**Thorium Assessment Study
Quarterly Progress Report
for Second Quarter Fiscal 1977**

OAK RIDGE NATIONAL LABORATORY

OPERATED BY UNION CARBIDE CORPORATION FOR THE ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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THORIUM ASSESSMENT STUDY
QUARTERLY PROGRESS REPORT FOR SECOND QUARTER FISCAL 1977

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ABSTRACT

The objective of this program is to contribute to the ongoing assessment of the potential role of thorium fuel cycles for alleviating safeguards concerns. Scenarios include (1) no fuel recycle permitted, (2) fuel recycle permitted only in secure regions ("energy parks") with denatured (chemically non-separable) fuels only outside these regions, and (3) no limits on fuel recycle. A further objective is to provide nuclear mass balance data on HTGRs required by ERDA contractors for comparative cost-benefit studies.

1. HIGHLIGHTS

During the second quarter of FY '77, mass balances were calculated for the reference design HTGR and for a current design PWR operating on the denatured UO_2/ThO_2 fuel cycle with self-generated ^{233}U recycle. Comparisons were made of Pu production in PWR and CANDU reactors, and of ore and separative work requirements for several fuel cycles in PWR and HTGR reactors. The assembling, production, and verification of methods and data continued. Preliminary LMFBR equilibrium cycle mass balances were obtained for two denatured ^{233}U reactors for dispersed location and for two ^{233}U production reactors to be located in the energy center. Potential symbiotic systems coupling reactors inside and outside the energy center were developed.

2. THERMAL REACTOR CALCULATIONS

J. C. Cleveland
W. E. Thomas

2.1 Methods Development

Mass balance calculations for various PWR fuel cycles are being performed with the LEOPARD¹ code. This is used in conjunction with cross section libraries based on ENDF/B-IV developed by EPRI under their program for cross section standardization for thermal reactors.² Thus far, efforts to obtain the more detailed calculational capability which has recently been developed by EPRI in their Advanced Recycle Methodology Program³ have not been successful due to an agreement between EPRI, ERDA and NRC to limit distribution of the codes to three ERDA laboratories.

In order to evaluate the adequacy of the LEOPARD code to provide a calculational capability for urania and urania-thoria fuels, several UO_2 and UO_2/ThO_2 critical experiments have been analyzed. Data involving various UO_2 criticals covering a wide range of fuel enrichments, pellet sizes, soluble boron concentration and moderator-to-fuel volume ratios is presented in Ref. 4. Table 2.1 presents the values of k_{eff} for those criticals which have been analyzed. These results for UO_2 criticals were obtained by W. B. Arthur of ORGDP. The mean value of the calculated k_{eff} is 1.0011 with a standard deviation of 0.0125.

Unfortunately, data for UO_2/ThO_2 critical experiments is quite limited. Results of critical experiments involving UO_2/ThO_2 fuel performed by Babcock and Wilcox under their Thorium-Uranium Physics Experiments program are given in Ref. 5. The uranium component of the fuel was

Table 2.1. Data for UO_2 Critical Experiments and Calculated k_{eff}

Critical designation	Enrichment at %	$\text{H}_2\text{O}:\text{U}$ volume ratio	Pellet diameter (cm)	$\frac{N(^{10}\text{B})}{N(^{235}\text{U})}$	Calculated k_{eff}
24.1	2.734	2.18	0.762	0.0	0.9914
24.2	2.734	2.93	0.762	0.0	0.9958
24.3	2.734	3.86	0.762	0.0	0.9956
25.1	2.734	7.02	0.762	0.0	1.0001
25.2	2.734	8.49	0.762	0.0	1.0010
25.3	2.734	10.38	0.762	0.0	0.9954
26.1	2.734	2.50	0.762	0.0	0.9914
26.2	2.734	4.51	0.762	0.0	0.9905
26.3	3.745	2.50	0.7544	0.0	0.9805
26.4	3.745	4.51	0.7544	0.0	0.9802
27.1	3.745	4.51	0.7544	0.0	0.9901
27.2	3.745	4.51	0.7544	0.0128	0.9964
27.3	3.745	4.51	0.7544	0.0199	0.9894
27.4	3.745	4.51	0.7544	0.0354	0.9984
27.5	3.745	4.51	0.7544	0.0374	0.9992
27.6	3.745	4.51	0.7544	0.0414	1.0101
28.1	4.069	2.55	1.1278	0.0495	1.0113
28.10	4.069	2.14	1.1278	0.0	0.9938
29.1	3.037	2.64	1.1268	0.0	1.0153
29.2	3.037	8.16	1.1268	0.0	1.0117
29.3	4.069	2.59	1.1268	0.0	1.0196
29.4	4.069	3.53	1.1268	0.0	1.0183
29.5	4.069	8.02	1.1268	0.0	1.0292
29.6	4.069	9.90	1.1268	0.0	1.0124
mean					1.0012
Std. Dev.					0.0125

fully enriched ^{235}U . These criticals were fully reflected, unborated and covered a range of moderator-to-fuel volume fractions. The net fissile enrichment of the experiments analyzed here was 6.3 w/o*. In the LEOPARD analysis, the leakage was accounted for by inputting a total buckling determined from the measured core dimensions and the radial and axial reflector savings.

The calculated values of k_{eff} for the UO_2/ThO_2 criticals are given in Table 2.2. Note that there is a trend in calculated k_{eff} with the moderator-to-fuel volume fraction. The value of this ratio for the CE System 80 fuel assembly is 1.71, and the calculated results suggest that good agreement with criticality would be expected at this moderator-to-fuel ratio.

Results of exponential experiments performed at BNL which involved fuel of 3.0 w/o $^{233}\text{UO}_2$ in ThO_2 are reported by Windsor, et al. in Ref. 6. If time permits, these experiments will be analyzed to obtain a criticality benchmark for ^{233}U -containing systems. The good agreement with experiment for the analysis of the various UO_2 criticals and the B&W UO_2/ThO_2 criticals over the range of parameters involved provides confidence in the ability of the methods employed to perform criticality calculations.

It would be very useful to also have a comparison of calculated results and actual measurements of various isotopic concentrations for irradiated thorium fuel. The possibility of obtaining such a comparison has been examined. A post-irradiation examination (PIE) of thorium-uranium fuel was performed by B&W on fuel rods discharged from Core A of the Indian Point Reactor. Information concerning the experimentally measured isotopics of this fuel is presented in Ref. 7.

*Weight percent

Table 2.2. Data for B & W Urania-Thoria Critical Experiments and Calculated k_{eff}

Critical designation	H ₂ O: fuel volume ratio	Calculated k_{eff}
15 A	1.380	0.9868
15 B	1.642	0.9913
15 D	2.945	1.0111

In order to obtain an acceptable fuel assembly power distribution, the Indian Point Core A assemblies were loaded with four, and sometimes five, enrichments with the pins nearest the water gaps having the lowest enrichment. The pins located inside the outer two rows of pins in each assembly had the highest enrichment. Unfortunately, the pins chosen for the PIE were in the outer row of the assembly since these were the pins which could be most easily removed. Since these pins were located adjacent to a water gap and neighboring pins were of a different enrichment, the spectrum at these outer locations was not typical of that represented by a model which assumes an array of identical pins--as does the LEOPARD point depletion code. For this reason, these experiments have not been analyzed.

