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Thorium Assessment Study Quarterly Progress Report for First Quarter Fiscal 1977

OAK RIDGE NATIONAL LABORATORY

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THORIUM ASSESSMENT STUDY
QUARTERLY PROGRESS REPORT FOR FIRST 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.

I. Highlights

During the first quarter of FY' 1977, a report of an extensive preliminary study, "The Case for the Thorium Fuel Cycle," was revised and published, with distribution slated for early January. Preliminary HTGR mass balances were obtained, and a feasibility study was performed for the denatured fuel cycle in LMFBR's. Methods and data were assembled and verified for both thermal and fast reactor systems. Scenarios involving denatured fuel cycles and energy parks were tentatively identified for study, along with the objectives for investigation in each scenario.

II. Thermal Reactor Calculations (J. C. Cleveland, W. E. Thomas)

A. Mass Balances for the Standard HTGR

Summary

Fuel cycle mass balance information for the standard HTGR design calculated by Bill Thomas in 1973, using methods available at ORNL at that time, is provided herein. The format should be acceptable for use by HEDL as it was developed from examples provided by Ron Omberg. Also included, in the same format, for internal comparison is mass balance information for the standard HTGR as calculated by GAC.

Bill is currently performing mass balance calculations for the standard HTGR under the Thorium Assessment Program. These calculations employ improved methods and will be made available to HEDL when available. They are scheduled for completion early in January 1977.

It is our understanding that the mass balance information included herein has been requested by HEDL for purposes of "debugging" their computer code and will be discarded once they receive the results of the calculations scheduled for completion in January.

Standard HTGR Reference Design Fuel Cycle Information

The HTGR fuel management scheme used for this analysis was obtained from Ref. 1. 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%.

Detailed descriptions of the standard HTGR fuel element and fuel particles used in this analysis are given in Appendix 1. These tables are included for our reference; the information was not requested by HEDL.

Mass Balance Information

Table 1-a and 1-b summarize ORNL's results for the fuel masses charged and discharged annually in an 1160 MW(e) standard HTGR with a constant capacity factor of 80%. Information provided by GAC² for the same reactor at the same capacity factor is shown in Table 2-a and 2-b. The ORNL calculations assumed equal reload segment volumes (0.25 of core volume for each segment) while the actual volumes of the four reload segments are not exactly equal as is reflected in the slight oscillation in fissile loading in GAC's "equilibrium" cycles. Therefore, when comparing total charged and discharged masses as calculated by ORNL and GAC, four year averages should be considered. Although ORNL results indicate slightly higher fissile loadings, 3 to 6%, the agreement with GAC appears acceptable and will probably improve when results of current calculations are available.

Tables 3-a and 3-b present a modification of ORNL's mass balances to reflect a variable plant capacity factor. These

Table 1-a. Mass balance information for the 1160 MW(e) standard HTGR — ORNL calculations

Annual Charge (kg)

