SSA-106

April, 1978

STUDIES OF ALTERNATIVE NUCLEAR TECHNOLOGIES

Prepared For

U.S. ARMS CONTROL AND DISARMAMENT AGENCY

Prepared By

Southern Science Applications, Inc.
Division of Black & Veatch
P.O. Box 10
Dunedin, Florida 33528

NOTICE

This report was prepared by Southern Science Applications, Inc., (SSA) a division of Black & Veatch, as an account of work sponsored by the United States Government. Neither the United States, the Arms Control and Disarmament Agency, SSA, Black & Veatch, nor any of their employees, contractors, subcontractors or their employees, nor any person acting on their behalf: (a) make any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or (b) assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this report.

SSA-106
Contract No. AC7NC114

April, 1978

STUDIES OF ALTERNATIVE NUCLEAR TECHNOLOGIES

Prepared For

U.S. ARMS CONTROL AND DISARMAMENT AGENCY

Prepared By

S. E. Turner, Project Manager
M. K. Gurley
K. D. Kirby
W. Mitchell, III
K. E. Roach

Southern Science Applications, Inc.
Division of Black & Veatch
P. O. Box 10
Dunedin, Florida 33528

TABLE OF CONTENTS

Section and Title		Page No.
1.0	INTRODUCTION	1
2.0	METHODS OF CALCULATION	2
3.0	FUEL UTILIZATION IN PRESSURIZED WATER REACTORS (PWRs)	4
4.0	ALTERNATE FUEL CYCLES	12
5.0	FUEL UTILIZATION IN HIGH TEMPERATURE GRAPHITE REACTORS (HTGRs)	15
6.0	FUEL UTILIZATION IN CANDU TYPE REACTORS	20
8.0	ISSUES IN THE UTILIZATION OF DENATURED FUEL IN LWRs	25
8.1	General	25
8.2	Safety/Licensing Issues	26
8.3	Operation/Performance Issues	27
8.4	Economic Issues	28
8.5	Subjective/Political Issues	29
8.6	Resolution and Initial Steps	30

TABLE OF CONTENTS (Cont'd)

Section and Title		Page No.
9.0	HEAVY WATER REACTOR DESIGN	31
9.1	Introduction	31
9.2	Basic Considerations	31
9.3	Nuclear Steam Supply System Conceptual Design	33

LIST OF TABLES

Table No.		Page No.
1	Fuel utilization for LWRs, 3-cycle loading at design water-to-fuel ratio, 75% plant factor.	5
2	Fuel utilization, reference design with extended fuel burnup.	6
3	Fuel utilization for LWRs, on-line refueling at reference design water-to-fuel ratio, 75% plant factor.	7
4	Fuel utilization in CANDU type reactors, on-line refueling at 75% plant factor.	8
5	Fuel utilization in LWRs, 3-cycle loading, water-to-fuel ratio of 1.5, 75% plant factor.	9
6	Fuel utilization in LWRs, on-line refueling, water-to-fuel ratio of 1.5.	10
7	Fuel utilization characteristics for LWRs under various fuel cycle options.	13
8	Fuel utilization in PWRs with the denatured fuel cycle.	14
9	Fuel utilization in high temperature graphite reactors.	19
10	Fuel utilization characteristics for CANDU reactors under various fuel cycle options.	21
11	Fuel utilization in natural uranium, CANDU type reactors with U-233 enrichment	22
12	Fuel utilization in pressurized water reactors with the tandem fuel cycle.	24

LIST OF ILLUSTRATIONS

Figure No.

1 U-235 enrichment requirements as a function of design fuel burnup.

2 Thirty year average resource requirements for various design fuel burnups.

3 Thirty year average plutonium production for various design fuel burnups.

4 Initial conversion ratio for HTGR-GASSAR fuel (weight % U-235 = 4.1) as a function of thorium content.

5 K_∞ as a function of fuel exposure for HTGR-GASSAR irradiation of mixed uranium and thorium oxide fuels.

6 Specific mass of fissile plutonium as a function of fuel exposure for HTGR-GASSAR irradiation of mixed uranium and thorium oxide fuels.

7 Conversion ratio for HTGR-GASSAR irradiation of mixed uranium and thorium oxide fuels for reference cycle reactivity.

8 K_∞ as a function of fuel exposure for HTGR-GASSAR irradiation of mixed U-233, natural uranium and thorium oxide fuel.

9 Specific Mass of fissile material as a function of fuel exposure for HTGR-GASSAR irradiation of mixed U-233, natural uranium and thorium oxide fuel.

10 K_∞ as a function of fuel exposure for HTGR-GASSAR irradiation of mixed U-233 enriched uranium and thorium oxide fuel.

11 Specific Mass of fissile material as a function of fuel exposure for HTGR-GASSAR irradiation of mixed U-233, enriched uranium and thorium oxide fuel for recycle use.

LIST OF ILLUSTRATIONS (Cont'd)

Figure No.	
12	K_∞ as a function of fuel exposure for HTGR-GASSAR irradiation of mixed uranium and thorium oxide fuels for a once-through fuel cycle.
13	Heavy water reactor vertical cross-section.
14	Heavy water reactor flow diagram.

1.0 INTRODUCTION

This report is a summary of tasks performed for the U.S. Arms Control and Disarmament Agency under Contract AC7NC114. The work is directly related to the Agency effort to examine potential alternative fuel cycles that might enhance uranium resource utilization, minimize plutonium production, and reduce the weapons proliferation risk from spent fuel reprocessing or early introduction of fast breeder reactors. Reported herein are summaries of various inter-related task assignments, including

- Fuel utilization in current light water reactors operating with the uranium fuel cycle;
- alternate fuel cycles, including the use of denatured fuel in LWRs and of the spectral shift concept for reactivity control;
- fuel utilization in high temperature graphite moderated reactors using the denatured fuel cycle;
- fuel utilization in heavy water reactors (CANDU type), including the use of enriched fuel, denatured fuel, and recycle of plutonium and U-233;
- the tandem fuel cycle (recovery of spent fuel and further irradiation in a CANDU type reactor);
- issues in the utilization of denatured fuel in LWRs; and
- preliminary conceptual evaluation of a heavy water moderated reactor suitable for use in the United States.

2.0 METHODS OF CALCULATION

The basic analytical tool is a zero-dimensional, multi-group, point-depletion cell calculation, employing microscopic cross-sections compiled from the ENDF/B evaluated cross-section set. This program has previously shown excellent correlation with results of a large number of critical experiments and with experimental fuel burnup data in LWRs. Spatial calculations were not made; instead, a simple average of the k_∞ at the end, at one-third, and at two-thirds of the fuel burnup was taken as the core-average k_∞. For the reference PWR, this yields an end-of-cycle k_∞ of 1.064. For other calculations, the attainable burnup or the enrichment required to achieve a given burnup was determined to give the same end-of-cycle reactivity. The approximation used here neglects spatial effects, but should give reasonable values of the equilibrium fuel burnup or enrichment requirements. All calculations were based on the geometry and mechanical design of the Indian Point-2 reactor core.

The initial conversion ratio (ICR) is a simple indicator of the potential rate of production of fissile material in an operating reactor. Although the integrated value of conversion ratio over the fuel lifetime will differ somewhat from the initial value, the difference is not large. The conversion ratio, however, must be normalized to the critical reactor core to have any real significance. Reaction rates calculated by a point-depletion program can be used to estimate the normalized ICR, assuming that thermal neutron absorption is added in the amount needed to reduce the reactor to a just-critical state. In effect, this reduces the thermal neutron group flux to achieve criticality. The fast group contribution to reactivity, k_1, is provided by the point-depletion program, and the thermal contribution is then $1-k_1$ for a critical system. Relative group fluxes are given by $k_1/\nu\sum_f^1$ in the fast group (ϕ_1) and by $(1-k_1)/\nu\sum_f^2$ in the thermal group (ϕ_2). With these relative flux values, the initial conversion ratio is the rate of neutron capture in fissile material (U-238 and Th-232) divided by the rate of neutron absorption in the fissile material (U-233 and U-235) initially present in the fuel, as follows:

$$ICR = \frac{\sum_{C,1}^{28}\phi_1 + \sum_{C,2}^{28}\phi_2 + \sum_{C,1}^{02}\phi_1 + \sum_{C,2}^{02}\phi_2}{\sum_{A,1}^{23}\phi_1 + \sum_{A,2}^{23}\phi_2 + \sum_{A,1}^{25}\phi_1 + \sum_{A,2}^{25}\phi_2},$$

where Σ is the macroscopic capture or absorption cross-section in the fast and thermal groups for U-238 ($\sum_C^{28}$), Thorium-232 ($\sum_C^{02}$), U-233 ($\sum_A^{23}$), and U-235 ($\sum_A^{25}$). Although the equation is an approximation, an independent check calculation for an exactly-critical reactor confirmed the validity of this normalization process. The integrated conversion ratio, defined as the time integral rate of fissile material production divided by the time integral rate of fissile material destruction, is computed by the point-depletion program and printed out at each time step.

In computing resource utilization, a 75% plant factor was assumed for all cases, and the annual requirements were normalized to a 1000 Mw(e) plant. The initial loading requirement was based upon a simple mixture of one-third of the core at the equilibrium loading, one-third at the U-235 loading corresponding to one-third of the final burnup, and one-third at the U-235 loading corresponding to two-thirds of the final burnup. This approximation, which is equal to twice the equilibrium annual loading plus the equilibrium discharge, neglects the perturbation due to the actual method used in the approach to equilibrium. The 30-year resource requirement is then 29 times the annual requirement plus the initial loading. Total plutonium production is 30 times the annual production rate. It should be noted again that the values obtained are those for a 1000 Mw(e) plant operating for 30 years at an average plant factor of 75%.

Throughout this report, fuel burnup in Mwd/mt refers to the loading of fuel in terms of the initial metric tons of heavy metal (ImtHM).

3.0 FUEL UTILIZATION IN PRESSURIZED WATER REACTORS (PWRs)

A number of calculations were performed to evaluate the long-range fuel utilization characteristics and plutonium production in PWRs, using the throwaway fuel cycle (sometimes called the stowaway or once-through cycle), as a function of enrichment, fuel burnup, and water-to-fuel ratio. Results of these calculations are summarized in Tables 1 through 5. For reference, similar results for the CANDU reactor, for an idealized on-line refueling scheme, are shown in Table 6.

In Table 1, different fuel burnups in the reference PWR are achieved by adjusting the U-235 enrichment, assuming the same 3-cycle loading scheme used in modern large plants. Examination of Table 1 reveals that the 30-year requirement for uranium ore decreases as the enrichment (and hence burnup) increases. However, beyond an enrichment of 3.2% U-235, the reduction of U_3O_8 requirements is not significant. Net annual plutonium production continues to decrease with increasing fuel burnup.

Table 2 shows the effect of increasing fuel burnup without changing enrichment. Such an increase in fuel burnup could only be accomplished by reducing the reactivity margin at the end-of-cycle, so as to provide the additional reactivity needed. Table 2 reveals that substantial improvement in resource utilization (and reduced plutonium production) are possible if the present burnup of PWR fuel (approximately 33,000 Mwd/mtU) could be extended (i.e., by more frequent refueling, allowing a PWR to approach on-line refueling and reducing the required operating reactivity margin). An upper limit is the hypothetical idealized on-line refueling where no excess reactivity for operation exists other than inherent neutron losses through leakage. Table 3 indicates the optimum in resource utilization occurs for an equilibrium enrichment of about 3.2% U-235 (corresponding to current designs). It may be noted that the discharge U-235 enrichment at the optimum is approximately equal to the usual tails enrichment (0.2%) in a diffusion plant, which eliminates any incentive for uranium recycle.

For comparison, in the throwaway fuel cycle, fuel utilization in CANDU reactors for three different enrichments, assuming a hypothetical idealized on-line refueling scheme, are summarized in Table 4. For the reference natural uranium system, the idealized burnup is 10,200 Mwd/mtU in contrast to 7500 to 8500 Mwd/mtU actually achieved in practice.

Table 1 FUEL UTILIZATION FOR LWRs, 3-CYCLE LOADING AT DESIGN WATER-TO-FUEL RATIO, 75% PLANT FACTOR

	E = 1.8%	E = 2.4%	E = 2.8%	E = 3.2%	E = 3.6%	E = 4.0%	E = 4.5%	
Discharge Burnup, Mwd/kg	11.5	21.6	27.6	33.0	39.0	44	50.4	
Equilibrium Fissile Enrichment, % HM	1.8	2.4	2.8	3.2	3.6	4.0	4.5	
Cycle time, yrs	1.268	2.382	3.044	3.64	4.301	4.853	5.559	
Initial conv. ratio	.647	.578	.546	.521	.501	.484	.465	
Integrated conv. ratio	.622	.596	.581	.567	.558	.546	.535	
Initial Loading * Requirements, kg/GWe-yr								
U-235	3364	2243	2005	1892	1773	1736	1688	
ST U_3O_8	760	523	473	451	426	420	409	
Equilibrium Loading, * kg/GWe-yr								
U-235	1354	961	877	839	798	786	772	
ST U_3O_8	306	224	207	200	192	190	187	
Equilibrium Discharge, * kg/GWe-yr, average								
U-235	656	321	251	214	177	164	144	
Fissile Pu	295	210	182	164	149	139	128	
U-235 discharge enrichment	.889	.828	.835	.858	.845	.845	.9017	
Annual Net Requirements, * kg/GWe-yr, average								
U-235 consumed	698	640	626	625	621	622	628	
U-235 fissioned	572	521	507	504	499	499	501	
Pu fissioned in situ	266	294	300	300	302	301	300	
Enrichment requirements, SWU	1.35×10^5	1.21×10^5	1.21×10^5	1.24×10^5	1.25×10^5	1.29×10^5	1.32×10^5	
30-Year Requirements *								
ST U_3O_8	9634	7019	6476	6251	5994	5930	5832	
kg Fissile Pu	8850	6300	5460	4920	4470	4170	3840	

* 1000 MWe plant operating at a 75% plant factor.