2.2 Mass Balances for a PWR Operating on the Denatured UO_2/ThO_2 Fuel Cycle

Mass balances have been calculated for a current design PWR operating on the denatured UO_2/ThO_2 fuel cycle with self-generated recycle of uranium. The fuel cycle considered is the stand alone cycle--i.e., no exogenous source of ^{233}U is assumed. The initial core calculations were performed by Combustion Engineering. The mass balances for reload fuel batches were determined by ORNL. Reactor specifications are given in Table 2.3.

The CE System 80TM fuel assembly and core design were used for this study. The fuel cycle scheme involved the self-generated recycle of uranium initially bred in UO_2/ThO_2 fuel assemblies in which the fissile uranium content is denatured ^{235}U (20% enriched). The bred plutonium is assumed to be separated from spent fuel but not recycled into this reactor. Recycle of bred uranium continues throughout the history of this reactor. Bred uranium is re-enriched by addition of fresh, fully enriched UO_2 to an allowable limit for denatured fuel (discussed below) involving both ^{233}U and ^{235}U . Denatured (20% enriched) $^{235}\text{U}/\text{ThO}_2$ fuel is utilized for the remainder of the fresh makeup requirements and is kept segregated from the re-enriched bred uranium in separate fuel assemblies. This segregation reduces the throughput requirements for the remote refabrication facility since the 20% enriched $^{235}\text{U}/\text{ThO}_2$ fuel does not require remote fabrication.

It is currently believed that an isotopic composition of 12% ^{233}U and 88% ^{238}U (or 20% ^{235}U and 80% ^{238}U) constitutes adequate denaturing of uranium. The recycle fuel contains both ^{233}U and ^{235}U . Therefore, in this study it was assumed that adequate denaturing of re-enriched bred

fuel would be obtained if the fissile fraction of this recycle fuel is not greater than the weighted average of 0.12 and 0.20, the weighting factors being the masses of ^{233}U and ^{235}U respectively. Fully enriched uranium was added to the bred fuel in order to raise its fissile fraction to this limit. Fully enriched uranium was chosen rather than uranium of lower enrichment (e.g., 20%) in order to maintain denaturing with a minimum of ^{238}U in the core. This minimizes plutonium production as well as the reactivity penalty which occurs when both fertile isotopes, ^{238}U and ^{232}Th , are present in the fuel.

As mentioned above, this fuel cycle scheme assumed no source of ^{233}U such as a fast reactor thorium blanket. This scheme provides a means of breeding and recycling ^{233}U in reactors outside of secured areas. The fact that the fissile component of the fuel in this fuel scheme is denatured provides a safeguards advantage over the conventional LWR plutonium recycle concept.

The fuel management scheme involves reloading one-third of the core annually. Each fuel batch--except for two of the three batches loaded in the initial core--resides in the reactor for three years. It is assumed that two years are required for reprocessing and refabrication of the bred fuel. Thus the bred fuel returns to the reactor at the beginning of the third cycle after it is discharged. Material losses of 1.5% are assumed for the reprocessing and refabrication processes.

The reactor fuel initial inventory data by fuel batch is presented in Table 2.4. The uranium makeup requirements are broken down into two parts--fully enriched makeup uranium used to re-enrich the bred fuel which is loaded into the batches designated as recycle batches and 20%

Table 2.3. Summary of Core Parameters* for Denatured
PWR with Self-Generated Uranium Recycle

Core Heat Output	3800 MW(t)
Electrical Output	1300 MW(e)
Primary System Pressure	2250 psia
Fuel Management	3 Batch, Mixed Central Zone
Number of Fuel Assemblies	241
Number of Control Element Assemblies (CEA)	89
Fuel Rod Array, Square	16 × 16
Fuel Rod Pitch	1.286 cm
Clad, Spacer Grid and CEA Material	Zircaloy 4
Pellet Density (% theoretical)	95.0
Stack Height Density (% theoretical)	91.7
Clad OR	0.485 cm
Pellet OR	0.413 cm
Spacing Between Fuel Assemblies (rod to rod)	0.516 cm
Annual Fuel Loading (Equilibrium Denatured Cycle)	31987 kg (U + Th)
Average Cycle Burnup (Equilibrium Denatured Cycle)	32521 MWD/MTHM
Capacity Factor	0.75
Core Conversion Ratio (Equilibrium Cycle)	0.67

*The design is that of the Combustion Engineering System 80TM
reactor.

Table 2.4. Reactor Fuel Initial Inventory Data

Batch ID*	Cycle number		Heavy metal loaded (kg)	Thorium charged (kg)	Fully enriched makeup uranium used to re-enrich bred fuel			Uranium charged [†] (kg)				
	Charged	Discharged			U-234	U-235	U-238	U-233	U-234	U-235	U-236	U-238
1-A	1	1	31181.0	27030.5	—	—	—	—	6.6	829.3	—	3314.6
1-B	1	2	31181.0	25459.5	—	—	—	—	9.1	1144.4	—	4568.0
1-C	1	3	31181.0	24554.3	—	—	—	—	10.6	1325.3	—	5290.8
2	2	4	31892.5	24876.2	—	—	—	—	11.2	1403.3	—	5601.9
3	3	5	31892.5	24876.2	—	—	—	—	11.2	1403.3	—	5601.9
4-R	4	6	16939.6	12962.7	0.2	14.0	0.9	197.9	12.5	509.3	56.2	3201.0
4-M	4	6	14978.0	11682.8	—	—	—	—	5.2	659.0	—	2630.9
5-R	5	7	22858.5	17421.2	2.4	194.3	12.2	294.2	31.6	659.0	116.4	4336.1
5-M	5	7	9066.8	7072.1	—	—	—	—	3.2	398.9	—	1592.6
6-R	6	8	26388.5	20127.6	4.2	338.7	21.3	333.5	42.9	766.9	150.8	4966.8
6-M	6	8	5541.3	4322.2	—	—	—	—	1.9	243.8	—	973.3
7-R	7	9	28308.4	21654.5	4.4	356.1	22.4	333.3	41.7	847.3	156.3	5275.4
7-M	7	9	3620.4	2823.9	—	—	—	—	1.3	159.3	—	635.9
8-R	8	10	28308.4	21654.5	4.4	356.1	22.4	333.3	41.7	847.3	156.3	5275.4
8-M	8	10	3620.4	2823.9	—	—	—	—	1.3	159.3	—	635.9
9-R	9	11	28718.5	21794.0	5.0	397.6	25.0	389.1	52.9	817.1	169.6	5495.9
9-M	9	11	3237.2	2525.0	—	—	—	—	1.1	142.4	—	568.6
10-R	10	12	29139.0	22041.1	5.3	420.7	26.4	421.9	71.0	801.9	213.5	5589.5
10-M	10	12	2817.6	2197.7	—	—	—	—	1.0	124.0	—	494.9
11-R	11	13	29123.8	21980.0	5.3	423.9	26.6	440.0	82.2	783.2	239.2	5599.2
11-M	11	13	2833.0	2209.7	—	—	—	—	1.0	124.7	—	497.6
12-R	12	14	29122.2	22008.2	5.3	425.7	26.8	429.0	80.6	794.1	240.0	5570.3
12-M	12	14	2834.6	2210.9	—	—	—	—	1.0	124.7	—	497.9
13-R	13	15	29122.2	22008.2	5.3	425.7	26.8	429.0	80.6	794.1	240.0	5570.3
13-M	13	15	2834.6	2210.9	—	—	—	—	1.0	124.7	—	497.9
14-R	14	16	29443.7	22130.4	5.6	449.3	28.2	446.8	90.6	807.5	245.6	5722.8
14-M	14	16	2527.0	1971.1	—	—	—	—	0.9	111.2	—	443.9
15-R	15	17	29660.0	22274.6	5.6	451.3	28.4	459.4	103.1	805.0	275.8	5742.1
15-M	15	17	2296.0	1790.9	—	—	—	—	0.8	101.0	—	403.3

*R designates recycle fuel batch; M designates denatured (20% enriched U-235) makeup fuel batch.