Year	Thorium	Makeup fuel		Recycle bred fuel			Recycle spent makeup fuel		Total uranium loaded	
		U235	U	U233	U235	U	U235	U	U-fissile	U
1	37496.9	1767.6	1898.9	-	-	-	-	-	1767.6	1898.9
2	8426.9	777.9	835.7	-	-	-	-	-	777.9	835.7
3	8426.9	391.0	420.0	136.4	0.8	146.8	192.9	253.8	721.1	820.6
4	8426.9	391.0	420.0	199.5	3.3	226.9	114.6	194.0	708.4	840.9
5	8426.9	391.0	420.0	230.7	7.0	276.2	70.2	159.8	698.9	856.0
6	8426.9	391.0	420.0	246.3	11.3	309.2	43.5	138.0	692.1	867.2
7	8426.9	409.9	440.3	224.1	9.8	279.3	78.6	227.9	722.4	947.5
8	8426.9	409.9	440.3	235.3	14.4	308.0	42.5	117.8	702.1	866.1
9	8426.9	409.9	440.3	241.9	18.7	330.1	44.7	120.1	715.2	890.5
10	8426.9	374.3	402.0	245.7	22.5	347.2	46.1	121.6	688.6	870.8
11	8426.9	340.7	366.0	248.3	25.8	361.2	47.3	122.8	662.1	850.0
12	8426.9	367.9	395.2	247.0	23.9	353.2	50.5	129.9	689.3	878.3
13	8426.9	367.9	395.2	247.2	27.0	364.7	50.5	129.8	692.6	889.7
14	8426.9	373.0	400.7	247.4	29.5	374.4	50.3	129.7	700.2	904.8
15	8426.9	378.6	406.7	247.6	31.6	382.6	45.5	118.0	703.3	907.3
16	8426.9	378.6	406.6	248.1	33.3	389.8	41.2	107.2	701.2	903.6
17	8426.9	378.6	406.6	248.8	32.3	386.5	45.0	116.1	704.7	909.2
18	8426.9	378.6	406.6	249.5	33.9	393.8	45.6	116.7	707.6	917.1
19	8426.9	373.6	406.6	250.2	35.3	400.1	46.9	119.1	711.0	925.8
20	8426.9	378.6	406.6	250.9	36.3	405.5	48.2	121.5	714.0	933.6
21	8426.9	378.6	406.6	251.5	37.2	410.3	48.8	122.1	716.1	939.0
22	8426.9	351.2	377.2	252.1	36.8	408.7	49.3	122.6	689.4	908.5
23	8426.9	362.5	389.4	252.8	37.6	413.3	49.8	123.1	702.7	925.8
24	8426.9	362.5	389.4	252.7	38.2	416.8	50.0	123.3	703.4	929.5
25	8426.9	367.5	394.8	252.6	38.6	419.6	50.1	123.4	708.8	937.8
26	8426.9	373.1	400.7	252.5	39.0	422.1	49.9	123.3	714.5	946.1
27	8426.9	373.1	400.7	252.5	38.8	420.7	46.1	114.1	710.5	935.5
28	8426.9	373.1	400.7	252.9	39.2	423.7	47.7	117.9	712.9	942.3
29	8426.9	373.1	400.7	253.3	39.5	426.3	47.9	118.1	713.8	945.1
30	8426.9	373.1	400.7	253.6	39.7	428.5	48.9	120.1	715.3	949.3

Table 1-b. Mass balance information for the 1160 MW(e) standard HTGR - CRNL calculations

Annual Discharge (kg)												
Year	Thorium	Bred uranium			Makeup uranium		Once recycled makeup uranium (Retired at discharge)		Total uranium discharged for recycle		Plutonium	
		U233	U235	U	U235	U	U235	U	U-fissile	U	Pu-fissile	Pu
1	9185.8	137.8	0.8	148.4	194.8	256.3	-	-	333.4	404.7	0.6	0.8
2	9007.9	201.5	3.3	229.2	115.8	195.9	-	-	320.6	425.1	0.8	2.5
3	8836.6	233.0	7.1	279.0	70.9	161.4	-	-	311.0	440.4	1.2	5.0
4	8670.2	248.8	11.4	312.4	43.9	139.4	-	-	304.1	451.8	1.5	7.9
5	7804.9	226.4	9.9	282.1	79.4	230.2	-	-	315.7	512.3	2.4	12.1
6	7810.4	237.8	14.5	311.2	43.0	119.0	21.4	92.1	295.3	430.2	2.5	13.3
7	7814.2	244.4	18.9	333.5	45.1	121.3	13.5	86.3	308.4	454.8	2.6	14.3
8	7816.8	248.2	22.7	350.8	46.6	122.9	8.7	82.6	317.5	473.7	2.7	15.0
9	7819.3	250.8	26.1	364.9	47.8	124.1	5.6	79.3	324.7	489.0	2.7	15.4
10	7821.2	249.5	24.2	356.9	51.0	131.1	10.3	128.2	324.7	488.0	3.7	20.9
11	7821.0	249.3	27.3	368.5	51.0	131.2	5.5	65.5	328.1	499.7	2.5	14.0
12	7820.5	250.0	29.8	378.2	50.8	131.0	5.8	66.1	330.6	509.2	2.5	14.3
13	7819.5	250.1	31.9	386.4	45.9	119.2	5.9	66.5	327.9	505.6	2.4	14.1
14	7819.1	250.7	33.6	393.8	41.7	108.2	6.0	66.8	326.0	502.0	2.4	13.9
15	7820.2	251.3	32.7	390.4	45.4	117.3	5.5	70.4	329.4	507.7	2.5	14.5
16	7821.5	252.1	34.3	397.8	46.0	117.9	5.6	70.5	332.4	515.7	2.6	14.8
17	7823.1	252.8	35.6	404.2	47.4	120.3	5.6	70.7	335.8	524.5	2.6	15.2
18	7824.5	253.4	36.7	409.7	48.7	122.7	6.1	64.5	338.8	532.4	2.5	14.8
19	7825.7	254.1	37.6	414.4	49.3	123.3	5.6	58.7	341.0	537.7	2.5	14.4
20	7826.8	254.7	37.2	412.6	49.8	123.8	5.2	63.6	341.7	536.4	2.5	14.8
21	7827.9	255.4	38.0	417.5	50.3	124.4	5.3	63.8	343.7	541.9	2.6	15.0
22	7828.3	255.3	38.6	421.0	50.5	124.6	5.5	65.0	344.4	545.6	2.6	15.4
23	7828.4	255.2	39.0	423.9	50.6	124.6	6.7	66.1	344.8	548.5	2.6	15.7
24	7828.1	255.1	39.4	426.4	50.4	124.5	5.7	56.3	344.9	550.9	2.7	16.0
25	7827.6	255.1	39.1	425.0	46.6	115.3	5.8	56.4	340.8	540.3	2.6	15.5
26	7827.9	255.5	39.5	428.0	48.2	119.1	5.9	56.6	343.2	547.1	2.6	15.9
27	7828.4	255.9	39.9	430.6	48.4	119.3	6.9	56.7	344.2	549.9	2.7	16.1
28	7829.1	256.2	40.1	432.9	49.4	121.3	7.0	56.7	345.7	554.2	2.7	16.3
29	7829.8	256.5	40.4	434.9	50.5	123.5	7.0	56.8	347.4	558.4	2.7	16.6
30	32203.1	1036.8	160.6	1745.7	498.5	765.2	65.2	316.2	1695.9	2510.9	7.8	37.1