Table 2 FUEL UTILIZATION, REFERENCE DESIGN WITH EXTENDED FUEL BURNUP

	Ref 33,000 Mwd/mtU	40,000 Mwd/mtU	50,000 Mwd/mtU	60,000 Mwd/mtU
Discharge Burnup, Mwd/kg	33	40	50	60
Equilibrium Fissile Enrichment, % HM	3.2	3.2	3.2	3.2
Cycle time, yrs.	3.64	4.412	5.514	6.617
Initial conv. ratio	.521	.521	.521	.521
Integrated conv. ratio	.567	.599	.644	.685
Initial Loading Requirements, kg/GWe-yr				
U-235	1892	1793	1679	1605
ST U_3O_8	451	428	400	383
Equilibrium Loading, kg/GWe-yr				
U-235	839	692	554	461
ST U_3O_8	200	165	132	110
Equilibrium Discharge, kg/GWe-yr, average				
U-235	214	121	54.5	24
Fissile Pu	164	142	116	96
U-235 discharge enrichment	.86	.59	.34	.18
Annual Net Requirements, kg/GWe-yr, average				
U-235 consumed	625	571	500	437
U-235 fissioned	504	461	403	353
Pu fissioned in situ	300	337	389	434
Enrichment requirements, SWU	1.24×10^5	1.0×10^5	$.82 \times 10^5$	$.68 \times 10^5$
30-Year Requirements				
ST U_3O_8	6251	5213	4228	3573
kg Fissile Pu	4920	4260	3480	2880

Table 3 FUEL UTILIZATION FOR LWRs, ON-LINE REFUELING AT REFERENCE DESIGN WATER-TO-FUEL RATIO, 75% PLANT FACTOR

	E = 1.2 %	E = 1.8 %	E = 2.4 %	E = 2.8 %	E = 3.2 %	E = 3.6 %	E = 4.0 %	E = 4.5 %
Discharge Burnup, Mwd/kg	6.00	26.0	42	51.5	60.3	69.0	77.0	86.5
Equilibrium Fissile Enrichment, % HM	1.2	1.8	2.4	2.8	3.2	3.6	4.0	4.5
Cycle time, yrs	.6618	2.868	4.632	5.681	6.65	7.609	8.493	9.542
Initial conv. ratio	.777	.647	.578	.546	.521	.501	.484	.465
Integrated conv. ratio	.718	.712	.702	.695	.686	.680	.672	.663
Initial Loading Requirements, kg/GWe-yr								
U-235	904	1020	1253	1426	1606	1784	1970	2204
ST U_3O_8	192	230	292	337	381	427	478	534
Equilibrium Loading, kg/GWe-yr								
U-235	1729	599	494	470	459	451	449	450
ST U_3O_8	367	135	115	111	109	108	109	109
Equilibrium Discharge, kg/GWe-yr, average								
U-235	1003	112	47	32	24	18	15	12
Fissile Pu	419	178	126	108	96	80	79	72
U-235 discharge enrichment	.704	.350	.240	.202	.18	.16	.15	.14
Annual Net Requirements, kg/GWe-yr, average								
U-235 consumed	726	487	447	438	435	433	434	438
U-235 fissioned	600	399	364	355	352	348	348	349
Pu fissioned in situ	269	411	431	435	436	438	438	434
Enrichment requirements, SWU	1.0×10^5	6.0×10^4	6.2×10^4	6.498×10^4	6.81×10^4	7.06×10^4	7.34×10^4	7.69×10^4
30-Year Requirements								
ST U_3O_8	10,835	4145	3627	3556	3542	3559	3639	3695
kg Fissile Pu	12,570	5340	3780	3240	2880	2400	2370	2160

Table 4 FUEL UTILIZATION IN CANDU TYPE REACTORS, ON-LINE REFUELING AT 75% PLANT FACTOR

	Ref.	E = 1.2%	E = 1.5%	
Discharge Burnup, Mwd/kg	10.2	25.7	35	
Equilibrium Fissile Enrichment, % HM	Nat U (0.711%)	1.2	1.5	
Cycle time, yrs.	2.108	5.311	7.234	
Initial conv. ratio	.767	.536	.465	
Integrated conv. ratio	.800	.754	.716	
Initial Loading Requirements, kg/GWe-yr				
U-235	757	1104	1353	
ST U_3O_8	138	234	299	
Equilibrium Loading, kg/GWe-yr				
U-235	611	405	371	
ST U_3O_8	112	86	82	
Equilibrium Discharge, kg/GWe-yr, average				
U-235	107	10.6	3.	
Fissile Pu	274	123	92	
U-235 discharge enrichment	.13	.032	.013	
Annual Net Requirements, kg/GWe-yr, average				
U-235 consumed	504	394	368	
U-235 fissioned	424	330	308	
Pu fissioned in situ	457	522	540	
Enrichment requirements, SWU	0	2.35×10^4	3.03×10^4	
30-Year Requirements				
ST U_3O_8	3386	2728	2677	
kg Fissile Pu	8220	3690	2760	

Table 5 FUEL UTILIZATION IN LWRs, 3-CYCLE LOADING, WATER-TO-FUEL RATIO OF 1.5, 75% PLANT FACTOR

	E = 1.8%	E = 2.8%	E = 3.6%	E = 4.0%
Discharge Burnup, Mwd/kg	7.6	21.7	31.4	35.9
Equilibrium Fissile Enrichment, % HM	1.8	2.8	3.6	4.0
Cycle time, yrs.	.9864	2.817	4.077	4.662
Initial conv. ratio	.739	.630	.580	.560
Integrated conv. ratio	.675	.627	.600	.589
Initial Loading Requirements, kg/GWe-yr				
U-235	5375	2681	2314	2228
ST U_3O_8	1207	634	556	539
Equilibrium Loading, kg/GWe-yr				
U-235	2049	1116	991	963
ST U_3O_8	460	264	238	233
Equilibrium Discharge, kg/GWe-yr, average				
U-235	1277	449	332	302
Fissile Pu	420	263	224	211
U-235 discharge enrichment	1.14	1.17	1.27	1.33
Annual Net Requirements, kg/GWe-yr, average				
U-235 consumed	772	667	659	661
U-235 fissioned	626	533	521	520
Pu fissioned in situ	225	291	295	293
Enrichment requirements, SWU	2.05×10^5	1.54×10^5	1.55×10^6	1.58×10^6
30-Year Requirements				
ST U_3O_8	14,547	8290	7458	7290
kg Fissile Pu	12,600	7890	6720	6330

Table 6 FUEL UTILIZATION IN LWRs, ON-LINE REFUELING, WATER-TO-FUEL RATIO OF 1.5

	E = 1.8%	E = 2.8%	E = 3.6%	E = 4.0%
Discharge Burnup, Mwd/kg	21.2	45.0	61.2	68.8
Equilibrium Fissile Enrichment, % HM	1.8	2.8	3.6	4.0
Cycle time, yrs.	2.752	5.843	7.946	8.934
Initial conv. ratio	.739	.630	.580	.560
Integrated conv. ratio	.753	.730	.712	.704
Initial Loading Requirements, kg/GWe-yr				
U-235	1298	1782	2213	2430
ST U_3O_8	293	421	530	585
Equilibrium Loading, kg/GWe-yr				
U-235	735	538	509	503
ST U_3O_8	166	127	122	121
Equilibrium Discharge, kg/GWe-yr, average				
U-235	208	72	48	41
Fissile Pu	255	167	140	131
U-235 discharge enrichment	.528	.40	.37	.36
Annual Net Requirements, kg/GWe-yr, average				
U-235 consumed	527	466	461	462
U-235 fissioned	425	371	362	362
Pu fissioned in situ	404	434	436	436
Enrichment requirements, SWU	7.33×10^4	7.44×10^4	7.96×10^4	8.22×10^4
30-Year Requirements				
ST U_3O_8	5107	4104	4068	4094
kg Fissile Pu	7650	5010	4200	3930

Tables 5 and 6 summarize similar information at a water-to-fuel ratio of 1.5 — a somewhat drier lattice with a higher conversion ratio. Despite the improved conversion ratio, the fuel utilization is not as good as for the reference design, largely because of the loss in reactivity due to the drier lattice (increased resonance absorption in U-238) and, consequently, the higher enrichment required. Calculations were not made for a wetter lattice because such a lattice spacing would result in positive temperature coefficients of reactivity, which would be unacceptable from the safety standpoint. Similar results were obtained in an MIT study*, which showed poorer resource utilization for both drier and wetter lattices. Thus, it is concluded that the current PWR lattice spacing is the optimum design (or very nearly so) from the standpoint of long-term resource utilization.

The data summarized in Tables 1 through 6 are shown graphically in Figs. 1 through 3. Figure 1 illustrates the enrichment required to achieve a given burnup. Figure 2 shows the 30-year average resource requirements (standard tons U_3O_8) as a function of fuel burnup, and Fig. 3 shows the corresponding fissile plutonium production.

Significant improvement in resource utilization is possible by any means that would accomplish some measure of rapid refueling to approach the idealized on-line refueling. Any such improvement, regardless of projected power demand in the future, would tend to defer a need for fast breeder reactors compared to the current PWR fuel cycle, by an amount related to the improvement actually accomplished.

---

* K. C. Garel and M. J. Driscoll, Fuel Cycle Optimization of Thorium and Uranium Fueled PWR Systems, MIT Energy Laboratory, MIT-2295T10-06, October 1977.

4.0 ALTERNATE FUEL CYCLES

Several alternate fuel cycles have also been surveyed, including the denatured fuel cycle in both PWRs and in the spectral shift reactor (modified PWR using a variable concentration of heavy water to compensate for excess reactivity). Results of these calculations are summarized in Table 7. Examination of Table 7 reveals that the denatured fuel cycle can effect significant reductions in the resource requirements and in the quantity of plutonium produced. However, the spectral shift reactor concept does not result in any significant reduction in resource requirements or in plutonium production. Generally, a higher D_2O content would lead to an increase in conversion ratio. However, the loss in reactivity (at the normal PWR lattice spacing) requires a higher enrichment that in turn tends to reduce the conversion ratio. The net effect is only a small increase in conversion ratio. Perhaps an optimized lattice spacing might improve the fuel utilization characteristics of the spectral shift reactor concept, but such calculations were not made in the present study.

The denatured fuel cycle (limited to 20% enrichment of uranium in U-235) requires approximately 43% less uranium ore than the reference PWR fuel cycle and produces approximately 2½ times less plutonium. With U-233 makeup (12% U-233 from an unspecified source), the plutonium production is approximately the same as for U-235 makeup.

Table 8 summarizes several cases calculated to illustrate the effect of enhanced burnup (obtained by a higher enrichment) and of thorium loading in a throwaway fuel cycle. Although there are a limited number of cases shown in Table 8, it tentatively appears that (1) some improvement in resource utilization can be accomplished in the denatured fuel cycle by increasing enrichment and fuel burnup, and that (2) the advantage of the denatured fuel cycle is realized only if U-233 is recycled. The thorium content, between 60% and approximately 80% in a throwaway fuel cycle, does not significantly affect resource utilization but does affect the quantity of plutonium produced. It is also interesting to note that the discharge U-235 enrichment is sufficiently high to warrant salvaging for either recycle or use in some other fuel cycle.

Table 7 FUEL UTILIZATION CHARACTERISTICS FOR LWRs UNDER VARIOUS FUEL CYCLE OPTIONS

	U-Cycle Throwaway	U-Cycle U-Recycle	Denatured Th-U-235 Makeup	SSCR U-Cycle Throwaway	SSCR U-Recycle	LWR Denatured Th-U-233 Makeup	SSCR Denatured Th-U-233 Makeup	
Discharge Burnup, Mwd/kg	34.3	33	35	33	33	35.3	35.3	
Equilibrium Fissile Enrichment, % HM	3.2	3.2	4.4	2.9	2.9	3.0	2.8	
Initial Loading Requirements, kg/GWe-yr								
U-235	1796	1796	2508	1648	1648	45	42	
ST U_3O_8	428	428	632	390	390			
U-233						1850	1769	
Equilibrium Loading, kg/GWe-yr								
U-235	807	807	736	740	740	15	14	
ST U_3O_8	192	192		175	175	1	1	
U-233			341			720	672	
Equilibrium Discharge, kg/GWe-yr, average								
U-235	182	182	256	168	168	16	18	
U-233			347			410	425	
Fissile Pu	160	160	66	200	200	69	83	
U-235 enrichment in discharge	.76	.76	5.43	.686	.686			
Annual Net Requirements, kg/GWe-yr, average								
U-235	625	625	485	572	572			
ST U_3O_8	192	159	123	175	145			
U-233						310	247	
Fissile Pu	-160	-160	-66	-200	-200	-69	-83	
Enrichment requirements, SWU	1.2×10^5	1.51×10^5	1.2×10^5	1.03×10^5	1.37×10^5			
Enrichment of makeup U, % (blending)	3.2	36.86	63.84	2.9	32.35			
30-Year Requirements								
ST U_3O_8	6000	5039	4199	5465	4595			
kg U-233	0	0	0	0	0	10,840	8932	
kg Fissile Pu	4800	-4800	1980	-6000	-6000	-2070	-2490	

Table 8 FUEL UTILIZATION IN PWRs WITH THE DENATURED FUEL CYCLE

	High Burnup (60% Th)	(60% Th)	(78% Th)	
Discharge Burnup, Mwd/kg	80	34.8	35.3	
Equilibrium Fissile Enrichment, % HM	8 (20% in U)	4.5 (11.25% in U)	4.4 (20% in U)	
Initial Loading Requirements, kg/GWe-yr				
U-235	1905	2627	2508	
ST U_3O_8	480	657	632	
Equilibrium Loading, kg/GWe-yr				
U-235	865	1119	1080	
ST U_3O_8	218	280	272	
Equilibrium Discharge, kg/GWe-yr, average				
U-235	175	389	349	
U-233	125	210	253	
Fissile Pu	61	102	66	
U-235 discharge enrichment	4.98 (3.56 in U-233)	4.26 (2.3 in U-233)	7.39 (5.4 in U-233)	
Annual Net Requirements, kg/GWe-yr, average				
U-235 consumed	690	730	731	
Enrichment requirements, SWU	1.98×10^5	2.38×10^5	2.47×10^5	
30-Year Requirements				
ST U_3O_8	6802	8777	8520	
kg Fissile Pu	1830	3060	1980	
kg U-233	3750	6300	7590	

5.0 FUEL UTILIZATION IN HIGH TEMPERATURE GRAPHITE REACTORS (HTGRs)

The attainment of good fuel utilization in high temperature graphite-moderated reactors, as currently designed, makes an attractive base from which alternate fuel cycles can be examined. A number of calculations have been made to examine the performance of such cycles utilizing the mechanical design described in GASSAR-6* as a basis. The reactor operates at 3000 Mw(t) with an output of 1160 Mw(e). The design fuel cycle for the HTGR described in GASSAR achieves a burnup of approximately 98,000 Mwd/mt with an initial conversion ratio of 0.65 (0.68 for initial core). The initial core is loaded with 37,487 kg thorium and 1747 kg uranium; uranium that is approximately 93% enriched is used as feed fissile material for the initial core and reload segments. The use of highly-enriched feed material is not advantageous from a nonproliferation standpoint, so a number of alternate fuel cycles were examined.

A parametric study of the effect of thorium content on conversion ratio was performed, while maintaining the reference U-235 loading (4.1 wt%). This results in a combined variation of U-238 and Th-232 and hence, a varying effect on resonance capture in these fertile isotopes. Figure 4 shows the effect of this variation of the initial conversion ratio and reveals a maximum ICR of 0.936 at a thorium content of about 47 wt%. The curve shows a rather broad maximum with little variation from the maximum ICR for thorium contents of 33 to 60 wt%. Other fuel cycle characteristics for varying thorium content were also obtained from the parametric study. The effect on k_∞ is shown in Fig. 5 as a function of fuel exposure. For all cases with a thorium content less than the reference loading, the reactivity does not appear to be sufficient to attain the reference burnup. In all cases, however, the reactivity curve is flatter towards high burnup than the reference case. Fissile plutonium inventory increases significantly as the thorium content is decreased, since the thorium is replaced by U-238. Figure 6 shows the fissile plutonium inventories as functions of exposure for various thorium contents.