[†]For the recycle fuel this includes the fully enriched makeup uranium used for re-enriching bred fuel.

Table 2.4. Continued

Batch ID*	Cycle number		Heavy metal loaded (kg)	Thcrium charged (kg)	Fully enriched makeup uranium used to re-enrich bred fuel			Uranium charged [†] (kg)				
	Charged	Discharged			U-234	U-235	U-238	U-233	U-234	U-235	U-236	U-238
16-R	16	18	29675.3	22274.6	5.6	451.3	28.4	460.8	110.4	803.6	299.6	5726.3
16-M	16	18	2296.0	1790.9	—	—	—	—	0.8	101.0	—	403.3
17-R	17	19	29675.3	22274.6	5.6	451.3	28.4	460.8	110.4	803.6	299.6	5726.3
17-M	17	19	2296.0	1790.9	—	—	—	—	0.8	101.0	—	403.3
18-R	18	20	29675.3	22274.6	5.6	451.3	28.4	460.8	110.4	803.6	299.6	5726.3
18-M	18	20	2296.0	1790.9	—	—	—	—	0.8	101.0	—	403.3
19-R	19	21	29513.2	21991.5	5.7	456.7	28.6	465.5	117.7	821.3	295.4	5822.4
19-M	19	21	2472.5	1928.6	—	—	—	—	0.9	108.8	—	434.3
20-R	20	22	29584.0	22028.8	5.6	458.2	28.9	472.1	131.3	817.7	347.7	5786.3
20-M	20	22	2402.7	1874.1	—	—	—	—	0.8	105.7	—	422.0
21-R	21	23	29584.0	22028.8	5.6	458.2	28.9	472.1	131.3	817.7	347.7	5786.3
21-M	21	23	2402.7	1874.1	—	—	—	—	0.8	105.7	—	422.0
22-R	22	24	29584.0	22028.8	5.6	458.2	28.9	472.1	131.3	817.7	347.7	5786.3
22-M	22	24	2402.7	1874.1	—	—	—	—	0.8	105.7	—	422.0
23-R	23	25	29584.0	22028.8	5.6	458.2	28.9	472.1	131.3	817.7	347.7	5786.3
23-M	23	25	2402.7	1874.1	—	—	—	—	0.8	105.7	—	422.0
24-R	24	26	29853.7	22144.2	5.8	465.0	29.2	477.4	132.5	845.0	343.8	5910.8
24-M	24	26	2144.3	1672.5	—	—	—	—	0.8	94.3	—	376.6
25-R	25	27	29891.1	22170.9	5.8	466.7	29.4	476.8	145.8	844.4	390.0	5863.3
25-M	25	27	2104.0	1641.1	—	—	—	—	0.7	92.6	—	369.6
26-R	26	28	29891.1	22170.9	5.8	466.7	29.4	476.8	145.8	844.4	390.0	5863.3
26-M	26	28	2104.0	1641.1	—	—	—	—	0.7	92.6	—	369.6
27-R	27	29	29891.1	22170.9	5.8	466.7	29.4	476.8	145.8	844.4	390.0	5863.3
27-M	27	29	2104.0	1641.1	—	—	—	—	0.7	92.6	—	369.6
28-R	28	30	29891.1	22170.9	5.8	466.7	29.4	476.8	145.8	844.4	390.0	5863.3
28-M	28	30	2104.0	1641.1	—	—	—	—	0.7	92.6	—	369.6
29-R	29	31	30267.4	22455.0	5.8	468.4	29.4	477.3	146.3	860.5	385.4	5932.4
29-M	29	31	1723.6	1348.4	—	—	—	—	0.5	76.1	—	303.6
30-R	30	32	30269.0	22450.7	5.8	468.5	29.4	479.2	155.3	858.7	427.3	5887.3
30-M	30	32	1725.8	1346.9	—	—	—	—	0.5	76.0	—	303.3

*R designates recycle fuel batch; M designates denatured (20% enriched U-235) makeup fuel batch.

[†]For the recycle fuel this includes the fully enriched makeup uranium used for re-enriching bred fuel.

enriched ^{235}U which is loaded into batches designated as denatured ^{235}U makeup. This breakdown is necessary in order that separative work requirements and fabrication costs can be determined. The denatured ^{235}U makeup fuel would not require remote fabrication. Table 2.5 provides the fuel discharge data by batch.

Mass balance information for this scheme is presented in Tables 2.6 through 2.9 in a format similar to that used by CE in their recent Thorium Assessment Study in order to facilitate comparison. Table 2.6 summarizes the mass balance data for each recycle generation. The isotopic masses in Table 2.6 represent the sum of re-enriched recycle fuel and the fresh denatured ^{233}U makeup fuel which is loaded at the same time as the recycle fuel. Each generation after the zeroth generation spans five years of reactor operation. The zeroth generation represents the denatured $^{235}\text{U}/\text{ThO}_2$ loadings from which the ^{233}U is initially bred. Bred fuel recovered from the discharged zeroth generation fuel is re-enriched and reloaded into the reactor as first generation recycle fuel along with 20% enriched $^{235}\text{U}/\text{ThO}_2$ fresh makeup fuel. Second generation recycle fuel is composed of re-enriched fuel discharged both from first generation recycle assemblies and from denatured $^{235}\text{U}/\text{ThO}_2$ makeup assemblies which are loaded into the reactor at the same time as the first generation recycle assemblies. Each succeeding generation is defined similarly.

Table 2.7 presents the makeup requirements. These requirements are broken down into the requirement for fully enriched uranium which is used to re-enrich bred fuel and the requirements for makeup denatured $^{235}\text{U}/\text{ThO}_2$ fuel.

