-37: Add the following at the ends of the titles of Tables 7.4-3 and 7.4-4:
- "Fully Meeting Projected Nuclear Power Demand"
- Page 7-38: Substitute attached new (gummed back) page

The high alpha activity of uranium containing ^{232}U will present two problems:

1. In the UF_6 there will be a strong (α, n) reaction. A crude estimate of the neutron emission from a 16-kg UF_6 product cylinder containing 0.6 wt% ^{232}U is 5.7×10^7 neutrons/sec at 10 days decay, 2.5×10^8 at 30 days decay, and 8.7×10^8 at 90 days decay.
2. The ^{232}U will provide a strong heat source in the UF_6 and the metal products. A crude estimate of the heat generation rate from pure ^{232}U as a function of time after purification is: 0.03 W/g at 10 days, 0.13 W/g at 30 days, and 0.46 W/g at 90 days.

The degree to which these properties will affect weapon manufacture or delivery is unknown.

Alternative Enrichment Arrangements to Reduce ^{232}U Content in the Product. In considering the complications introduced to the final uranium metal product, i.e., the radiation level and heat generation resulting from ^{232}U , it is apparent that removal of the ^{232}U may be beneficial. Enrichment cascades can be designed to accomplish this. An effective arrangement would be to first strip most of the ^{238}U from the uranium isotopic mixture and then feed the low mass isotopes to a second cascade where the ^{232}U can be removed. This arrangement is illustrated in Fig. 3.3-2.

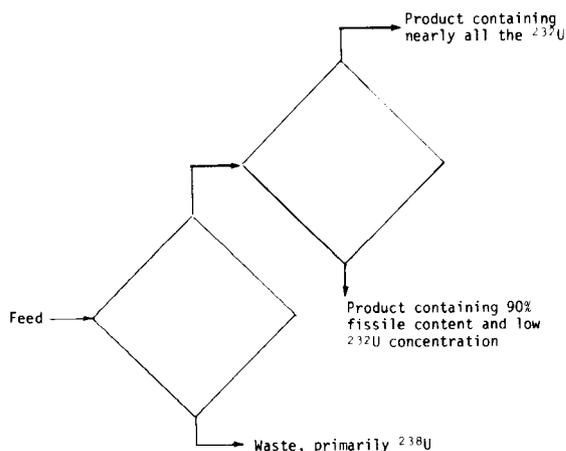


Fig. 3.3-2. Illustration of Enrichment Arrangement to Produce Low ^{232}U Content Uranium

Such enrichment cascades can be independent of the specific enriching device. Based on the discussion of the gas centrifuge process in Appendix A and at the beginning of this section, a small, low separative work capacity machine may be within the technical capabilities of a would-be diverter (see Appendix A).

Although no information exists on the separative work capacity of a Zippe machine in a cascade, a reasonable estimate of its separative capacity is about 0.3 kg SWU/yr when separating ^{235}U from ^{238}U .

To further specify the plant, it can be assumed that the diverter would like to:

1. Minimize the feed and waste stream flows in the first and second cascades consistent with limiting the number of centrifuges required.
2. Achieve a significant weapons-grade product flow rate. (A flow rate of 100 kg U/yr having a fissile content of 90% $^{233}\text{U} + ^{235}\text{U}$ was chosen.)
3. Reduce the ^{232}U content in the metal product so that contact manufacture can be achieved without serious radiation hazard.

A 90% fissile content was arbitrarily chosen for this analysis. It is recognized that other lower enrichments, down to and including the feed enrichments of 12% ^{233}U , will allow the construction of a much larger, cumbersome, but probably still workable device. There is, of course, a distinct tradeoff in the weapon enrichment choice between centrifuge requirements and uranium feed requirements. A further consideration is the role envisioned for the weapon.