In order to increase the reactivity of an HTGR fuel cycle with near-optimum ICR and achieve the reference burnup,

---

* GASSAR-6, General Atomic Standard Safety Analysis Report.

the initial U-235 enrichment was increased. For this analysis, a fixed thorium content of 60 wt% was used. The U-235 enrichment was increased until the burnup-averaged k_∞ was equal to the reference value of 1.0036. This was essentially attained (k_∞ = 1.0028) at an enrichment of 27.5% (110 kg U-235, 290 kg U-238, and 600 kg Th-232 per metric ton of heavy metal). The increased U-235 loading causes a decrease in conversion ratio to 0.676.

The trade-off in conversion ratio in order to increase reactivity is present at varying degrees over the whole range of thorium loadings. This can be seen in Fig. 7, where the initial conversion ratio is shown as a function of thorium content with the constraint that all points satisfy the reference cycle reactivity. Comparing this curve to Fig. 4 shows the decrease in the initial conversion ratio that occurs in order to increase the reactivity to the reference value. Figure 7 also shows the integrated conversion ratio, which is larger than the initial conversion ratio and, increasingly so, as thorium content increases. The difference between the initial and integrated values here appears to be most likely due to the increasing reactivity importance of U-233 feed during the fuel cycle. Since a lower proportional fissile inventory can exist for the same reactivity as the U-233 inventory increases, the improved neutron economy yields a higher conversion ratio. Both curves also show that the reference GASSAR fuel cycle appears to have the maximum conversion ratio.

Both the initial conversion ratio and the integrated ratio could differ slightly from the values presented here because of self-shielding in the fuel channels themselves. This effect is not considered in these data, but calculations show that the U-238 resonance integral may be reduced an estimated 5 to 10%. This effect, combined with the self-shielding of the fuel hole and attendant reactivity effects, could influence the conversion ratio.

Analyses were also performed to determine the fuel loading for a U-233/natural uranium/thorium fuel cycle. The thorium content was again fixed at 60 wt% to achieve near optimal conversion. U-233 was assumed to be readily available for mixing with natural uranium for initial loading. The U-233 content was varied to achieve the reference burnup-averaged reactivity, and the resulting fuel loading was 60 kg U-233, 2.4 kg U-235, 337.6 kg U-238, and 600 kg Th-232 per ImtHM. The initial conversion ratio for this case was 0.90. Figures 8 and 9 show k_∞ and specific masses for isotopes of interest for this case.

Additional analyses of fuel cycles with initial U-233 loading were also performed to investigate the self-sustaining recycle potential. In this case, the discharge U-233 content was designed to be the same as the initial loading. Reactivity requirements were satisfied by varying both the U-233 and U-235 initial loading. Thorium loading for this cycle was again fixed at 60 wt%. Combinations of U-233 loading and various U-235 enrichments were examined to attain the desired U-233 discharge, as well as the reference burnup-averaged k_∞ . For an initial loading of 29 kg U-233, 58 kg U-235, 313 kg U-238, and 600 kg Th-232 per ImtHM, these conditions were essentially met. Figures 10 and 11 depict pertinent characteristics for this case.

A once-through fuel cycle was also examined utilizing enriched uranium and thorium. The maximum burnup attainable, constrained by maintaining the reference case reactivity averaged over the complete fuel exposure, was determined for three U-235 enrichment values with a fixed thorium content of 60 wt%. The cases examined and the burnup allowed under the given constraint are listed below.

Loading (kg/ImtHM) U-235/U-238/Th-232	U-235 Enrichment (%)	Allowable Burnup (Mwd/ImtHM)
91/209/600	22.75	67,600
110/190/600	27.5	96,400
132/268/600	33.0	136,800

Figure 12 shows k_∞ as a function of burnup for the three cycles noted above. Discharge fissile plutonium inventory increases primarily due to exposure only, since the initial U-238 inventory differs only slightly for the three cases.

The information developed from the HTGR analyses described above was re-evaluated to optimize the fuel cycle with even better nonproliferation features and to provide fuel utilization data on a basis that could be easily compared to other reactor types and fuel cycles. Table 9 provides a summary of fuel utilization for alternate HTGR fuel cycles normalized to a 1000 Mw(e) plant. Data for three alternate cycles plus the reference thorium cycle are presented.

The once-through and recycle alternates are presented for the case of 20% enrichment in uranium, which is considered an upper limit for acceptable nonproliferation. Thorium

content for the two cycles is 20%. As noted earlier, the fuel utilization of the reference HTGR fuel cycle is quite good, but highly enriched fuel material is required. Compared to the once-through and recycle alternates, the reference cycle is more favorable for resource utilization and low plutonium production. The production of U-233 is higher in the reference case, however, due to a larger loading of thorium.

Table 9 also shows another alternate HTGR fuel cycle that provides some interesting points. This cycle has a thorium content of 80% and uses feed uranium of 35% enrichment. This enrichment is greater than the nonproliferation limit, but also much lower than the enrichment for the reference cycle. Although the enrichment is greater than the proposed limit, and would appear to have a disadvantage from a nonproliferation standpoint, the plutonium production is much lower than the once-through or recycle alternates. This is obviously advantageous for nonproliferation. This alternative has significant U-233 production, which is about 15% higher than the reference cycle. This cycle points out the usual conflict of optimizing a fuel cycle to minimize proliferation risk while providing good fuel utilization. However, the graphite-moderated system can have a range of reasonable values for fissile material production that is attractive for nonproliferation aims as well as fuel utilization. The optimization of such a cycle, then, depends largely on the degree of nonproliferation constraints on the front-end versus the back-end of the fuel cycle.

Table 9 FUEL UTILIZATION IN HIGH TEMPERATURE GRAPHITE REACTORS

	HTGR Denatured Once-Thru	HTGR Denatured U-Recycle	HTGR Reference Th-Cycle	HTGR Denatured High Th
Discharge Burnup, Mwd/kg	98	98	96.4	98
Equilibrium Fissile Enrichment, % HM	16 (20% in U)	16 (20% in U)	4.1 (92.6% in U)	7 (35% in U)
Initial Loading Requirements, kg/GWe-yr				
U-235	2134	2134	780	1385
ST U_3O_8	538	538	198	350
Equilibrium Loading, kg/GWe-yr				
U-235	1156	1048	303	506
ST U_3O_8	291	264	77	128
U-233	--	108.7	--	--
Equilibrium Discharge, kg/GWe-yr, average				
U-235	504	472	15	80
U-233	76	112	156	178
Fissile Pu	215	216	0.5	31.8
U-235 discharge enrichment	10.66	9.96	8.06	7.9
Annual Net Requirements, kg/GWe-yr, average				
U-235	652	576	288	426
ST U_3O_8	291	148	77	128
U-233	-76	--	-156	-178
Fissile Pu	-215	-216	-0.5	-31.8
Enrichment requirements, SWU	2.65×10^5	1.44×10^5	7.67×10^4	1.21×10^5
Enrichment of Make-up U, % (blending)	20	51.6	92.6	35
30-Year Requirements				
ST U_3O_8	8977	4830	2431	4062
kg U-233	-2280	--	-4680	-5340
kg Fissile Pu	-6450	-6480	-15	-954

6.0 FUEL UTILIZATION IN CANDU TYPE REACTORS

Calculations were performed for various fuel cycles in a CANDU-type reactor. Results of these calculations are summarized in Table 10 (see also Table 4 for additional data). On the basis of these calculations, the following tentative conclusions can be made.

- Increased U-235 enrichment and correspondingly higher fuel burnup for the uranium cycle results in improved resource utilization and reduced annual plutonium production.

- The denatured fuel cycle, with either U-233 or U-235 makeup (and U-233 recycle) results in a significantly improved resource utilization and reduced annual plutonium production over the uranium-only throwaway fuel cycle. To realize the advantage, recycle of the U-233 is necessary.

- Enrichment with plutonium (and plutonium recycle) will also accomplish significant reductions in uranium resource requirements.

- Enrichment with U-233 in a throwaway fuel cycle (see Table 11) significantly reduces both the long-term uranium resource requirements and the net plutonium production.

Because of the lower enrichment requirements, higher conversion ratio, and capability for on-line refueling, CANDU-type reactor systems generally show better resource utilization than conventional LWR systems.

Table 10 FUEL UTILIZATION CHARACTERISTICS FOR CANDU REACTORS UNDER VARIOUS FUEL CYCLE OPTIONS

	Natural U	Enriched U (No Recycle)	Pu Cycle	Denatured Th Cycle U-233 Makeup	Denatured Th Cycle U-235 Makeup	Denatured Th Cycle Once-through		
Discharge Burnup, Mwd/kg	7.5	16	16	16	16	16		
Equilibrium Fissile Enrichment, % HM	.711	1.0	N + .3	1.46 (11% in U)	1.65 (13% in U)	1.88 (20% in U)		
Initial Loading Requirements, kg/GWe-yr								
U-235	962	1146	843		1946	2343		
ST U_3O_8	176	233	154		487	590		
Pu			354					
U-233				2581				
Equilibrium Loading, kg/GWe-yr								
U-235	880	573	412		292	1077		
ST U_3O_8	162	117	75		73	271		
Fissile Pu			173					
U-233				836	666			
Equilibrium Discharge, kg/GWe-yr, average								
U-235	250	58	54		95	344		
U-233				667	666	445		
Fissile Pu	367	205	218	50	56	38		
U-235 discharge enrichment	.207	.104	.095		1.37	6.87		
Annual Net Requirements, kg/GWe-yr, average								
U-235	630	515	358		197	733		
ST U_3O_8	162	117	76		48	271		
U-233				169		-445		
Fissile Pu	-367	-205	-45	-50	-56	-38		
Enrichment requirements, SWU		2.17×10^4			5×10^4	2.5×10^5		
Enrichment of Makeup U, % (blending)	.711	1.0	.711		4.41	20.0		
30-Year Requirements								
ST U_3O_8	4874	3626	2358	0	1879	8449		
kg U-233	0	0	0	7482	0	-1140		
kg Fissile Pu	-11,010	-6150	-1350	-1500	-1680	-13,350		

Table 11 FUEL UTILIZATION IN NATURAL URANIUM, CANDU-TYPE REACTORS WITH U-233 ENRICHMENT

	1.0% Enrichment	1.2% Enrichment	1.5% Enrichment	
Discharge Burnup, Mwd/kg	22	29	40	
Equilibrium Fissile Enrichment, % HM	N + .282 U-233	N + .484 U-233	N + .785 U-233	
Cycle time, yrs.	5.0	6.58	9.08	
Initial Loading Requirements, kg/GWe-yr				
U-235	782	767	749	
ST U_3O_8	144	142	137	
U-233	305	513	831	
Equilibrium Loading, kg/GWe-yr				
U-235	299	227	164	
ST U_3O_8	55	42	30	
U-233	117	153	180	
Equilibrium Discharge, kg/GWe-yr, average				
U-235	14	6	1	
U-233	5	3	3	
Fissile Pu	155	120	88	
U-235 discharge enrichment	.034 (.012 U-233)	.02 (.01 in U-233)	.01 (.004 in U-233)	
Annual Net Requirements, kg/GWe-yr, average				
U-235 consumed	285	221	163	
Enrichment requirements, SWU	0	0	0	
30-Year Requirements				
ST U_3O_8	1739	1360	1007	
kg Fissile Pu	-4650	-3600	-2640	
U-233	3698	4950	6051	

7.0 TANDEM FUEL CYCLE

During the course of the study, results of prior calculations on the tandem fuel cycle were compiled and a paper prepared for presentation at the 1977 Winter Meeting of the American Nuclear Society.* This particular tandem fuel cycle considered the irradiation of spent LWR fuel elements (after refabrication) in heavy water reactors of the CANDU type. Other conceivable tandem fuel cycle concepts include the following:

- Metallic fuel elements irradiated in an LWR followed by a second irradiation in an HWR,
- spent fuel from a spectral shift reactor irradiated in an HWR, and
- spent HTGR fuel (refabricated) irradiated in an LWR with a possible third cycle in an HWR.

The effect of the tandem fuel cycle is to salvage the residual reactivity and to effectively extend the fuel burnup by extracting additional energy in the second irradiation. Although a significant period of time will elapse following the first irradiation (cooling, refabrication, and second irradiation), an approximate indication of the overall average resource utilization and plutonium production can be obtained by calculating these factors for fuel of a higher burnup using the discharge fuel compositions at the end of the second irradiation. Table 12 summarizes results of these calculations for several tandem fuel cycles that have been investigated.

---

* Moss, M. K. and Roach, K. E., the LWR-HWR Tandem Fuel Flow Concept, presented at the ANS meeting, San Francisco, California, November 28, 1977.

Table 12 FUEL UTILIZATION IN PRESSURIZED WATER REACTORS WITH THE TANDEM FUEL CYCLE

	Ref PWR	Ref PWR WFP Cleanup	Denatured Fuel	
Discharge Burnup, Mwd/kg	44.4	49.7	77.4	
Equilibrium Fissile Enrichment, % HM	3.2	3.2	4.5	
Initial Loading Requirements, kg/GWe-yr				
U-235	1308	1143	1027	
ST U_3O_8	312	273	257	
Equilibrium Loading, kg/GWe-yr				
U-235	624	557	503	
ST U_3O_8	149	133	126	
Equilibrium Discharge, kg/GWe-yr, average				
U-235	60	29	21	
U-233	--	--	95	
Fissile Pu	100	81	33	
U-235 discharge enrichment	.33	.18	5.5	
Annual Net Requirements, kg/GWe-yr, average				
U-235 consumed	564	528	482	
Enrichment requirements, SWU	9.2×10^4	8.3×10^4	1.07×10^5	
30-Year Requirements				
ST U_3O_8	4633	4130	3911	
kg Fissile Pu	3000	2430	990	
kg U-233	--	--	2850	

8.0 ISSUES IN THE UTILIZATION OF DENATURED FUEL IN LWRs

8.1 General

One of the more promising methods of improving resource utilization and reducing the proliferation risk from plutonium production involves the use of a mixture of uranium and thorium oxides as reactor fuel (denatured fuel cycle). By reducing U-238 content, the amount of plutonium produced can be correspondingly reduced. However, the thorium in the fuel results in the production of U-233, a fissile material nearly as good as plutonium for weapons use. Consequently, it is necessary to have some U-238 in the fuel to dilute the U-233 produced, thereby precluding its use as weapons material (at least without isotope separation, a difficult and expensive process). Thus, a compromise is necessary between the reduction in quantity of plutonium produced and the percent U-233 in the uranium of the discharged fuel. For current LWRs, a reasonable compromise would reduce plutonium production by a factor of 4 or more, while avoiding the existence of uranium enriched to more than 12% U-233 or 20% U-235. Further reduction in plutonium production could be accomplished, but only by using fuel more highly enriched in U-233 or U-235.

Recognizing the potential for improved proliferation-resistance by the denatured fuel cycle, it is necessary to consider the factors that would affect its acceptance and use by the nuclear power industry. These considerations involve technical and economic issues, as well as possible incentives that may be necessary for adoption of the denatured cycle. Subsequent paragraphs present discussions of the initial steps necessary to identify the issues and the development program needed to resolve them.