Table 2.5. Reactor Fuel Discharge Data

Batch ID	Thorium discharged (kg)	Uranium discharged					Plutonium discharged				Average Exposure (MWD/MTHM)
		U-233*	U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	
1-A	26776.3	200.9	12.5	502.8	57.0	3248.8	28.6	5.9	3.2	0.4	10290
1-B	24980.0	298.7	29.6	471.8	118.2	4389.7	57.4	17.2	12.2	2.6	24124
1-C	23941.4	338.6	39.3	434.7	153.1	5020.8	70.2	23.4	18.4	4.7	33876
2	24282.6	338.4	37.6	498.8	158.5	5333.1	73.4	23.9	18.5	4.8	32614
3	24282.6	338.4	37.6	498.8	158.4	5333.1	73.4	23.9	18.5	4.8	32614
4-R	12657.3	236.1	30.8	191.6	97.7	3049.6	43.0	13.7	10.3	2.5	32572
4-M	11404.1	158.9	17.8	234.3	74.4	2504.6	34.5	11.2	8.7	2.3	32614
5-R	17009.0	332.1	56.0	245.3	171.7	4131.7	58.3	18.7	14.2	3.6	32572
5-M	6903.3	96.2	10.8	141.8	45.1	1516.1	20.9	6.8	5.3	1.4	32614
6-R	19651.8	337.9	71.5	278.1	215.3	4730.9	67.0	21.4	16.1	4.0	32572
6-M	4219.1	58.8	6.6	86.7	27.5	926.6	12.7	4.2	3.2	0.8	32614
7-R	21144.1	397.2	72.2	317.3	225.6	5022.5	70.8	22.9	17.3	4.3	32572
7-M	2756.5	38.4	4.3	56.6	18.0	605.4	8.3	2.7	2.1	0.5	32614
8-R	21144.1	397.2	72.2	317.3	225.6	5022.5	70.8	22.9	17.3	4.3	32572
8-M	2756.5	38.4	4.3	56.6	18.0	605.4	8.3	2.7	2.1	0.5	32614
9-R	21283.3	419.3	82.4	313.0	233.2	5240.0	73.5	23.3	17.2	4.3	32546
9-M	2464.8	34.3	3.9	50.6	16.1	541.3	7.4	2.4	1.9	0.5	32614
10-R	21526.4	436.5	95.6	315.0	266.0	5329.5	75.5	23.6	17.5	4.4	32546
10-M	2145.3	29.9	3.4	44.1	14.0	471.2	6.5	2.1	1.6	0.4	32614
11-R	21466.0	437.7	103.1	313.4	290.1	5339.6	75.8	23.7	17.6	4.4	32546
11-M	2157.0	30.1	3.4	44.3	14.1	474.0	6.5	2.1	1.6	0.4	32614
12-R	21493.9	437.7	103.1	313.4	290.1	5310.7	75.4	23.6	17.5	4.4	32546
12-M	2158.2	30.1	3.4	44.3	14.1	474.0	6.5	2.1	1.6	0.4	32614
13-R	21493.9	437.7	103.1	313.4	290.1	5310.7	75.4	23.6	17.5	4.4	32546
13-M	2158.2	30.1	3.4	44.3	14.1	474.0	6.5	2.1	1.6	0.4	32614
14-R	21618.4	445.8	110.7	330.7	287.4	5459.5	78.3	24.4	18.0	4.4	32531
14-M	1924.0	26.8	3.0	39.5	12.6	422.6	5.8	1.9	1.5	0.4	32614
15-R	21758.5	454.9	124.9	329.1	341.6	5461.2	78.3	24.3	18.1	4.2	32531
15-M	1748.1	24.4	2.7	35.9	11.4	383.9	5.3	1.7	1.3	0.3	32614

*Includes Pa-233.

Table 2.5. Continued

Batch ID	Thorium discharged (kg)	Uranium discharged					Plutonium discharged				Average Exposure (MWD/MTHM)
		U-233*	U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	
16-R	21758.5	454.9	124.9	329.1	341.6	5461.2	78.3	24.3	18.1	4.2	32531
16-M	1748.1	24.4	2.7	35.9	11.4	383.9	5.3	1.7	1.3	0.3	32614
17-R	21758.5	454.9	124.9	329.1	341.6	5461.2	78.3	24.3	18.1	4.2	32531
17-M	1748.1	24.4	2.7	35.9	11.4	383.9	5.3	1.7	1.3	0.3	32614
18-R	21758.5	454.9	124.9	329.1	341.6	5461.2	78.3	24.3	18.1	4.2	32531
18-M	1748.1	24.4	2.7	35.9	11.4	383.9	5.3	1.7	1.3	0.3	32614
19-R	21488.7	455.4	125.7	347.1	336.8	5557.7	80.6	24.5	18.0	4.1	32514
19-M	1882.5	26.2	2.9	38.7	12.3	413.5	5.7	1.9	1.4	0.4	32614
20-R	21524.7	458.5	139.3	345.8	384.0	5521.0	80.2	24.6	18.0	4.1	32514
20-M	1829.4	25.5	2.9	37.6	11.9	401.8	5.5	1.8	1.4	0.4	32614
21-R	21524.7	458.5	139.3	345.8	384.0	5521.0	80.2	24.6	18.0	4.1	32514
21-M	1829.4	25.5	2.9	37.6	11.9	401.8	5.5	1.8	1.4	0.4	32614
22-R	21524.7	458.5	139.3	345.8	384.0	5521.0	80.2	24.6	18.0	4.1	32514
22-M	1829.4	25.5	2.9	37.6	11.9	401.8	5.5	1.8	1.4	0.4	32614
23-R	21524.7	458.5	139.3	345.8	384.0	5521.0	80.2	24.6	18.0	4.1	32514
23-M	1829.4	25.5	2.9	37.6	11.9	401.8	5.5	1.8	1.4	0.4	32614
24-R	21640.3	461.8	140.6	364.5	380.6	5634.3	82.7	24.8	18.2	4.2	32506
24-M	1632.6	22.8	2.6	33.5	10.7	358.6	4.9	1.6	1.2	0.3	32614
25-R	21666.3	464.2	149.8	363.2	423.3	5595.3	82.2	24.8	18.2	4.2	32506
25-M	1601.9	22.3	2.5	32.9	10.5	351.8	4.8	1.6	1.2	0.3	32614
26-R	21666.3	464.2	149.8	363.2	423.3	5595.3	82.2	24.8	18.2	4.2	32506
26-M	1601.9	22.3	2.5	32.9	10.5	351.8	4.8	1.6	1.2	0.3	32614
27-R	21666.3	464.2	149.8	363.2	423.3	5595.3	82.2	24.8	18.2	4.2	32506
27-M	1601.9	22.3	2.5	32.9	10.5	351.8	4.8	1.6	1.2	0.3	32614
28-R	21666.3	464.2	149.8	363.2	423.3	5595.3	82.2	24.8	18.2	4.2	32506
28-M	1601.9	22.3	2.5	32.9	10.5	351.8	4.8	1.6	1.2	0.3	32614
29-R	21954.2	464.6	150.7	373.2	420.4	5661.2	83.2	25.1	18.5	4.2	32506
29-M	1316.2	18.3	2.1	27.0	8.6	289.1	4.0	1.3	1.0	0.3	32614
30-R	21949.9	466.5	159.5	371.4	457.3	5616.1	83.2	25.1	18.5	4.2	32506
30-M	1314.8	18.3	2.1	27.0	8.6	288.8	4.0	1.3	1.0	0.3	32614

*Includes Pa-233.

Table 2.6. Reload Batch Inventories for Denatured
Fuel Cycles (recycle + makeup) 1300 MW(e)
PWR, 75% Capacity Factor

		Generation					
		0*	1	2	3	4	5
Initial Inventories (kg)							
U233			333.3	429.0	460.8	472.1	476.8
U234	11.2		43.0	81.6	111.2	132.2	146.6
U235	1403.3		1006.6	918.8	904.6	923.5	937.0
U236			156.3	240.0	299.6	347.7	390.0
U238	5601.9		5911.3	6068.1	6129.6	6208.3	6232.8
Th232	24876.2		24478.4	24219.1	24065.5	23902.9	23812.0
Total	31892.5		31928.8	31956.8	31971.3	31986.7	31995.1
Discharge Inventories (kg)							
U233	316.7		414.4	446.9	458.7	464.1	466.6
U234	37.6		76.5	106.5	127.7	142.2	152.3
U235	498.8		374.0	357.7	365.0	383.4	396.1
U236	158.5		243.6	304.2	352.9	395.9	433.7
U238	5333.1		5627.9	5784.7	5845.1	5922.7	5947.1
Pa233	21.7		21.1	20.9	20.6	20.0	20.0
Th232	24282.6		23900.7	23652.2	23506.6	23354.1	23268.2
Pu239	73.4		79.1	82.0	83.6	85.7	87.0
Pu240	23.9		25.7	25.7	26.1	26.4	26.4
Pu241	18.5		19.4	19.1	19.4	19.4	19.5
Pu242	4.8		4.8	4.8	4.5	4.5	4.5

*Initial core not included; zeroth generation represents fresh denatured U^{233}/ThO_2 fuel loaded in cycles 2 and 3.