Based on these assumptions and considering the fuel types listed in Table 3.3-2, a series of enrichment cascades, flows and selected isotopic parameters are presented in Table 3.3-7. The basic criterion chosen for the final uranium product was that the ^{232}U concentration was about 1 ppm ^{232}U in total uranium. At this level the gamma emission rate from the final metal product is sufficiently low that most fabrication and subsequent handling operations can be carried out in unshielded facilities using contact methods. As higher ^{232}U product concentrations are permitted, the enrichment requirements are reduced. (Note: The relationship between the number of centrifuges at a higher capacity than the 0.3 kg SWU/yr machine on which Table 3.3-7 is based can be determined with reasonable accuracy by a simple ratio of the centrifuge capacities. Thus, 1000 centrifuges of 0.3 kg SWU/yr may be replaced with $1000 \times \frac{0.3}{5.0} = 60$ centrifuges of 5 kg SWU/yr capacity.)

The top of the second enrichment cascade will be very radioactive. But it will be only slightly more radioactive than a single cascade used to enrich the uranium feed without stripping the ^{232}U .

The data presented in Table 3.3-7 have been analyzed with respect to the number of centrifuges needed and the amount of uranium feed required to produce enough 90% fissile uranium for one weapon per year, the assumed weapon masses being consistent with those reported in the open literature (i.e., 10 kg of 90 wt% ^{233}U , 25 kg of 90 wt% ^{235}U , or a linear combination of the two). Based on this analysis, the following conclusions can be drawn with respect to desirability of fuels for diversion:

1. If a highly decontaminated 90 wt% fissile product is needed (i.e., ^{232}U at approximately 1 ppm in the weapon), the denatured fuels will require approximately 10 to 20% of the number of centrifuges required for natural uranium, approximately 50 to 90% of the number of centrifuges as 3.2 wt% ^{235}U fuel and 3 to 5 times the number for 20 wt% ^{235}U fuel.
2. At higher allowable ^{232}U levels in the product, the centrifuge requirements for the denatured uranium fuels reduce to, and at the highest ^{232}U levels may be significantly lower than, those for 20 wt% ^{235}U fuel. The ^{232}U levels at which the selected fuel types require the same separative capacity for a weapon's worth of 90% fissile product as 20 wt% ^{235}U fuel are: for type B, ~ 570 ppm; for type C, ~ 3000 ppm; for type E, ~ 400 ppm; and for type G, ~ 380 ppm. Fuel types H and F require more separative

capacity even with no ^{232}U removal.

If no ^{232}U decontamination is acceptable in the final product the denatured fuels will require between 3% and 20% of the number of centrifuges as 3.2 wt% ^{235}U fuel and between 20% and 120% of the number needed for 20 wt% ^{235}U fuel. Thus, the ranking of the denatured uranium-thorium fuels with respect to ^{235}U fuels is obscured both by the type of feed to which the proliferator has access and by the ^{232}U content that he will accept in his weapon design. This in turn is a function of his requirements for the weapon.

3. There are two apparent classes of fuels that result from consideration of the number of standard Westinghouse PWR fuel assemblies that must be diverted to produce a given number of 90% fissile weapons. For example, for one weapon per year, fuel types B and C require diversion of one assembly per year. The remaining denatured cycles, the 20 wt% ^{235}U -thorium cycle and the 3.2 wt% ^{235}U would require two to three assemblies per weapon per year. While these ratios are not as exact for greater weapon production rates, they are still suggestive of a classification.
4. For all cases considered and at equal fissile recovery fractions, the denatured uranium-thorium fuels require less enrichment (i.e., fewer numbers of centrifuges) and less feed to produce a nominal 90% fissile weapon mass than does LWR fuel of 3.2 wt% ^{235}U .

Reliability of Centrifuge Enrichment Plants. As a final item, the average centrifuge failure rate and its impact on the maintainability and production rate of a centrifuge enrichment plant must be considered. Information on the reliability and operating life of centrifuges is scarce. The URENCO-CENTEC organization has over the years made claims of very long average operating life and correspondingly low failure rates. Typical examples of these claims can be found in some of their sales brochures.¹³ These claim an average 10-year operating life and a failure rate of less than 0.5%/year. It is not clear how much periodic maintenance (e.g., oil changes and bearing inspection) is required to achieve these low failure rates.