Utilization of the denatured U-Th oxide fuel cycle depends on the successful identification and subsequent resolution of a number of issues. Some of these issues are technical, some are non-technical, and all have an economic aspect. To be considered by the nuclear power industry, the denatured fuel cycle must be economically competitive with the existing fuel cycle, either directly or as a result of government incentive programs or legislative prohibitions. Consequently, the major issue is one of economics. However, there are certain issues relating to safety, licensing, and operation that must be resolved even if the requisite economic incentives are

present or may be reasonably expected to exist in the future. There are also subjective or philosophical issues that affect the nuclear industry's potential acceptance of an alternate fuel cycle, although the economic impact of those issues cannot be quantified.

The principle issues in the ultimate acceptance of the denatured fuel cycle by the nuclear industry are discussed in more detail in the following paragraphs.

8.2 Safety/Licensing Issues

Introduction of an alternate fuel cycle concept (or even the introduction of a new fuel design within an existing concept) requires that the safety of the new cycle be demonstrated both analytically and in a practical demonstration and prototype test program. Generally, the denatured fuel cycle is not expected to require plant modification: the safety/licensing issues revolve about the performance characteristics of the U-Th oxide fuel. The licensing issue, next to economics, will likely be the principle underlying reason for industry opposition to the denatured fuel cycle. Much of the anticipated industry resistance to a new fuel cycle would derive from a reluctance to become entangled in a licensing process that could involve considerable uncertainties, extensive time delays, and unpredictable expenses.

There is little irradiation experience with thorium oxide fuel in the United States (only the Indian Point-1 and Elk River first cores), and no irradiation data has been found on mixed U-Th oxides of the composition that would be employed in the denatured fuel cycles. Certain safety analyses (e.g., the loss of coolant analysis) may be affected by the physical characteristics of the mixed oxide fuel. Other, more readily apparent, safety concerns include the possibility of fuel densification, eutectic formation, ratcheting with clad, fuel swelling, and fission gas pressure within the fuel rods. In some cases, where information is not already available, a research and development program may be required to measure physical properties (such as specific heat, melting temperature, thermal conductivity, etc.) of the U-Th oxide fuel in pellet form.

8.3 Operation/Performance Issues

Operation/performance characteristics of the U-Th oxide fuel that must be evaluated in assessing the acceptability of the denatured fuel cycle include, in addition to in-reactor performance during the power-production period, all factors related to the nuclear fuel cycle. These factors include

- Uranium and thorium ore availability;
- conversion to oxide form and suitable blending operations;
- fuel fabrication;
- reactor operation;
- spent fuel storage;
- reprocessing; and
- recycle and waste management.

Within the reactor core, operation with the denatured fuel is not expected to differ greatly from corresponding operation with conventional uranium oxide fuel, except to the extent operations may be affected by the physical properties of the mixed oxide fuel, as mentioned above. Presumably, plentiful supplies of thorium ore are available. However, there is considerable concern about appropriate methods of mixing the uranium and thorium oxides — i.e., blending or co-precipitation — to assure a uniform mixture that will not segregate or result in unacceptable hot-spots during reactor operation. Fuel fabrication techniques must also assure acceptable performance characteristics of pressed-and-sintered pellets (or vibratory-compacted fuel elements).

Storage of spent denatured fuel will not likely differ significantly from storage of conventional uranium fuel. However, in chemical reprocessing, it is known that thorium oxide is more difficult to dissolve than uranium oxide, so a different head-end process (modified Thorex process) than that used for the uranium fuel cycle will likely be required. In addition to recovery of the thorium and uranium, some plutonium will be recovered. Disposition of the plutonium (and fission-product wastes if different) must also be considered. In addition, recycle of the uranium (then containing U-233) would impose additional requirements, such as remote fuel fabrication facilities as a result of U-232 in the fuel, blending with highly-enriched uranium to restore initial

reactivity, and the accommodation of increasing parasitic absorption due to U-234 and U-236 accumulation.

8.4 Economic Issues

Assuming that the technical problems can be solved (it is frequently assumed that, with enough money, they can be), then the fundamental issue relating to acceptance of the denatured fuel cycle by the nuclear power industry is one of economics. The economic issues, however, include not only the actual costs of the denatured fuel cycle, but also any government incentive programs or legislative constraints that affect comparative fuel cycle costs. For acceptance and introduction of the denatured fuel cycle entirely by the private sector, all cost burdens would have to be accepted by the industry and the sole incentive would be a reduction in the fuel cycle costs.

At the present time, it is doubtful that the denatured fuel cycle could compete economically with the the conventional uranium fuel cycle if the total cost burden were to be borne by industry. Even in the future, assuming chemical reprocessing is permitted, it is unlikely that the denatured fuel cycle can compete successfully, at least until uranium ore costs have risen substantially above present levels. Government funding will likely be necessary to support the requisite research and development program for the denatured fuel cycle. In addition, government incentive programs may be required to induce acceptance of the denatured fuel cycle by industry. These may be direct subsidies, indirect subsidies in the form of cost guarantees or buy-back policies, or legislative restrictions — for example, prohibiting recycle of uranium-only fuel while permitting recycle of the denatured fuel.

The principle factors that result in a nominally-higher cost for the denatured fuel cycle include the following:

- Thorium ore mining and procurement;
- additional cost of blending operations;
- higher enrichment and SWU requirements for the initial core loading;
- greater difficulty of reprocessing fuel containing thoria; and

- remote recycle fuel fabrication and additional shipping costs resulting from the inherent U-232 contamination and its associated gamma radioactivity.

Offsetting these factors are the better neutronic properties of U-233 (conversion ratio and reactivity), the reduced power peaking problems of recycled fuel, the smaller radiological hazard of U-233 compared to plutonium, improved uranium resource utilization, and the reduced risk of weapons proliferation.

8.5 Subjective/Political Issues

The subjective issues affecting the utilization of a denatured U-Th fuel cycle lie behind the basic question — "Why change?" If the alternate cycle were sufficiently attractive, economically and technically, its inherent merits would cause it to be accepted by the industry. However, the principle attractive feature of the alternate cycle — nonproliferation — does not naturally fit into the commercial arena, especially in the case of reactors-for-export, where the higher cost expected for the denatured fuel cycle (in the absence of government subsidies) could be a major competitive disadvantage. This difficulty is probably compounded by the seeming lack of confidence on the part of industry that change to this fuel cycle would really have a significant international impact on proliferation. Although the industry realizes that an unsuitable fuel cycle can lead to proliferation problems, the experience accumulated with the present uranium cycle is frequently interpreted to indicate reasonable proliferation resistance. Furthermore, since it is probably impossible to develop a reactor system and fuel cycle that is proliferation-proof, the basic industry concept of the problem is a matter of degree. Therefore, the basic subjective/political issue affecting utilization of a denatured U-Th fuel cycle is whether the industry, including the consumer, believes that the use of this cycle would have a measurable effect on an international political issue and that the advantages justify any additional cost.

8.6 Resolution and Initial Steps

In seeking to resolve the interrelated issues discussed above, the initial steps should include a detailed review to identify those areas in which there are substantial differences from the existing uranium fuel cycle. From this survey/identification stage, it will be possible to develop a program plan or sequence of integrated projects to resolve the technical issues and to establish a responsible data base for economic projections. This preliminary survey should be able to identify major licensing and operational issues that are sufficiently different from those of conventional UO_2 fuel cycles to require extensive evaluation and analysis.

It seems likely that at least two proof-test irradiations will be necessary: a demonstration irradiation of three or four fuel assemblies, followed by prototype irradiation of a full core loading of U-Th oxide fuel. Preceding, and concurrent with, these proof-test irradiations, a program for direct measurement of some important design parameters (e.g., thermal conductivity, melting points, eutectic formation, material segregation, fission gas release, etc.) will likely be necessary. These irradiation tests will not only provide a base of experimental data, but, of almost equal importance, will allow some experience to be gained in the licensing process.

Concurrently, other R&D projects will be required to establish costs associated with ore availability, fuel fabrication, reprocessing, and recycling. These data will serve as a base for defining the denatured fuel cycle costs and for identifying any government incentive programs necessary to encourage industry acceptance.

Some provision should also be included in the overall program plan to allow industry participation and feedback. Surveys of industry attitudes, and information exchange meetings, are among the possibilities that could lead to industry participation and support. Resolution of the subjective or political issues can be quite difficult, particularly if it is realized that the industry is inclined to translate all other considerations into one of economics. Convincing the nuclear industry that the denatured fuel cycle could have a positive effect on the international proliferation issue, and that the benefits are worth the additional expense and inconvenience, will be a major task.

9.0 HEAVY WATER REACTOR DESIGN

9.1 Introduction

At the request of the Agency, Southern Science performed a preliminary, conceptual design of a heavy water reactor that might be licensed in the United States. In order to achieve the good fuel utilization obtained in the Canadian pressurized heavy water reactors (CANDU), the investigation centered on the heavy water moderated, heavy water cooled concept. Probably the most limiting restraint placed on a U.S. heavy water reactor is the conclusion that a pressure tube-pigtail-header arrangement (as in the CANDU reactors) is unacceptable; it is felt that the failure of a single pigtail would require an accident analysis based on the assumption that the single failure initiates a propagating failure of all pigtails associated with one end of the reactor. On the basis that the assumed consequences of a pigtail failure make acceptance of the CANDU system uncertain in the United States, Southern Science proceeded to consider many heavy water reactor designs that have been proposed or utilized, in an attempt to obtain a hybrid design that might be acceptable for construction and operation in this country. Some of the basic considerations in the evaluation, and the conceptual design that resulted from that evaluation, are presented in the following sections.

9.2 Basic Considerations

9.2.1 Core Arrangement

Both horizontal and vertical orientations of fuel within the reactor core were considered, and it was concluded that a vertical orientation would most nearly resemble the typical layout seen present in U.S. commercial nuclear power plants. Consequently, the vertical orientation was adopted. Consistent with the access requirement for on-line refueling for maximum resource utilization and optimum economics, the vertical fuel rods were grouped into assemblies, with the assemblies separated by moderator, as in the CANDU reactors.

Since it had been concluded that a pigtail arrangement could not be utilized, the CANDU calandria arrangement, in which the moderator is cool, was not deemed usable. Instead, an

arrangement in which the fuel columns are surrounded by process tubes that are immersed directly in the moderator was adopted. Consequently, although there is flow separation, there is no insulation between the primary coolant and the moderator: thus, energy will be transferred to the moderator as well ad deposited in it, and it will operate at "warm" temperatures as compared to the "cool" moderator in CANDU plants and the "hot" moderator in LWRs.

The fuel will be slightly enriched in order to obtain a longer lifetime than that achieved in the CANDU plants, and the spacing of the process tubes will likely be somewhat smaller than the lattice in the Canadian plants, in order to obtain a negative power coefficient of reactivity in the U.S. plant.

With the vertical arrangement of the process tubes in the reactor, the only way the Canadian bi-directional push-through method of on-line refueling can be accomplished is by having bottom access to the fuel positions. This means opening a process tube (or its extension) at the bottom, consequently risking a loss of primary coolant from the bottom of the tube in the event of a poor refueling-machine connection. Such an arrangement would likely not be acceptable to licensing authorities. Therefore, it was decided that the process tubes would be accessible for refueling only from the top ends.

With one-end-only refueling access, the refueling machine tends to become very complicated. This complexity is increased by considering such possibilities as partial-length fuel bundles, axial shuffling of the fuel, or any operation involving the removal and replacement of a component of the fuel column. In view of the inherent complication of a one-end refueling machine and the relatively small gains from the bi-directional push-through mode, it was decided that the fuel for the reactor would simply be inserted, irradiated, and removed, with no shuffling or repositioning within the core.

With the in-out arrangement, the use of full-length bundles is indicated. With full-length bundles, fabrication costs are lower and end-peaking in the core is avoided. In addition, fission gas spaces can be provided at the ends of the fuel rods, above and below the active core region. Frequency of fuel manipulation is decreased. For example, with an average specific power of around 20 Kw/kg, and a fuel enrichment sufficient to give a fuel exposure of around 22,000 Mwd/mtU,

the fuel lifetime in the core is 1100 days. If one assumes that the plant produces around 600 Mw(e), then the reactor will probably require around 350 to 450 process channels. For a core lifetime of 1100 days, then, one channel must be unloaded and loaded each 2½ or 3 days. Such a schedule will not place unreasonable requirements on the refueling machine and other components in the refueling system.

On the basis of the above considerations, a conceptual design of a heavy water reactor for use in the United States was developed. That design is described in the following sections.

9.3 Nuclear Steam Supply System Conceptual Design

9.3.1 Physical Arrangement

As stated above, the use of pigtails and headers, similar to the arrangement used in the CANDU plants, is considered unacceptable in this country. Consequently, the true pressure-tube concept, using small pressure-containing components, is unacceptable. This leaves for consideration the vessel concept, which, because of the large lattice spacing in a D_2O reactor, requires a very large pressure vessel. Although the conceptual design presented here is adaptable to practically any size reactor, the development of the concept was based on the possibility of designing plants to produce as little as 500 or 600 Mw(e). Even at that power level, the vessel required for the heavy water reactor exceeds present capacities for fabrication and shipment of traditional steel vessels. In view of this, the vessel selected for the conceptual design is made of prestressed concrete (a PCRV — prestressed concrete reactor vessel).

Once one makes the decision to utilize a PCRV, the next consideration is whether or not to consider an integrated nuclear steam supply system: that is, one in which the components of the primary system are housed within a single, large, prestressed concrete vessel. In the case of the heavy water reactor, the use of a very compact system is indicated in order to reduce the inventory of heavy water and, consequently, the cost of the power produced by the plant. One of the advantages of the PCRV integrated concept is that, with proper design, it should be possible to achieve a system that requires only a relatively small heavy water inventory. In view of these considerations, it was concluded that the integrated plant concept, utilizing the PCRV, would be used.

A review of the basic integrated-plant arrangements indicated that the one-above-the-other arrangement of major components (reactor, pumps, steam generators) proposed by the French in gas cooled reactor designs some years ago was not desirable for a heavy water system. German and Brazilian designs were also unacceptable, although each incorporated some features that were deemed useful in a heavy water plant. The best arrangement was considered to be that adopted by General Atomic for its HTGR designs. The GA design is basically a short, vertical cylinder, with axial "holes" for the major components of the nuclear steam supply system. Such a design lends itself to the use of a top-acting refueling machine, and also permits use of a containment building of reasonable size. In addition, the seismic response of the vessel is good compared to the taller cylinders proposed by others.

The general arrangement of the U.S. heavy water reactor design is shown in Fig. 13. The centerline of the vertical, cylindrical reactor core is located on the centerline of the PCRV. Six steam generators are spaced at equal radii around the core, with a primary coolant pump located beneath each steam generator. The vertical section represented by the illustration is not a true section, as it shows only one of the steam-generator positions in order to include a portion of the refueling system indicated in the left part of the PCRV. The refueling machine is located above the reactor, as shown in the illustration, and the transfer of spent (irradiated) fuel to storage is accomplished through the use of the equipment shown at the left of the illustration. Refueling operations are described in later paragraphs.

Primary system piping consists of short runs and is housed within the PCRV. Moderator and primary coolant flow paths will be described in the next section, but it should be noted here that primary coolant inlet and outlet pipes are above, or at the top of, the core region, precluding the draining of coolant from the fuel assemblies in the event of a pipe or nozzle leak. As the illustration shows, there is an access gallery beneath each primary coolant pump. There are six such galleries, each extending radially in the PCRV support and located beneath a pump position. A control system equipment space is provided beneath the reactor, and a portion of the refueling system extends into the PCRV support region.