Table 2.7. Uranium Makeup Requirements

	Generation				
	1	2	3	4	5
Fully enriched (93% U^{235}) uranium for re-enrichment of bred fuel discharged from generation n-1 (kg)					
U^{234}	4.4	5.3	5.6	5.6	5.8
U^{235}	356.1	425.7	451.3	458.2	466.7
U^{238}	22.4	26.8	28.4	28.9	29.4
U	382.9	457.8	485.3	492.7	501.9
Denatured (20% U^{235}) makeup (kg)					
U^{234}	1.3	1.0	0.8	0.8	0.7
U^{235}	159.3	124.7	101.0	105.7	92.6
U^{238}	635.9	497.9	403.3	422.0	369.6
U	796.5	623.6	505.1	528.5	462.9
Totals (fully enriched + denatured U^{235}) (kg)					
U^{234}	5.7	6.3	6.4	6.4	6.5
U^{235}	515.4	550.4	552.3	563.9	559.3
U^{238}	658.3	524.7	431.7	450.9	399.0
U	1179.4	1081.4	990.4	1021.2	964.8

Table 2.8 gives the initial and discharge inventory for the 20% denatured $^{235}\text{U}/\text{ThO}_2$ fuel, and Table 2.9 gives the same information for each generation of recycle fuel. Also presented in these tables is the conversion ratio (CR) and the apparent fissile consumption (AFC) by fuel type. The apparent fissile consumption is defined as the net fissile consumption in grams per Megawatt Day.

The mass balances were determined with the LEOPARD¹ code combined with cross section libraries based on ENDF/B-IV. These libraries were recently developed by EPRI under their program for cross section standardization for thermal reactors.²

The mass balances were obtained by depleting each single enrichment fuel batch over its exposure to the desired core residence time and burnup. The soluble boron concentration was varied over each cycle to simulate the actual reactor operating environment. The depletion calculations were carried out in the critical spectrum mode by performing a buckling search at each step of the depletion. For the CE System 80 reactor operating at 75% capacity factor on the LEU throwaway cycle, the fissile requirement for each fresh fuel batch under equilibrium conditions is 3.00 w/o.⁸ Therefore the fissile requirement for denatured fuel was determined by requiring that the average end of cycle infinite multiplication factor for each denatured batch be equal to the average end of cycle multiplication factor for a 3.00 w/o uranium fuel batch.

The correlations developed by CE⁸ for the heterogeneous resonance integrals of ^{233}U and ^{232}Th were used to obtain self-shielding factors

Table 2.8. Inventories for Denatured U^{235}/ThO_2 Zeroth Generation and Makeup Assemblies, 1300 MW(e) PWR

	Initial inventory (kg/Initial TeM)	Discharge Inventory (kg/Initial TeM)
Fissile enrichment	4.40 w/o	
U-233		9.93
U-234	0.35	1.19
U-235	44.00	15.64
U-236		4.97
U-238	175.65	167.22
Pa-233		0.68
Th-232	780.00	761.39
Pu-239		2.30
Pu-240		0.75
Pu-241		0.58
Pu-242		0.15
CR*	0.635	
AFC [†]	0.456	

*Integral Conversion Ratio $\equiv$ Total fissile atoms bred through capture in fertile material during fuel exposure/Total fissile atoms consumed during fuel exposure.

[†]Apparent Fissile Consumption $\equiv$ net fissile material consumed in grams per megawatt day.

Table 2.9. Inventories for ThO₂ (denatured recycle U) Assemblies

	Generation				
	1	2	3	4	5
Initial Inventories (kg/Initial TeM)					
Fissile Enrichment	4.17 w/o	4.20 w/o	4.26 w/o	4.36 w/o	4.42 w/o
U-233	11.77	14.73	15.53	15.96	15.95
U-234	1.47	2.77	3.72	4.44	4.88
U-235	29.93	27.27	27.08	27.64	28.25
U-236	5.52	8.24	10.09	11.75	13.05
U-238	186.36	191.28	192.97	195.59	196.15
Th-232	764.95	755.72	750.61	744.62	741.72
Discharge Inventories (kg/Initial TeM)					
U-233	13.37	14.38	14.69	14.88	14.91
U-234	2.55	3.54	4.21	4.71	5.01
U-235	11.21	10.76	11.09	11.69	12.15
U-236	7.97	9.96	11.51	12.98	14.16
U-238	177.42	182.36	184.03	186.62	187.19
Pa-233	0.66	0.65	0.64	0.62	0.62
Th-232	746.92	738.06	733.22	727.58	724.84
Pu-239	2.50	2.59	2.64	2.71	2.75
Pu-240	0.81	0.81	0.82	0.83	0.83
Pu-241	0.61	0.60	0.61	0.61	0.61
Pu-242	0.15	0.15	0.14	0.14	0.14
CR	0.664	0.669	0.672	0.669	0.667
AFC	0.410	0.400	0.398	0.403	0.405

for these isotopes to account for shielding due to fuel lumping and for Doppler broadening. Due to the much lower density of the ^{238}U in denatured thorium fuel compared to LEU fuel, the ^{238}U resonance integral is larger in the denatured fuel--therefore the absorbing power per ^{238}U atom is increased. A correct value for the ^{238}U resonance integral is important both in determining the fissile requirement necessary to achieve the desired lifetime and in determining the fissile plutonium production. Foell and Connolly⁹ have performed experiments to determine the ^{238}U resonance integral over a range of low ^{238}U densities including that of ^{238}U in denatured fuel. These experimental results were used in the mass balance calculations. To account for Doppler broadening, the Hellstrand¹⁰ temperature correction was applied. Since the value of this resonance integral is a key factor in denatured fuel cycle analysis, calculated values using ORNL methods¹¹ were compared with those derived from the experiments of Foell and Connolly. The comparison was quite good and provided a verification of the technique used to obtain the resonance integral for the depletion calculations.

Another factor which should be considered in mass balance calculations involving denatured fuel is the interference between ^{238}U and ^{232}Th in the resonance energy range. The spacing of resonances in mixtures of resonance absorbers is obviously closer than for each individual absorber, making assumptions of widely spaced resonances less accurate. Interference effects for mixtures of ^{232}Th and ^{238}U measured by Foell and Connolly were in the range of a 1 to 4% reduction in the resonance integrals of each isotope while their corresponding calculated values were lower--between 0.5 and 1.5%. Based on their analysis a reduction of 1.5% was

applied to the resonance integrals of these nuclides for the denatured fuel cycle analysis. The interference between ^{233}U resonances and ^{232}Th resonances was also accounted for based on CE calculations reported in Ref. 8.

2.3 Plutonium Production in PWRs and CANDUs

Plutonium production in PWRs and CANDUs has been compared for a near equilibrium fuel cycle for the following cases:

1. A PWR operating on the low enriched uranium cycle with no recycle.
2. A PWR operating on the denatured UO_2/ThO_2 fuel cycle in the self-generated uranium recycle mode.
3. A HWR operating with natural uranium with no recycle.