If these claims are accepted as a goal of a long-term development project, then it can be assumed that in the early part of the development somewhat higher failure rates would occur, perhaps greater by a factor of 10. This factor might be further justified in a highly radioactive plant since periodic maintenance would not be practical.

The effect of centrifuge failures on the production rate in a radioactive plant has not been determined; however, some qualitative statements can be made. All centrifuge plants must be designed so that failed units or groups of units can be immediately isolated from the rest of the plant. It should also be possible, for a specific cascade layout, an assumed failure rate, and a specified plant operating life, to provide

Table 3.3-7. Summary of Results of Centrifuge Enrichment Survey of Potential Fuel Mixtures

Fuel Type ^a	²³² U Content		Overall Fissile Recovery Fraction		For 100 kg U/yr of 90% Fissile Product		Number of Fuel Assemblies to Supply Needed Feed
	In Feed (ppm)	In Product (ppm)	²³³ U	²³⁵ U	Annual feed (kg U/yr)	No. of Zippes (each at 0.3 kg SWU/yr) ^b	
A	0	0	--	0.94	2,993	29,220	6.7
	0	0	--	0.9	3,124	24,984	7.0
	0	0	--	0.8	3,516	19,924	7.9
	0	0	--	0.7	4,016	17,220	9.0
B	502	1	0.913	0	825	25,938	7.0
	502	1	0.800	0	925	25,150	7.9
	502	1	0.722	0	1,050	24,762	9.0
	502	1	0.626	0	1,213	24,600	10.4
	502	50	0.722	0	1,050	13,938	9.0
	502	100	0.722	0	1,050	12,025	9.0
	502	500	0.722	0	1,038	7,600	9.0
	502	4,403	0.706	0	1,075	1,863	9.2
	502	4,023	0.906	0	838	2,562	7.2
C	656.4	1	0.813	0	963	32,063	7.8
	656.4	1	0.720	0	1,088	30,575	8.8
	656.4	1	0.626	0	1,250	29,838	10.1
	656.4	50	0.720	0	1,088	19,512	8.8
	656.4	100	0.720	0	1,088	17,538	8.8
	656.4	500	0.728	0	1,075	12,988	8.8
	656.4	6,723	0.704	0	1,113	5,025	9.0
	656.4	5,661	0.907	0	863	7,462	7.0
	D	0	0	--	0.9615	468	4,991
0		0	--	0.9	500	3,976	5.1
0		0	--	0.8	564	3,288	5.8
0		0	--	0.7	644	2,916	6.6
E		123.6	1	0.647	0.553	870	10,980
	123.6	1	0.598	0.422	1,073	11,150	10.4
	123.6	1	0.518	0.338	1,305	11,284	12.6
	123.6	50	0.612	0.421	1,068	6,198	10.3
	123.6	100	0.625	0.419	1,063	5,356	10.3
	123.6	500	0.687	0.426	1,016	3,452	9.8
	123.6	933	0.878	0.558	783	2,989	7.6
	123.6	1,146	0.709	0.340	1,157	2,438	11.2
	F	244.5	1	0.858	0.008	1,843	38,771
244.5		1	0.709	0.011	2,082	35,141	17.9
244.5		1	0.616	0.014	2,368	33,057	20.4
244.5		50	0.709	0.011	2,082	24,329	17.9
244.5		100	0.713	0.011	2,071	22,424	17.8
244.5		500	0.713	0.011	2,071	18,000	17.8
244.5		4,026	0.899	0.008	1,663	14,735	14.3
244.5		4,704	0.702	0.016	2,080	9,437	17.9
244.5		0.33	0	0.927	856	26,572	7.4
G	113.4	1	0.706	0.578	818	10,340	8.1
	113.4	1	0.583	0.426	1,075	10,407	10.7
	113.4	1	0.499	0.345	1,305	10,467	13.0
	113.4	50	0.588	0.430	1,065	5,779	10.7
	113.4	100	0.602	0.428	1,061	5,005	10.6
	113.4	500	0.689	0.428	1,016	3,290	10.1
	113.4	850	0.894	0.555	783	2,984	7.8
	113.4	1,048	0.700	0.345	1,158	2,435	11.5
	H	233.1	1	0.798	0.012	1,941	39,459
233.1		1	0.705	0.017	2,159	35,409	18.8
233.1		1	0.609	0.018	2,467	33,289	21.5
233.1		50	0.705	0.017	2,159	24,659	18.8
233.1		100	0.705	0.017	2,159	22,739	18.8
233.1		500	0.708	0.017	2,148	18,364	18.7
233.1		4,020	0.903	0.010	1,726	15,202	15.0
233.1		4,687	0.697	0.017	2,182	9,784	19.0
233.1		0.85	0	0.656	1,168	29,732	10.2
Natural U		0	0	--	0.72	17,575	77,918
	0	0	--	0.9	14,248	110,688	NA
	0	0	--	0.8	15,824	87,420	NA
	0	0	--	0.7	18,084	74,984	NA