The reactor control system itself is not shown on the drawing. The system could consist of combinations of the following: vertical control rods, horizontal control rods, skewed control

rods (as in the Seimens designs), control chains, hydraulically-actuated ball control systems, soluble poison, or moderator poison. In this design, "moderator" poison would actually be primary-coolant poison, but the term moderator poison is used to imply a chemical, such as cadmium sulfate, injected for the purpose of shutting down the reactor as compared to the use of soluble boron for shim control. Note that a moderator dump feature is not included, inasmuch as the system operates at a pressure of approximately 100 atmospheres.

The fuel is located in process tubes, one of which is shown in the drawing. Others are indicated by single, vertical lines in the core region. The drawing also shows that each process tube is served by an access tube that extends out the top of the PCRV. In order to avoid thermal expansion problems, the process tube and its associated access tube are not connected. The fuel column within the process tube is supported by a pedestal that extends out the bottom of the process tube and rests upon the bottom portion of the reactor vessel liner. Positive fuel hold-down in the upward-flow reactor is accomplished by a combination member that also includes a flow orifice and is located near the bottom of the access tube, extending downward into the process tube. Above the orifice/hold-down member in the access tube is a shield plug. The inside diameter of the access tube is greater than the outside diameter of the process tube, permitting off-line replacement of the process tube during the projected lifetime of the plant.

Moderator heat exchangers are located adjacent to the core and reflector region, as shown on the illustration, and an end shield is located immediately above the core. The end shield consists of alternate, horizontal layers of steel and D_2O. An inlet plenum for the primary coolant D_2O is located at the bottom of the reactor, while an outlet plenum that serves both the primary D_2O flow and the moderator D_2O flow is located above the reactor and top shield. The flow arrangement is described in the next section.

9.3.2 Moderator and Coolant Flow Paths

A simplified flow diagram for the nuclear steam supply system is shown in Fig. 14. Most of the flow from the primary coolant pumps is directed to the reactor inlet plenum, from which it enters the individual process tubes (fuel channels). The process tubes are orificed at the outlet ends to achieve a radial match of channel power and coolant flow.

A portion of the pump output is directed through the moderator heat exchangers and then enters the moderator. This flow, which is probably around 10% of the total reactor flow, is subcooled in the moderator heat exchangers to achieve a relatively low temperature for the warm moderator employed in the design. After traversing the moderator region and the end shield region, the moderator D_2O is combined with the primary coolant D_2O in the reactor outlet plenum. The hot D_2O flows from this plenum to the steam generators. By using the moderator heat exchangers, the moderator is operated at a temperature below that of the primary coolant, while at the same time the moderator heat exchangers serve as economizers for the feedwater flow. The feasibility of mixing the outlet flow from the moderator with that from the fuel channels is an economic matter beyond the scope of this conceptual design work, but it does not appear that an undue penalty in mixed outlet temperature is incurred by this approach. If the penalty is, in fact, substantial, a moderator cooling circuit independent of the primary coolant loop can be used. In any case, an economic evaluation will almost certainly indicate the desirability of operating the moderator at a temperature lower than that of the primary coolant.

At a nominal D_2O system pressure of 100 atmospheres, primary coolant enters the process tubes at a temperature of approximately 290°C and leaves at around 310°C. Moderator inlet temperature is about 210°C, and the exit temperature before mixing with the D_2O from the process tubes is around 260°C. With a feedwater temperature of about 180°C, steam is produced at 60 atmospheres and 275°C.

9.3.3 Refueling System and Refueling Operations

As stated earlier, full-length fuel elements are used. When refueling of a lattice position is undertaken, all of the fuel (one assembly) in that position is removed and replaced, with no shuffling attempted.

The refueling machine, which is located above the reactor, is mounted on a carriage which, in turn, is placed upon a transverse carriage, as shown in Fig. 13. The limits of movement of the combined carriages is sufficient to enable the refueling machine to be positioned above any reactor access tube, the transfer machine (shown in Fig. 13), or the new-fuel supply equipment (not shown). The refueling machine is a

pressure vessel that contains a storage turret that rotates about a vertical axis, a guide-tube assembly, and a tool-post assembly. The guide tube, the tool post, and any selected one of the turret storage positions are aligned with a port in the bottom of the machine. The port leads to a coupling assembly (nozzle), which is used to effect a leak-tight seal between the refueling machine and the end of an access tube, the transfer-machine tube, or the new-fuel supply equipment. The refueling machine is filled with heavy water and is maintained at reactor coolant pressure by a helium gas system. It is equipped with a heat exchanger.

The transfer machine, located within the PCRV, provides temporary storage and cooling for spent or defective fuel elements that are in transit from the reactor to the spent fuel pool. The transfer machine contains a turret, with vertical storage positions, that rotates about a vertical axis. The storage tubes in the turret can be aligned with the external tube that extends upward for access by the refueling machine, or with the tool post and bottom-exit transfer tube. The transfer machine is filled with pressurized helium gas or D_2O, and is provided with a heat removal system.

A receiving machine for spent fuel elements is located beneath the transfer machine. The receiving machine is mounted on a turntable and can rotate 180° in order to index to a position beneath the spent fuel hoist.

When a process tube is to be refueled, the refueling machine is moved to a position above the new-fuel supply equipment. The refueling machine nozzle is attached to the equipment, and a new fuel element is drawn into a position in the turret of the refueling machine. The machine then moves to a location above the access tube for the specified lattice position. The machine nozzle is connected to the access tube, the intermediate space is pressurized, the integrity of the connection is checked, and the top of the access tube is opened. The refueling machine then removes the shield plug and the orifice/hold-down member from the access tube, stores them in an unused position in the turret, rotates the turret to a different position, and removes the fuel column from the process tube. The refueling machine turret position containing the new fuel element is rotated until it is in line with the access tube, the new fuel is inserted into the process tube, the internals of the access tube are replaced, the tube is closed at the top, and, following a leakage check, the refueling machine is disconnected from the access tube.

The refueling machine, containing the spent fuel element, is moved to a position above the transfer machine. Following a procedure similar to that employed at the access tube, the refueling machine is used to insert the irradiated fuel element into the turret of the transfer machine. When the fuel is to be removed from the transfer machine, the machine turret is rotated to align the fuel element with the tool post and the bottom-exit transfer tube. The receiving machine is attached to the bottom of the transfer tube, and the irradiated fuel element is lowered into the receiving machine. The receiving and transfer machines are disconnected, the receiving machine rotates 180° about its vertical axis, and the spent fuel hoist lifts the irradiated fuel from the receiving machine and moves it to the spent fuel pool. From that pool, the irradiated fuel element is moved through a spent fuel port to the fuel storage pool, which is located outside the reactor containment. That operation is the last in the sequence of movements associated with one fuel element.

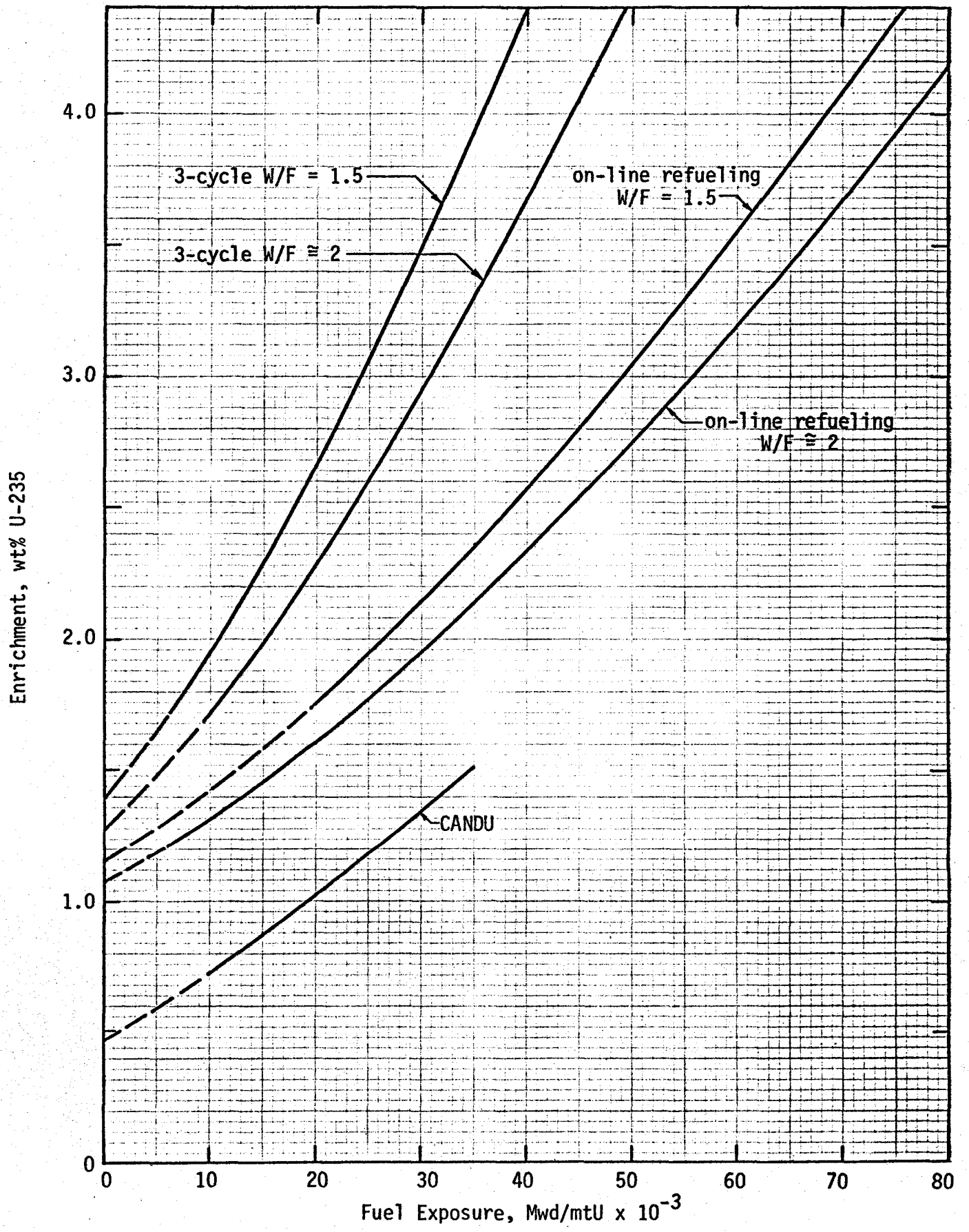

Fig. 1 U-235 enrichment requirements as a function of design fuel burnup.

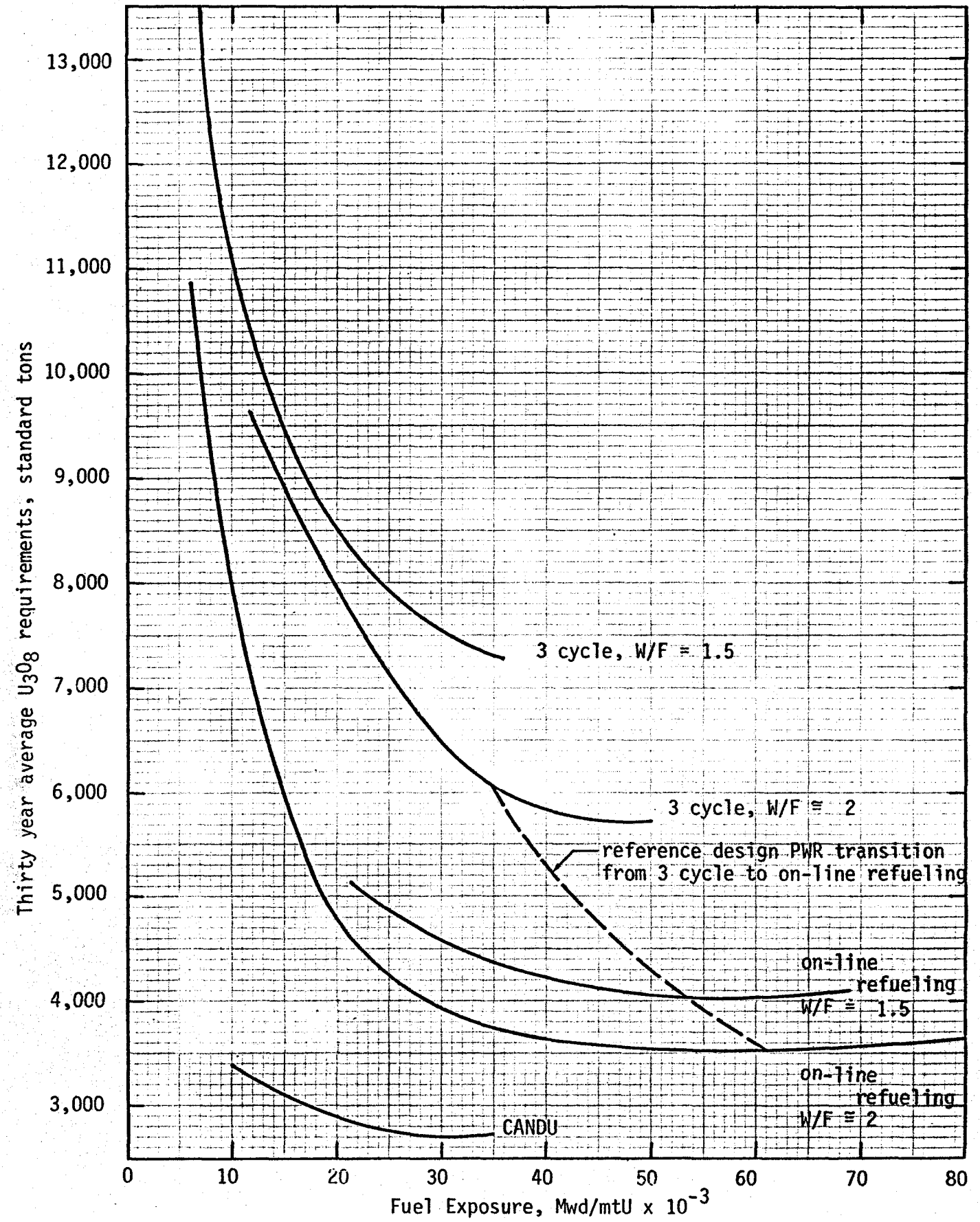

Fig. 2 Thirty year average resource requirements for various design fuel burnups.