The PWR cases considered a 1300 MW(e) Combustion Engineering System 80 reactor operating at 75% capacity factor with annual refueling of one-third of the core. Therefore, the annual energy production is $0.75 \times 1300 \text{ MW(e)} \times 1 \text{ yr} = 0.975 \text{ GW(e)-yr}$. In the first case, low enriched uranium (LEU) fuel (~3.0 w/o) is loaded annually; the uranium and plutonium content of the discharged fuel is not recycled. The denatured PWR case involves the self-generated recycle of uranium (including both ^{233}U and ^{235}U) initially bred in UO_2/ThO_2 fuel assemblies in which the fissile uranium content is denatured ^{235}U (20% enriched). No exogenous source of ^{233}U is assumed for this case. From information presented in Ref. 8 for the above annual energy production, 214 kg of fissile plutonium is discharged for case 1. For case 2, for the same annual energy production, 101 kg of fissile plutonium is discharged.

The HWR case involves the use of natural uranium in the 508 MW(e) Pickering CANDU reactor. This reactor is composed of 4680 fuel bundles, each initially containing 19.8 kg of uranium. An automatic on-line refueling machine moves the fuel bundles in a series of steps through the reactor as burnup proceeds: therefore plutonium-bearing elements are

discharged "continuously" rather than once annually. The elements are discharged at an average burnup of 7.5 MWD/kgU. From information presented in Ref. 12, 345 kg of fissile plutonium is discharged from natural uranium fueled CANDUs for each 800 MW(e)-yr of energy produced.

The above information is summarized in Table 2.10, which shows the amount of fissile plutonium discharged for each GW(e)-yr of energy produced. It can be seen from Table 2.10 that, for a given energy production, the CANDU reactor fueled with natural uranium discharges approximately twice as much fissile plutonium as the PWR on the LEU cycle and approximately four times as much plutonium as the PWR using denatured UO_2/ThO_2 fuel with self-generated recycle.

Table 2.10. Plutonium Production in PWRs and CANDUs

Case	Fissile plutonium discharged per unit of electrical energy produced [kg/GW(e)-yr]
PWR - LEU fuel, no recycle	214
PWR - Denatured UO_2/ThO_2 fuel with self- generated uranium recycle	104
CANDU - natural uranium fuel, no recycle	431

2.4 Mass Balance Information for the Standard HTGR

Fuel cycle mass balance information has been calculated for the 1160 MW(e) reference design HTGR. The HTGR fuel management scheme used for this analysis was obtained from Ref. 13. With the exception of the initial core, fuel in each of four approximately equal volume segments remains in the reactor for four cycles. The case presented here involves self-generated recycle of the bred uranium with one recycle of the fully enriched makeup uranium. Recycle of uranium bred from thorium continues throughout the history of the reactor. The calculations assumed annual refueling with a constant capacity factor of 80%. Reactor specifications are given in Table 2.11

The C/Th ratio for the first core is 214 while the C/Th ratio for reload fuel is 238. Detailed descriptions of the standard HTGR fuel element and fuel particles used in this analysis are given in Appendix 1, Ref. 14. Table 2.12a and 2.12b summarize ORNL's results for the fuel masses charged and discharged. Tables presenting results obtained by GAC for the same reactor at the same capacity factor were presented in the previous quarterly report (Ref. 13). That information shows that the 30 year cumulative ^{235}U requirement calculated by GAC is 13,570 kg while the ORNL calculation yields 13,114 kg, which is 3.4% less than the GAC value. The ORNL results yield a 30 year cumulative uranium ore requirement of 3328 ST U_3O_8 and a 30 year cumulative separative work requirement of 3321 MT(SWU) for this 1160 MW(e) reactor operating at a constant capacity factor of 80%.

Table 2.11. Summary of Core Parameters Used for HTGR Analysis

Core Heat Output	3000 MW(t)
Electrical Output	1160 MW(e)
Average Power Density	8.4 W/cc
Fuel Management	4 Batch, Annual refueling
Primary System Pressure	700 psia
Fuel Segment Volume Fractions	Segment A = 0.2698 Segment B,C,D = 0.2434 (each)
Carbon/Thorium ratio; Initial/Reload	214/238
Number of Fuel Elements	3944
Fuel Element Information:	
Coolant Passage Radius (Number per Element)	1.05 cm (66) 0.913 cm (6)
Fuel Hole*	0.794 cm
Annual Fuel Loading (Equilibrium Cycle)	9352 kg (U + Th)
Average Cycle Burnup (Equilibrium Cycle)	93670 MWD/MTHM
Capacity Factor	0.80
Core Conversion Ratio (Equilibrium Cycle)	0.68

*132 in fuel element; 76 in control element.

Table 2.12a. Mass balance information for the 1160 MW(e) standard HTGR

HEAVY METAL DISCHARGED (KG)											
CYCLE	THORIUM	*FRESH MAKEUP*		**BRED URANIUM REMOVED*		SPENT U RETIRED		TOTAL U REMOVED		***PLUTONIUM***	
		U-235	URANIUM	U-233	U-235	U-235	URANIUM	FISSILE	TOTAL	FISSILE	TOTAL
1	9906.0	221.6	294.0	149.5	0.9	161.4	0.0	371.9	455.5	0.8	1.4
2	8760.2	104.0	180.7	194.8	3.5	222.9	0.0	302.3	403.6	0.9	2.7
3	8590.6	56.4	135.1	223.4	7.3	269.6	0.0	287.2	404.7	1.1	4.8
4	8426.7	31.6	108.1	237.6	11.6	300.7	0.0	280.8	408.7	1.4	7.2
5	8444.0	72.1	222.1	241.0	11.2	302.5	0.0	324.3	524.6	2.7	13.5
6	7623.6	32.7	95.8	227.8	15.0	301.8	21.1	296.6	497.6	2.7	14.6
7	7627.4	41.3	117.2	232.2	18.7	319.0	10.5	302.7	513.2	2.8	15.1
8	7629.1	42.3	118.3	235.0	22.2	334.2	5.9	305.4	520.1	2.7	15.0
9	8457.9	43.3	119.9	260.8	26.3	376.3	3.4	333.8	556.5	2.7	14.8
10	7632.0	39.7	108.9	237.7	25.0	346.6	7.9	310.4	576.6	3.8	21.5
11	7632.1	43.4	118.9	236.5	26.6	351.1	3.6	310.0	521.4	2.4	13.5
12	7632.5	43.6	119.1	236.9	28.5	358.9	4.5	313.6	540.3	2.7	15.0
13	8460.4	44.6	121.9	261.4	31.4	396.2	4.6	342.1	580.7	2.7	15.4
14	7632.2	34.5	94.3	239.1	33.2	379.3	4.7	311.6	536.9	2.5	14.3
15	7633.3	41.6	112.9	238.2	31.7	372.7	4.4	315.9	542.9	2.5	14.4
16	7634.1	41.9	113.2	238.5	32.4	376.1	4.8	317.6	552.0	2.7	15.3
17	8462.8	43.1	116.0	262.9	34.3	410.4	4.9	345.2	589.2	2.7	15.6
18	7636.0	39.4	105.2	241.2	35.6	392.3	5.1	321.3	561.8	2.6	15.5
19	7636.3	41.9	111.7	239.6	35.6	391.7	3.9	321.0	553.3	2.4	14.3
20	7636.8	42.1	111.9	239.8	34.8	388.7	4.8	321.4	560.2	2.6	15.3
21	8465.8	47.0	124.4	264.3	36.1	420.8	4.8	352.2	604.8	2.8	16.1
22	7638.0	37.2	98.1	242.4	37.0	401.0	5.0	321.6	560.3	2.5	15.2
23	7638.7	43.1	113.3	240.9	36.7	399.8	4.6	325.4	568.4	2.6	15.4
24	7639.2	43.4	113.6	241.0	36.6	400.5	4.9	325.8	572.8	2.7	16.0
25	8468.2	48.3	126.1	265.4	37.2	428.7	4.9	355.8	613.7	2.8	16.5
26	7640.0	37.9	98.7	243.4	37.9	407.6	5.5	324.7	571.7	2.6	16.1
27	7640.3	43.7	113.8	241.7	37.4	405.2	4.4	327.2	570.5	2.5	15.3
28	7640.6	43.9	113.9	241.8	37.1	405.5	5.1	327.9	578.8	2.7	16.4
29	8469.8	48.9	126.5	266.3	38.1	436.2	5.2	358.4	622.2	2.9	17.2
30	32319.9	490.5	747.9	992.9	150.0	1653.9	53.7	1687.2	2695.3	8.1	37.0