^aSee Table 3.3-2 for description of fuel types.

^bThe number of machines at other capacities can be estimated by ratio of the capacities; that is, for 5-kg-SWU/yr machines, the number can be reduced by a factor of 0.315 = 0.06.

^cNA = not applicable.

The isotopic composition of the spent fuel inventories is also of interest from a proliferation standpoint. 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. The use of 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.

Recycle Systems

If recycling of the fissile material in the thermal reactors is permitted, then ^{233}U (and plutonium) produced in the MEU(235)/Th is recoverable on a schedule dictated by the production rate of the system. Table 7.2-3 gives estimates of the net lifetime consumption and production of various fissile materials for the MEU(235)/Th fuel cycle under the assumption that the capability for uranium recycle is available. (The ^{235}U consumption tabulated does *not* reflect the ^{235}U lost to the enrichment tailings.) For comparison purposes, the MEU(233)/Th fuel cycle estimates are also provided. The most striking aspect of Table 7.2-3 is the apparent 30% reduction of fissile consumption achieved with the ^{233}U system, indicating the higher value of ^{233}U as a thermal reactor fuel. In fact, the true extent of this effect is masked somewhat since a large fraction of the recycled fuel for the ^{235}U makeup case is in fact ^{233}U .

Table 7.2-3. Estimated 30-Year Fissile Consumption and Production for MEU/Th Cycles with Uranium Recycle^a

Reactor	MT/GWe			
	With ^{235}U Loading and Makeup		With ^{233}U Loading and Makeup	
	^{235}U Consumption	Fissile Pu Production	^{233}U Consumption	Fissile Pu Production
PWR	14.0	1.9	10.0	1.9
HWR	6.4	0.9	3.8	1.0
HTGR	18.8	0.8	9.0	0.8
SSCR	13.1	1.9	8.0	2.2

^aAt 75% capacity factor.

As has been stated earlier, the consideration of an MEU/Th cycle that utilizes ^{233}U makeup presumes the existence of a source of the requisite ^{233}U . Although the ^{233}U in the spent fuel elements would be recovered, the amount would be inadequate to maintain the system and an exogenous source must be developed. One means for generating ^{233}U is by using a Pu/Th-oxide-fueled thermal reactor. Table 7.2-4 summarizes some pertinent results for the various thermal reactors operating on the Pu/Th cycle. It should be noted that the HTGR case given in Table 7.2-4 is for a case in which the full core is refueled every 5 yr and is not optimized for ^{233}U production. Thus, much of the ^{233}U bred during this period is consumed in providing power, and the transmutation efficiency (tons of plutonium "transmuted" into tons of ^{233}U) is significantly reduced relative to the PWR and SSCR. The transmutation efficiency of 0.40 for the PWR and SSCR is also rather poor, however, compared to the 1.20 value for a Pu/Th-fueled FBR (see Section 4.5). Production of ^{233}U via plutonium-consuming transmuters is more suited to fast reactors. On the other hand, it is recognized that Pu/Th-fueled thermal reactors could provide an interim source of ^{233}U .