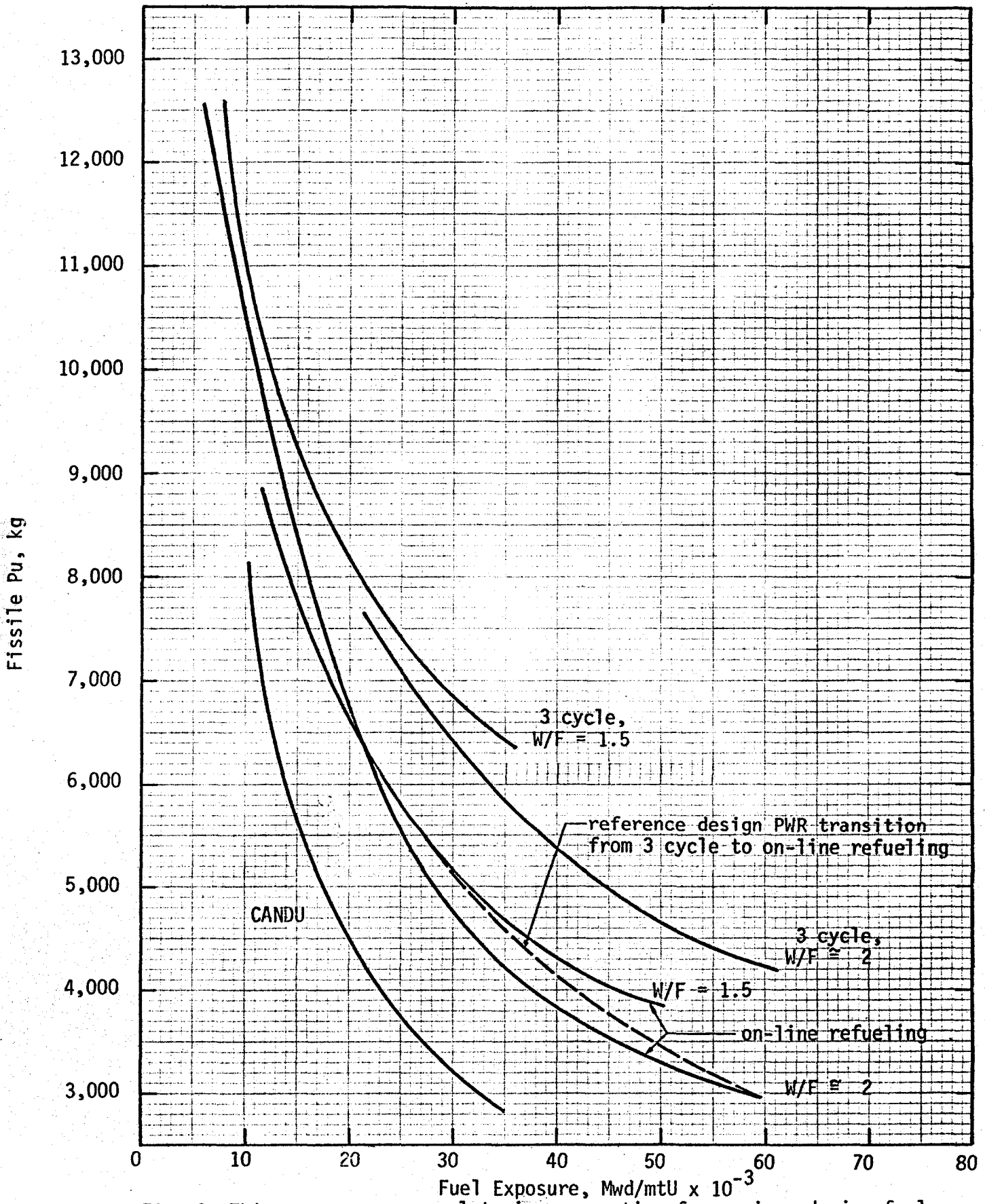

Fig. 3 Thirty year average plutonium production for various design fuel burnups.

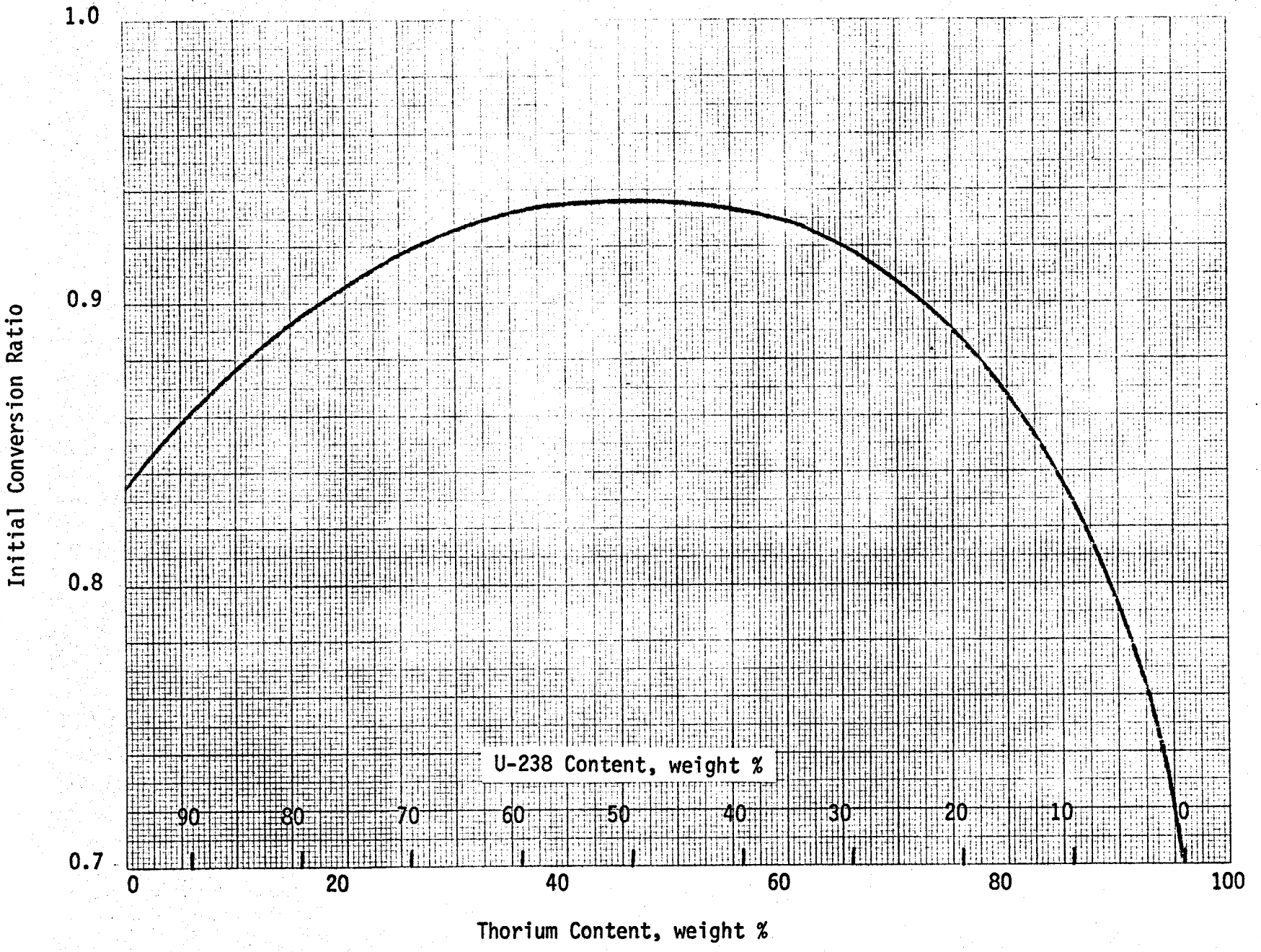

Fig. 4 Initial conversion ratio for HTGR-GASSAR fuel (weight % U-235 = 4.1) as a function of thorium content.

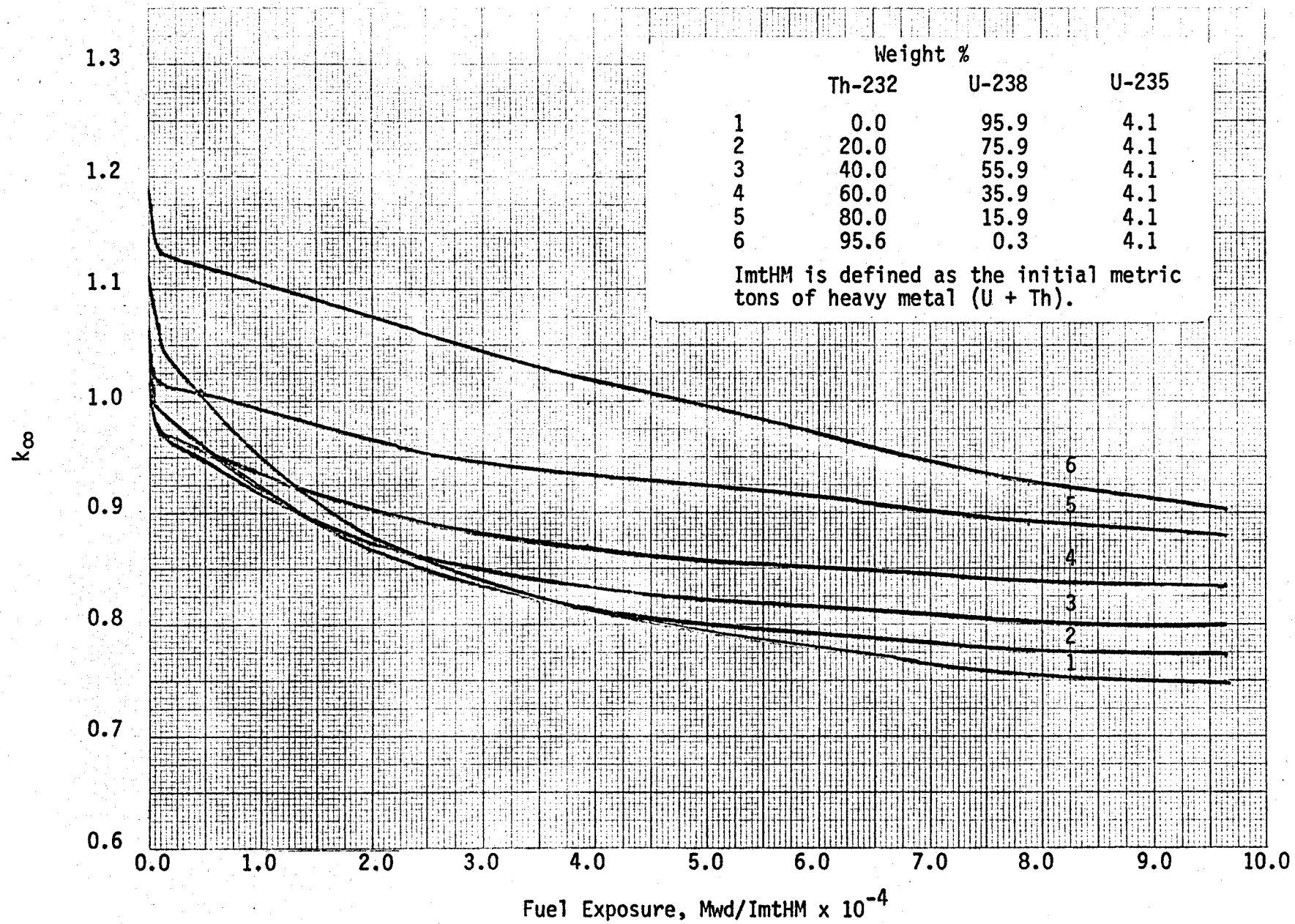

Fig. 5 $K\infty$ as a function of fuel exposure for HTGR-GASSAR irradiation of mixed uranium and thorium oxide fuels.

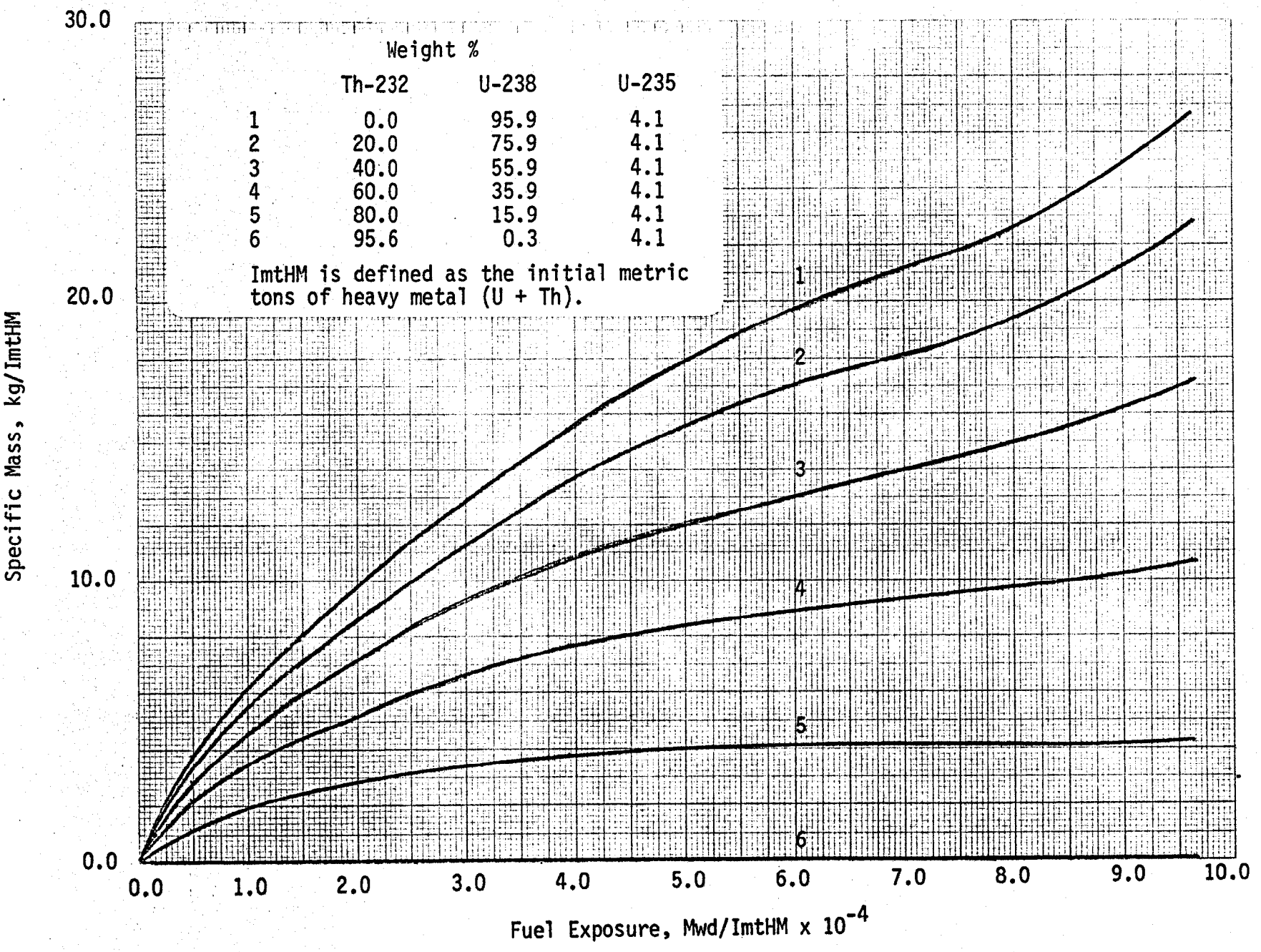

Fig. 6 Specific mass of fissile plutonium as a function of fuel exposure for HTGR-GASSAR irradiation of mixed uranium and thorium oxide fuels.