Table 2.12b. Mass balance information for the 1160 MW(e) standard HTGR

HEAVY METAL CHARGED (KG)										
CYCLE	THORIUM	*FRESH MAKEUP*		*** RECYCLED BRED U ***			*RECY.SPENT U*		TOTAL U CHARGED	
		U-235	URANIUM	U-233	U-235	URANIUM	U-235	URANIUM	FISSILE	URANIUM
1	37482.8	1701.8	1827.0	0.0	0.0	0.0	0.0	0.0	1701.8	1827.0
2	9127.9	808.4	867.9	0.0	0.0	0.0	0.0	0.0	808.4	867.9
3	8234.7	338.4	363.3	146.4	0.9	158.2	217.2	288.2	702.9	809.6
4	8234.7	406.3	436.1	190.8	3.4	218.4	101.9	177.1	702.5	831.7
5	8234.7	406.3	436.1	218.9	7.2	264.2	55.3	132.4	687.7	832.7
6	9127.9	410.0	440.1	232.8	11.4	294.6	30.9	105.9	685.1	840.7
7	8234.7	369.9	397.1	236.2	11.0	296.4	70.6	217.7	687.7	911.2
8	8234.7	403.2	432.8	223.2	14.7	295.7	32.1	93.9	673.2	822.5
9	8234.7	403.2	432.8	227.6	18.3	312.6	40.5	114.9	689.5	860.3
10	9127.9	412.5	442.8	230.3	21.8	327.5	41.5	115.9	706.0	886.2
11	8234.7	319.5	343.0	255.5	25.7	368.7	42.4	117.5	643.2	829.3
12	8234.7	381.0	409.0	232.9	24.5	339.7	39.0	106.8	677.4	855.5
13	8234.7	381.0	409.0	231.7	26.0	344.1	42.5	116.5	681.3	869.6
14	9127.9	389.6	418.3	232.2	27.9	351.7	42.7	116.7	692.5	886.7
15	8234.7	351.5	377.4	256.2	30.8	388.2	43.8	119.4	682.2	885.0
16	8234.7	372.8	400.2	234.3	32.6	371.7	33.8	92.4	673.4	864.3
17	8234.7	372.8	400.2	233.4	31.1	365.2	40.7	110.5	678.0	876.0
18	9127.9	413.2	443.6	233.7	31.7	368.5	41.1	111.0	719.7	923.1
19	8234.7	325.4	349.3	257.6	33.6	402.2	42.2	113.7	658.9	865.2
20	8234.7	374.8	402.3	236.3	34.9	384.4	38.6	103.1	684.7	889.9
21	8234.7	374.8	402.3	234.7	34.9	383.9	41.1	109.5	685.5	895.7
22	9127.9	415.4	446.0	234.9	34.1	380.9	41.3	109.7	725.7	936.6
23	8234.7	324.9	348.8	259.0	35.4	412.3	46.0	121.9	665.3	883.0
24	8234.7	373.9	401.4	237.5	36.3	393.0	36.5	96.2	684.2	890.5
25	8234.7	373.9	401.4	236.1	36.0	391.8	42.3	111.1	688.3	904.2
26	9127.9	414.4	444.9	236.1	35.8	392.5	42.5	111.3	728.9	948.7
27	8234.7	321.9	345.6	260.1	36.5	420.2	47.3	123.6	665.8	889.3
28	8234.7	377.3	405.0	238.5	37.1	399.5	37.1	96.8	690.1	901.3
29	8234.7	377.3	405.0	236.8	36.7	397.1	42.9	111.5	693.7	913.7
30	9127.9	418.2	449.0	236.9	36.4	397.4	43.0	111.7	734.5	958.0

2.5 Comparison of Equilibrium Makeup, Cumulative Ore and Cumulative Separative Work Requirements for Various Fuel Cycles in Thermal Reactors

A comparison of equilibrium makeup, cumulative ore and cumulative separative work requirements has been made for various fuel cycles in PWRs and HTGRs. The annual makeup requirements in kg of ^{235}U (or ^{233}U) are shown in Fig. 2.1 for the following cases:

1. A PWR on the low enriched uranium fuel cycle with no recycle.
2. A PWR with self-generated recycle of uranium and plutonium.
3. A PWR using fully enriched UO_2/ThO_2 fuel with self-generated uranium recycle.
4. A PWR using denatured UO_2/ThO_2 fuel with self-generated uranium recycle (see section 2.2).
5. A PWR using denatured $^{233}\text{UO}_2/\text{ThO}_2$ fuel. For this case an exogenous source of ^{233}U is assumed.
6. A standard HTGR (conversion ratio = 0.67) utilizing fully enriched UO_2/ThO_2 fuel with self-generated uranium recycle.
7. A hi-gain HTGR (conversion ratio = 0.83) utilizing fully enriched UO_2/ThO_2 fuel with self-generated uranium recycle.

Results for cases 1, 2, and 3 were obtained from Ref. 8, for case 5 from Ref. 15, and for case 7 from Ref. 16. Cases 4 and 6 were calculated at ORNL.

As can be seen, the thorium based fuels have less makeup requirements. The makeup requirements for case 4 are greater than case 3 because some of the bred fuel in case 4 is plutonium, which cannot be recycled into the denatured PWR.

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IN THERMAL REACTORS MAKEUP FISSILE
REQUIREMENTS FOR THORIUM CYCLES
COMPARE FAVORABLY WITH THOSE FOR
U-PU CYCLES

NEAR EQUILIBRIUM ANNUAL MAKE-UP
BASIS: ANNUAL ENERGY PRODUCTION = 1 Gwe-yr

Fig. 2.1. Equilibrium makeup fissile requirements for various thermal reactor fuel cycles.

Cumulative ore requirements for the PWR LEU no recycle case and the self-generated uranium cases (cases 2, 3, 4, 6, and 7) are shown in Fig. 2.2. The thorium based cycles have less ore requirements than the PWR with U and Pu recycle--except for the denatured PWR with self-generated uranium recycle which has a slightly greater cumulative ore requirement. The standard HTGR and the hi-gain HTGR have the lowest ore requirements of the cases considered.

Cumulative separative work requirements for the same cases are shown in Fig. 2.3. As can be seen, the thorium based reactors have higher separative work requirements than the PWR U-Pu recycle case--except for the hi-gain HTGR.

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IN THERMAL REACTORS ORE REQUIREMENTS FOR
THORIUM CYCLES ARE GENERALLY LESS
THAN FOR U-PU CYCLES

30 YR REQUIREMENT
BASIS: 1000 MW(e) PLANT AT 75% CF

*Assumes 0.2% tails.