Table 7.2-4. Net 30-Year Fissile Consumption and Production for Pu/Th Cycles^a

Reactor	MT/GWe		Transmutation Efficiency
	Fissile Pu Consumption	^{233}U Output	
PWR	21.7	8.2	0.38
HWR ^b	19.9	11.8	0.59
HTGR	15.3	2.8	0.18
SSCR	24.5	8.2	0.33

^aAt 75% capacity factor, using equilibrium cycle values.

^bFrom data in Table 6.1-3.

7.2.2. Fast Reactors

In this study fast reactors have been 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 LMFBRs used as the model, the denatured FBRs were analyzed for a range of $^{233}\text{U}/\text{U}$ enrichments to parameterize the impact of the fuel on the reactor performance (see Section 4.5), and the transmuter FBRs were analyzed both for a Pu/ ^{238}U core driving a ThO₂ blanket and for a Pu/Th system in which the thorium was included in both the core and the blanket.

The specified $^{233}\text{U}/\text{U}$ enrichment is a crucial parameter for the denatured fast reactors. Increasing the allowable enrichment permits more thorium to be used in the fuel material and hence allows the reactors to be more self-sufficient (i.e., reduces the

are generally less than those for thermal systems in the Pu-U recycle mode (Option 2). Discarding Pu from the recycle of denatured thermal systems (Option 4) reduces the efficiency of the denatured cycle.

The nuclear power systems that include fast breeders (Options 3, 6, 7, and 8) have cumulative U_3O_8 requirements through year 2049 within the range of 2.71 to 4.41 million ST U_3O_8 in the case of the intermediate-cost U_3O_8 supply and within 2.6 to 3.2 million ST U_3O_8 in the case of the high-cost supply. The maximum U_3O_8 consumption varies from 66,000 to 93,000 ST/yr for the intermediate-cost supply and from 52,000 to 68,000 ST/yr for the high-cost supply. The breeder-containing options are able to adjust the reactor mix effectively to reduce U_3O_8 consumption in the event U_3O_8 costs are high. The larger the fraction of breeders in the reactor mix, the lower the U_3O_8 requirements.

It should be noted that the U_3O_8 requirements for the systems containing breeders with Pu/U cores and Th blankets (Options 6 and 7) are similar to the U_3O_8 requirements for the system containing the classical Pu/U breeder (Option 3). The systems containing breeders with Pu/Th cores and Th blankets require somewhat more U_3O_8 on an integrated basis.

The U_3O_8 requirements presented in Table 7.4-4 qualitatively support the ranking of cycles in the cost-constrained runs. Specifically, the power systems operating on once-through cycles require 5.6 to 7.1 million ST U_3O_8 to satisfy the demand for nuclear power through 2050, the thermal-recycle systems require 3.3 to 5.4 million ST U_3O_8 , and the breeder-containing systems require 2.6 to 4.4 million ST U_3O_8 . The systems including denatured ^{235}U reactors require approximately the same cumulative amount of U_3O_8 as their Pu/U counterparts. The results presented in Table 7.4-5 also support these statements: the required production rates are highest for the once-through systems; they are reduced somewhat for the thermal recycle cases; and they are lowest for the breeder-containing scenarios.

7.4.5. Systems Employing Improved LWRs and Enrichment Technology

While not considered in the analysis summarized above, it is possible to optimize LWR designs to greatly enhance their utilization of U_3O_8 per unit energy produced. These optimized designs may result in reduced U_3O_8 requirements of up to 30% relative to more conventional LWR designs. The 30% improvement in LWR U_3O_8 requirements assumes no spent fuel reprocessing, the improvements being the result of increased discharge exposure fuels and/or reconfigured reactor cores.

The effect of developing these LWR cores optimized for throwaway/stowaway operation was examined by assuming that the U_3O_8 utilization would be improved in sequential increments of 10%. Thus it was assumed that reactors starting up between 1981 and 1991 would have U_3O_8 requirements equal to 90% of the standard LWR. It was also assumed that this improvement would be retrofitted into existing reactors.[†] Similarly, reactors starting up

[†]Neither the down time required for retrofitting nor the associated costs were addressed in this analysis.