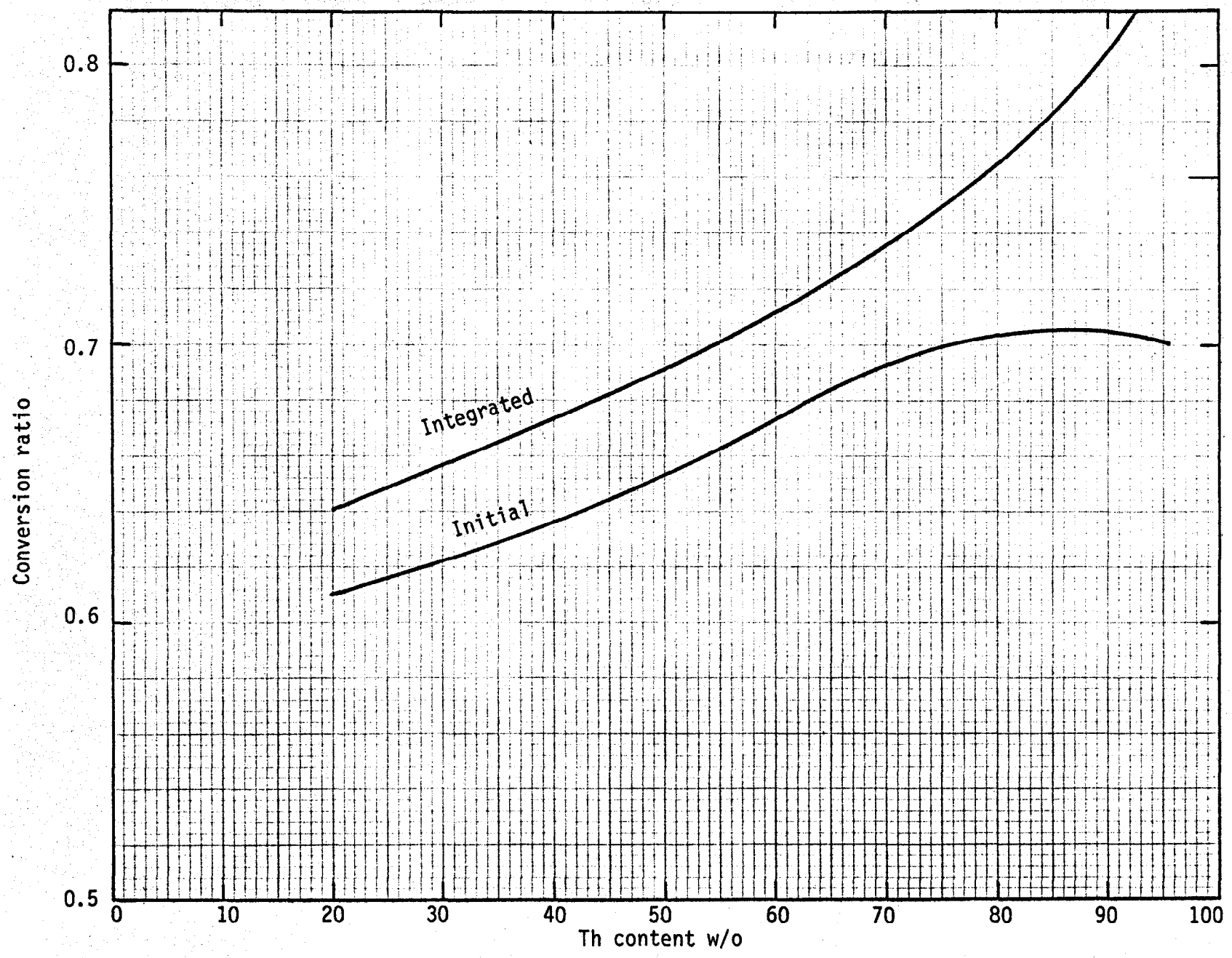

Fig. 7 Conversion ratio for HTGR-GASSAR irradiation of mixed uranium and thorium oxide fuels for reference cycle reactivity

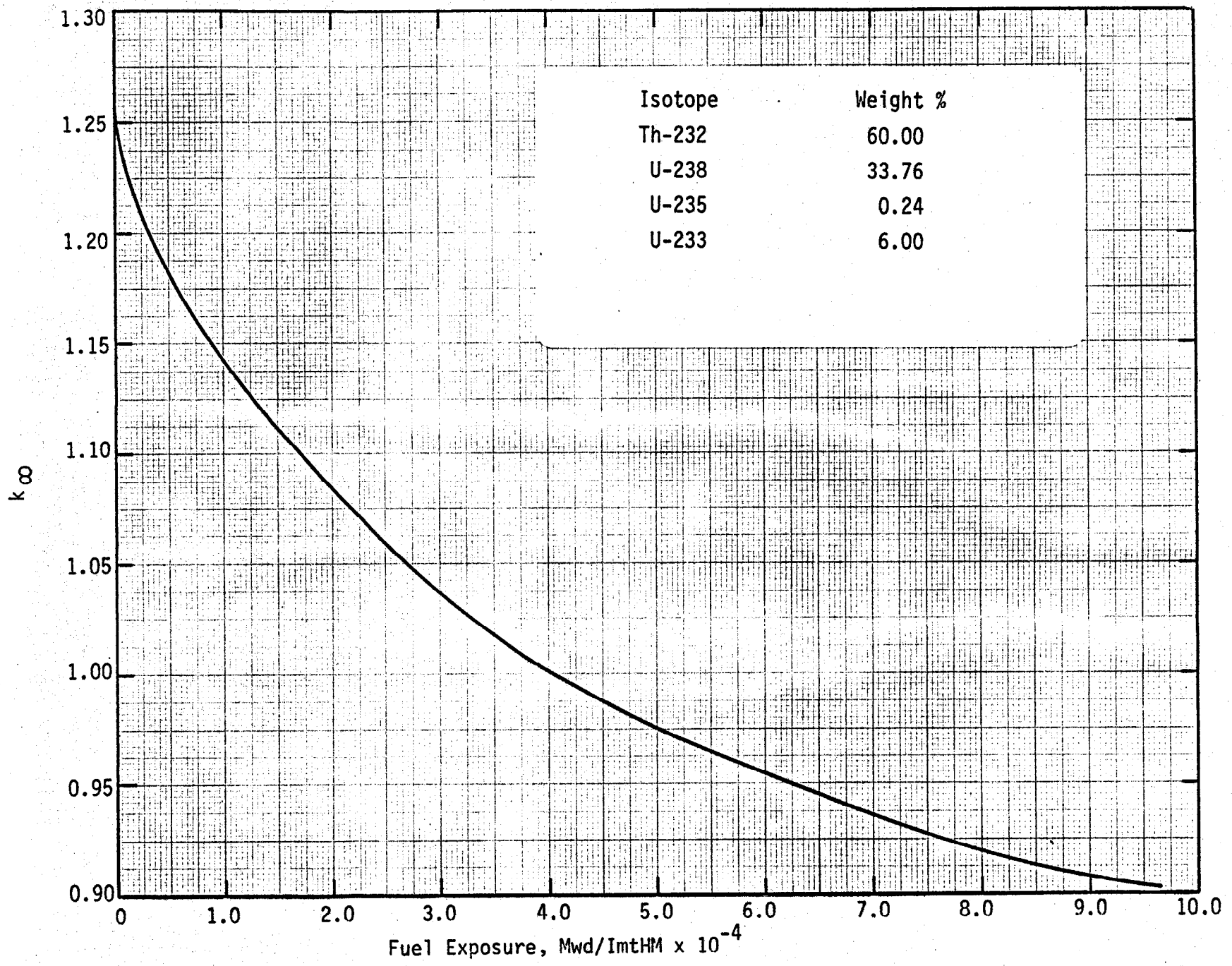

Fig. 8 K_∞ as a function of fuel exposure for HTGR-GASSAR irradiation of mixed U-233, natural uranium and thorium oxide fuel

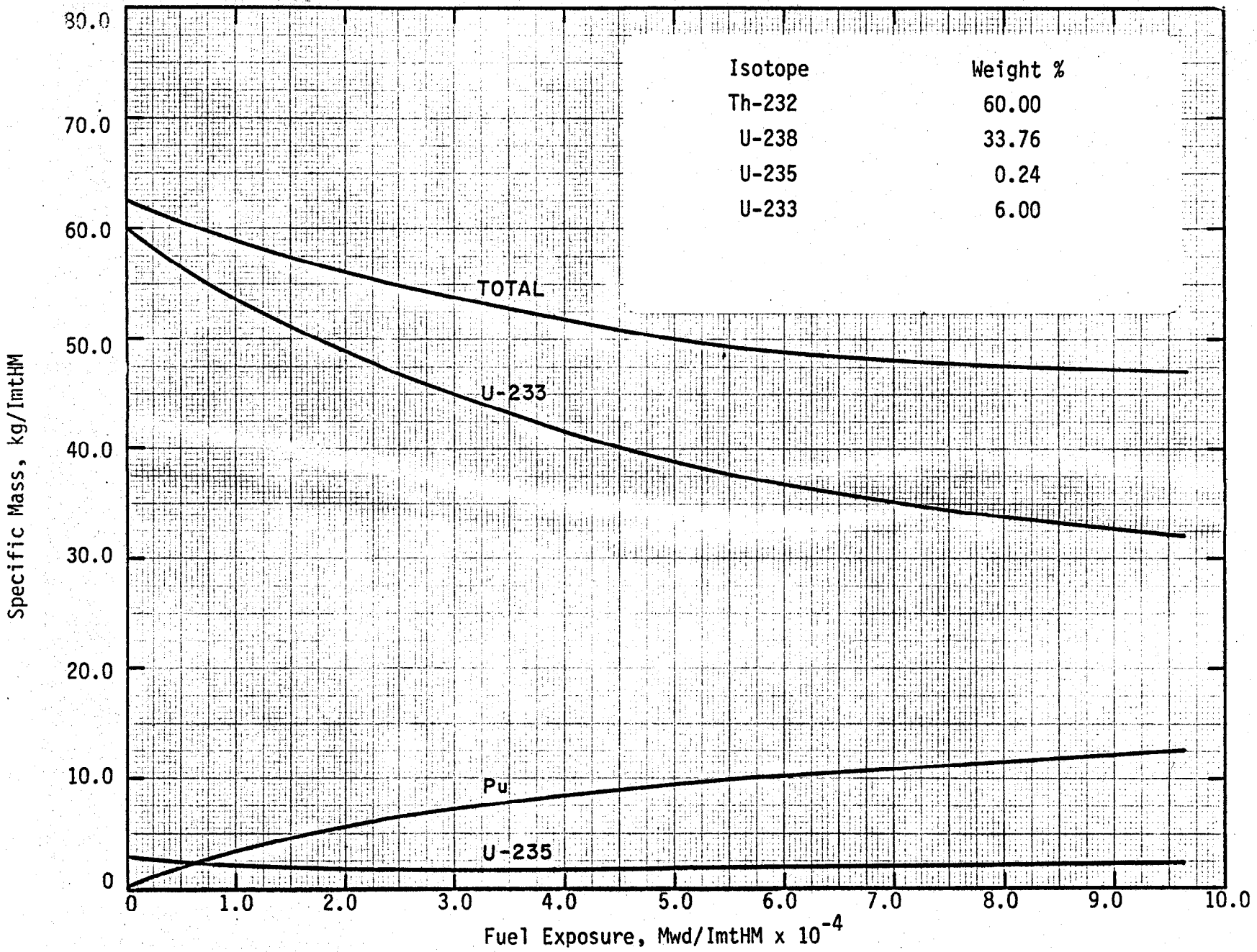

Fig. 9 Specific Mass of fissile material as a function of fuel exposure for HTGR-GASSAR irradiation of mixed U-233, natural uranium and thorium oxide fuel.

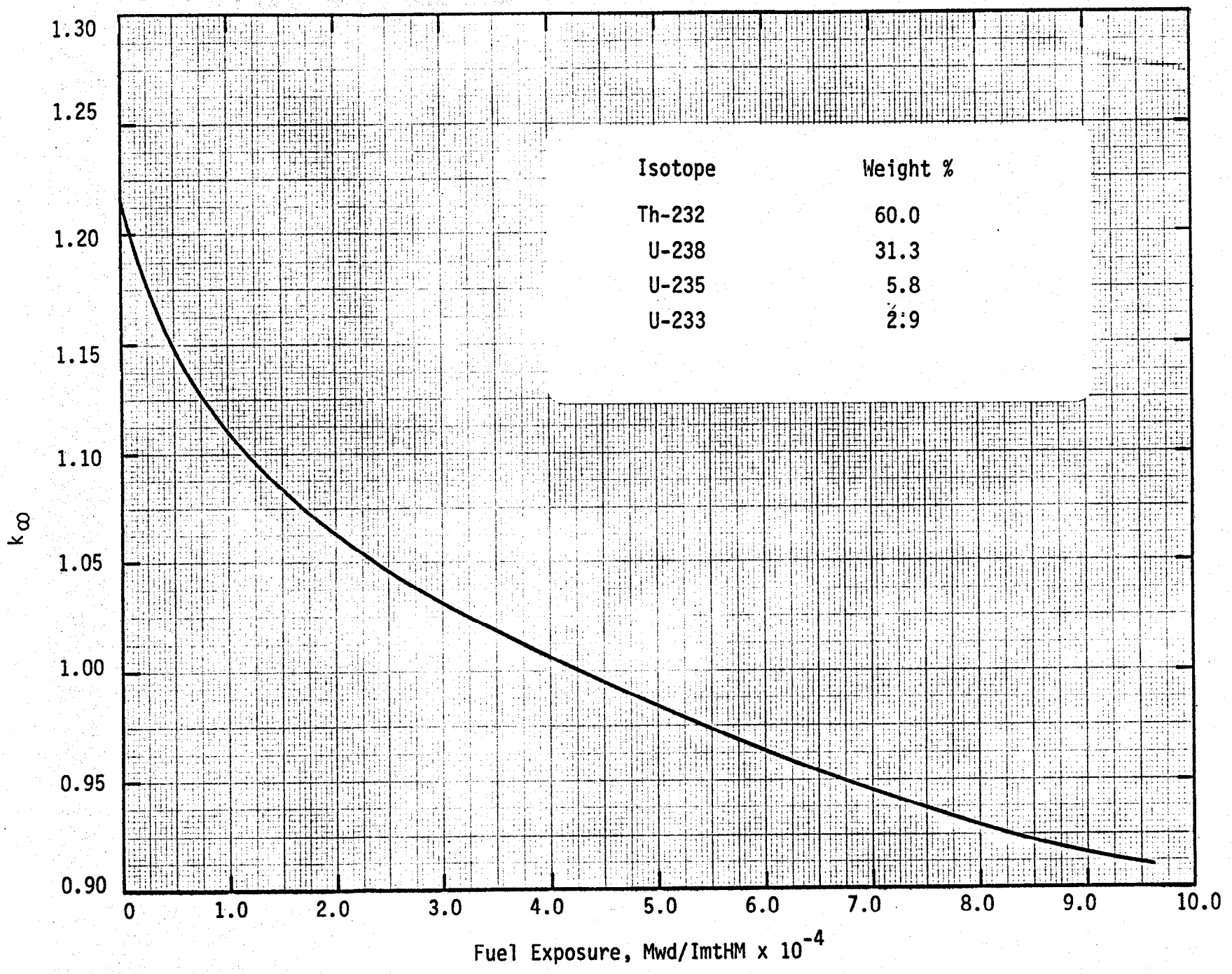

Fig. 10 K_∞ as a function of fuel exposure for HTGR-GASSAR irradiation of mixed U-233 enriched uranium and thorium oxide fuel.