Fig. 2.2. Cumulative uranium ore requirements for various fuel cycles in thermal reactors.

IN THERMAL REACTORS SEPARATIVE WORK
REQUIREMENTS FOR THORIUM CYCLES ARE
GENERALLY GREATER THAN FOR U-PU CYCLES

30 YR REQUIREMENT
BASIS: 1000 MW(e) PLANT AT 75% CF

*Assumes 0.2% tails.

Fig. 2.3. Cumulative separative work requirements for various thermal reactor fuel cycles.

3. FAST REACTOR CALCULATIONS

T. J. Burns
J. R. White

3.1 Methods and Data Development

Mass balance calculations for various LMFBR fuel cycles are being performed using the CITATION¹⁷ computer code. Modifications to CITATION have been made in order to determine the appropriate equilibrium feed requirements for a variety of recycle options. Cross sections required for the fast reactor mass balance calculations are being generated from a 126 energy group, pseudo-composition-independent multigroup library, using the SPHINX¹⁸ self-shielding code. The AMPX¹¹ code system is being used to collapse the 126 group library to a 9 group library suitable for the mass balance calculations. A small computer program has been written to reformat the AMPX generated cross sections into CITATION format. This program is currently being verified and documented and will subsequently be included in the AMPX system. Preliminary cross sections for the LMFBR models have been generated via the above calculational system and are currently being verified.

3.2 Mass Balances for Denatured LMFBRs

Preliminary calculations for equilibrium cycle mass balances for two denatured LMFBRs have been completed. The two denatured fuels correspond to denaturing criteria (i.e., allowable ^{233}U in ^{238}U) of 12% and 20%. Although updated mass balances for all reactors will be produced from the production cross section library which is currently being generated, these preliminary results serve to indicate trends and to point out possible areas for additional investigation.

The reactor model utilized evolved from that proposed by the Large Core Code Evaluation Working Group¹⁹. Although the model does not correspond to a particular commercial design, it is felt to be "representative" of a large core commercial LMFBR. An axial blanket (at the same oxide density as the core) was added since the original model consisted only of the reactor midplane. The equilibrium mass balances for the two denatured reactors were calculated assuming that the reference 2 zone $\text{PuO}_2/^{238}\text{UO}_2$ fuel was replaced by the denatured fuel and the original depleted uranium oxide blankets replaced by ThO_2 . The original heavy metal volume fraction was maintained for the denatured reactors, the only correction being for the reduced density of ThO_2 . Thus, these preliminary calculations do not address the possibility of modifying the fuel elements to account for the higher melting point and thermal conductivity of ThO_2 relative to UO_2 . For all calculations, a fuel management scheme which replaced 1/3 of each core zone and 1/6 of the radial blanket each year was employed. A capacity factor of 0.75 was assumed.

The results of these calculations are summarized in Table 3.1. Certain characteristics of the denatured reactors are evident. The decreased breeding ratios of the denatured reactors reflect two primary effects: the substitution of ^{233}U for fissile Pu and corresponding reduction in the average number of neutrons produced per fission, and a reduction in the number of fertile nuclei fissions (due to the significantly smaller fission cross section of ^{232}Th relative to ^{238}U). One especially noteworthy feature of the denatured reactors is the marked reduction in the amount of fissile Pu discharged per cycle compared to that of the reference system.

Table 3.1. LMFBR Equilibrium Cycle Parameters

Fuel	Breeding Ratio	Reload Requirements (KG/GW(e) - yr)		Discharge (KG/GW(e) - yr)		Excess Fissile (KG/GW(e) - yr)	
		²³³ U	Pu	²³³ U	Pu	²³³ U	Pu
Pu/ ²³⁸ U	1.30	-	746	-	961	-	+215
²³³ U/ ²³⁸ U (12%)	1.13	778	-	518	361	-260	+361
²³³ U/ ²³⁸ U - Th (20%)	1.10	815	-	698	203	-111	+203

Table 3.2. Energy Center LMFBR Equilibrium Cycle Parameters

Fuel	Breeding Ratio	Reload Requirements (KG/GW(e) - yr)		Discharge (KG/GW(e) - yr)		Excess Fissile (KG/GW(e) - yr)	
		²³³ U	Pu	²³³ U	Pu	²³³ U	Pu
Pu/ ²³⁸ U - ThO ₂ RE	1.28	761	-	143	835	+143	+ 74
Pu/Th	1.19	-	852	580	394	+580	-458

As indicated in Table 3.1, the two denatured reactors considered are net ^{233}U consumers. Although both have breeding ratios greater than unity, a portion of the bred fissile material is Pu, which cannot be recycled back into the denatured reactors. As Fig. 3.1 illustrates, the isotopic composition of the bred fissile material is strongly dependent on the allowable fissile enrichment. In particular the region from 12% to 20% is the most crucial. (The 12% limitation (^{233}U in ^{238}U) corresponds to the average enrichment required if ThO_2 is excluded from the core zones.) Since the major gain in ^{233}U production occurs in this region, it is clear that there is a strong incentive to set the allowable percentage of ^{233}U as high as practicable from a proliferation standpoint.

3.3 ^{233}U Sources

As mentioned above, both denatured reactors considered require a source of ^{233}U to provide makeup requirements. Since ^{233}U is not a naturally occurring isotope, it must be "manufactured" from ^{232}Th . Two possible means of generating the required ^{233}U supply have been considered: the reference Pu/ ^{238}U system with a ThO_2 radial blanket, and a Pu/Th LMFBR. Both of these reactors must operate within a secure energy center because they are not denatured. The results of equilibrium cycle calculations are summarized in Table 3.2. The Pu/U/Th blanket LMFBR generates slightly more Pu than it requires, while the Pu/Th LMFBR is a net Pu consumer. However, the Pu/Th reactor produces over a factor of 4 more ^{233}U per cycle than does the Pu/U/Th case.

Fig. 3.1. Fissile Isotope Production Versus Allowable ^{233}U Enrichment.

3.4 Symbiotic Reactor Systems

Since the denatured reactors must be supplied with makeup ^{233}U , it is possible to combine the denatured reactors with a ^{233}U production reactor operating inside the energy center, in effect creating a symbiotic relationship. The dispersed denatured reactors are net ^{233}U consumers and Pu producers while the Pu/Th reactors inside the energy center produce the required ^{233}U makeup. Typically, meeting the fissile requirements for both types of reactors results in a symbiotic "window" in terms of the number of outside reactors supported per inside reactor. The upper bound on the window is governed by the ^{233}U production while the lower bound is set by the Pu production. Table 3.3 summarizes some results, assuming equilibrium conditions and 2% reprocessing and fabrication losses. As indicated by Table 3.3, the denatured reactor systems coupled to the Pu/Th LMFBR result in much higher outside to inside power ratios.

Table 3.3. Fast Reactor Symbiotic System Parameters

Inside Reactor	Outside Reactor	Power Ratio [GW(e) Outside/GW(e) Inside]	
		Minimum ^a	Maximum ^b
Pu/ ²³⁸ U-ThO ₂ RB LMFBR	12% Denatured LMFBR	0.00	0.52
Pu/ ²³⁸ U-ThO ₂ RB LMFBR	20% Denatured LMFBR	0.00	1.07
Pu/Th LMFBR	12% Denatured LMFBR	1.32	2.10
Pu/Th LMFBR	20% Denatured LMFBR	2.34	4.34

^aSymbiosis constrained by Pu production

^bSymbiosis constrained by ²³³U production

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