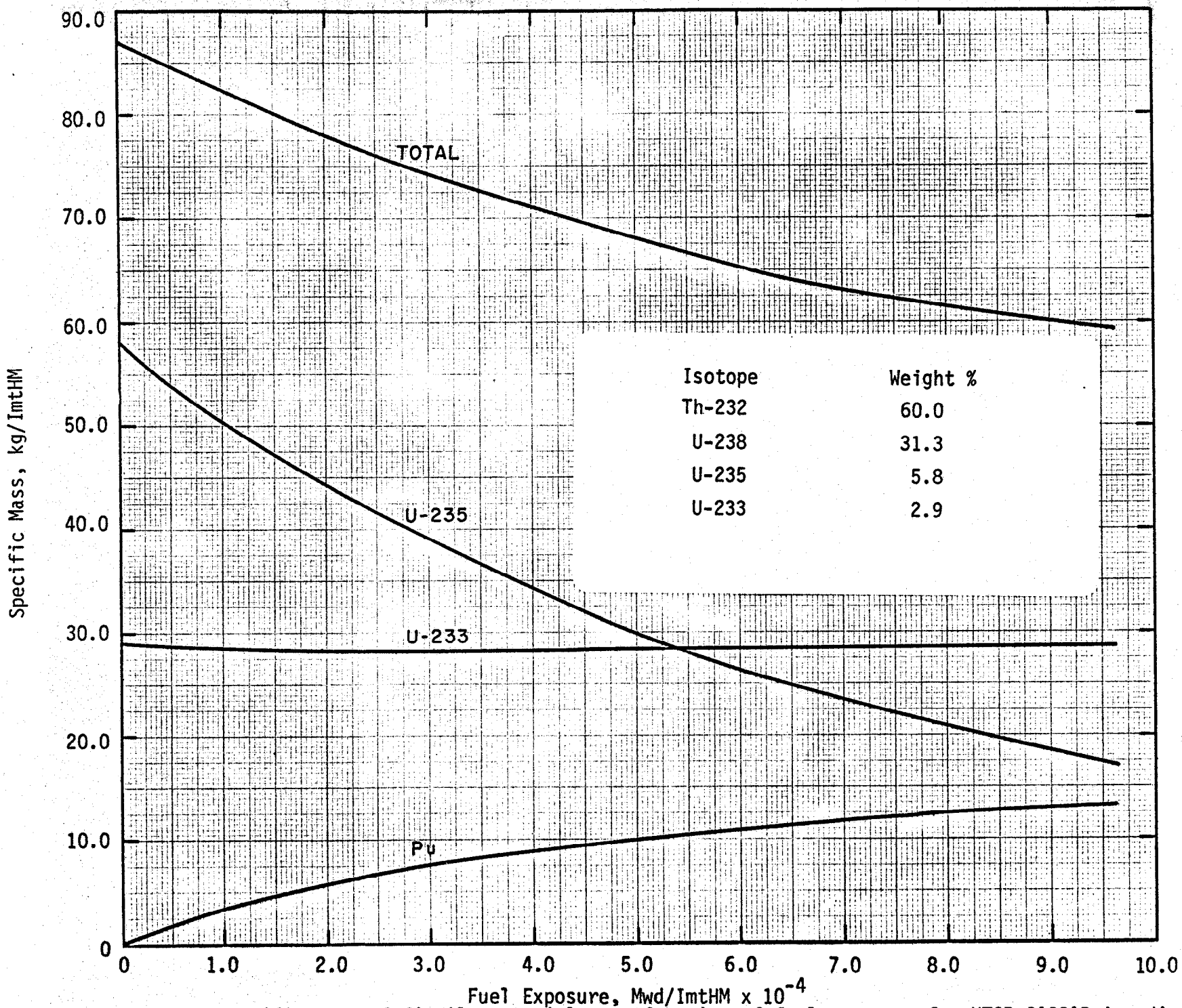

Fig. 11 Specific Mass of fissile material as a function of fuel exposure for HTGR-GASSAR irradiation of mixed U-233, enriched uranium and thorium oxide fuel for recycle use.

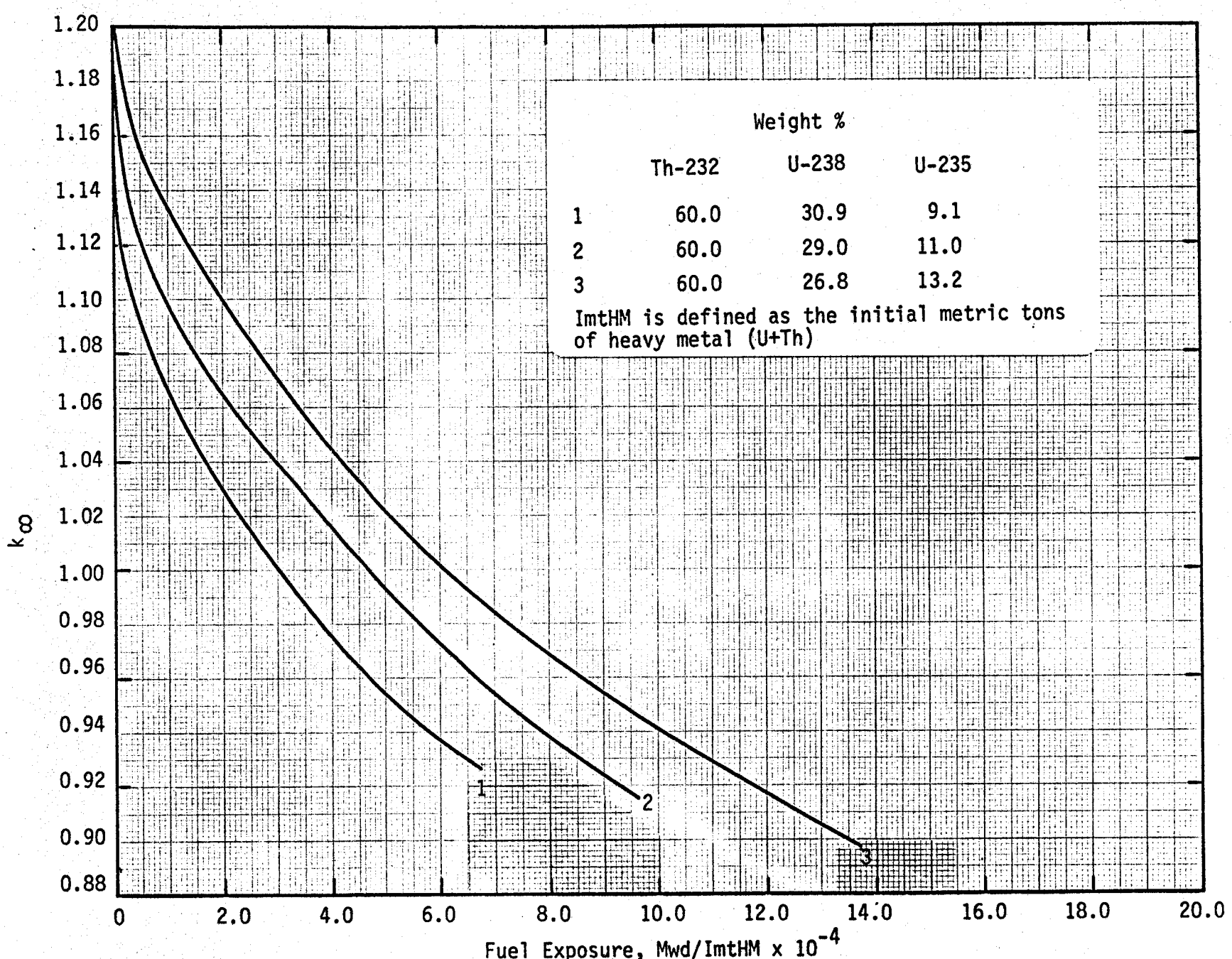

Fig. 12 K_∞ as a function of fuel exposure for HTGR-GASSAR irradiation of mixed uranium and thorium oxide fuels for a once-through fuel cycle.

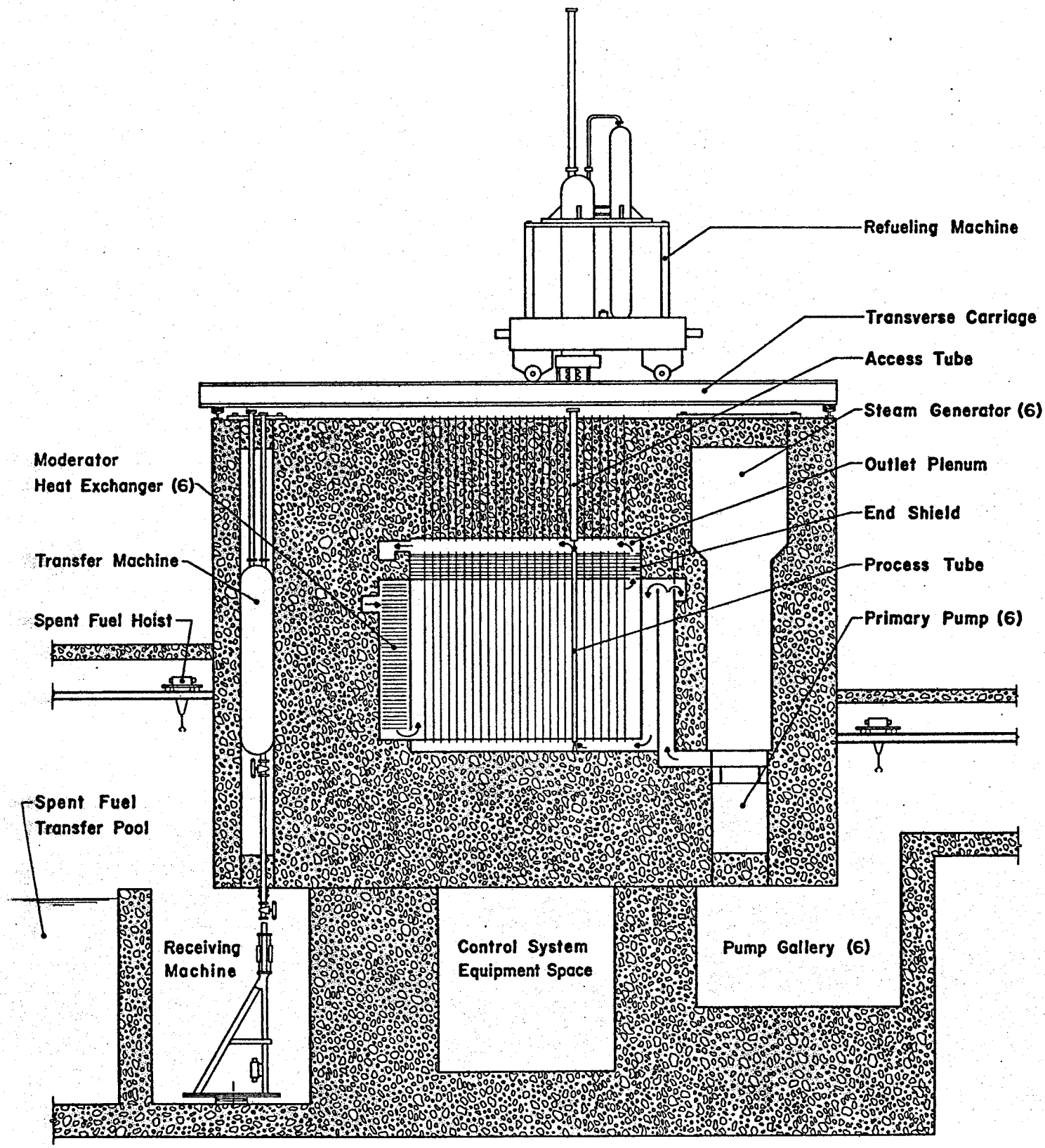

Fig. 13 Heavy water reactor vertical cross-section.

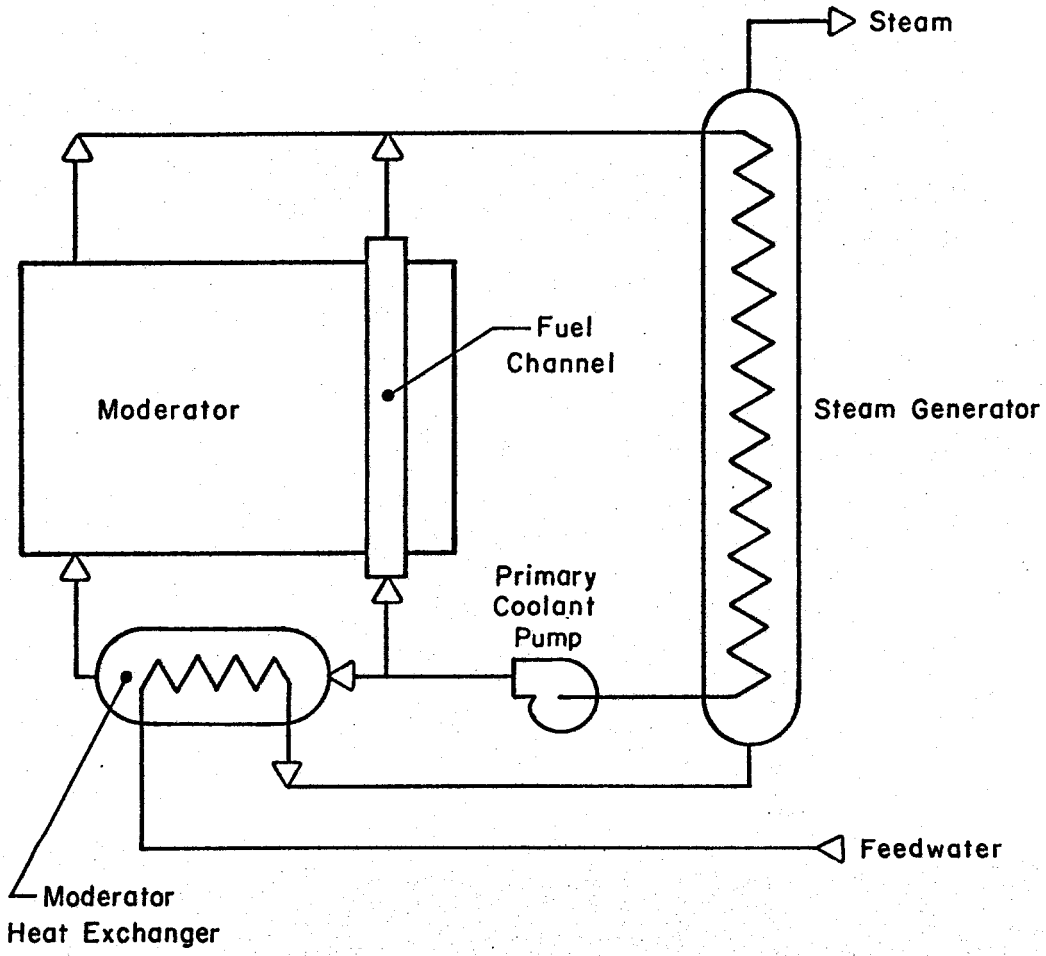

Fig. 14 Heavy water reactor flow diagram.