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Table 2-a. Mass balance information for the 1160 MW(e) standard HTGR - GA calculations

Annual Charge (kg)

Year	Thorium	Makeup fuel		Recycle bred fuel			Recycle spent makeup fuel		Total uranium loaded	
		U235	U	U233	U235	U	U235	U	U-fissile	U
1	37503.3	1669.3	1792.5	—	—	—	—	—	1669.3	1792.5
2	9115.7	740.6	795.3	—	—	—	—	—	740.6	795.3
3	8198.7	349.6	375.4	138.0	0.8	149.0	204.2	276.2	692.6	800.6
4	8633.6	416.8	447.6	176.3	3.0	202.5	88.4	165.3	684.5	815.4
5	7763.9	393.4	422.5	209.7	6.6	254.8	46.5	130.2	656.2	807.5
6	9115.7	452.5	485.9	198.2	9.3	253.5	21.4	95.0	681.4	834.4
7	8198.7	370.7	398.1	215.3	9.7	273.4	46.2	188.9	641.9	860.3
8	8633.6	425.9	457.3	200.7	12.7	269.3	23.5	90.8	662.8	817.4
9	7763.9	379.9	407.9	213.5	15.9	296.0	29.1	109.5	638.4	813.4
10	9115.7	453.6	487.1	194.9	18.3	285.1	27.9	103.9	694.7	876.1
11	8198.7	359.6	386.1	226.8	20.7	329.8	32.4	119.7	639.5	835.7
12	8633.6	417.4	448.2	206.4	20.5	306.5	26.8	98.4	671.1	853.1
13	7763.9	372.7	400.3	216.5	22.0	323.6	31.1	113.4	642.3	837.3
14	9115.7	443.7	476.5	197.0	23.1	307.4	28.2	101.5	692.0	885.4
15	8198.7	360.0	386.6	229.0	24.9	350.1	34.1	121.7	648.0	858.4
16	8633.6	413.1	443.6	209.0	26.2	334.3	27.4	96.7	675.7	874.7
17	7763.9	368.6	395.8	218.8	25.9	343.4	32.1	112.8	645.4	852.0
18	9115.7	444.1	476.9	199.0	26.0	323.0	29.0	101.0	698.1	900.9
19	8198.7	356.9	383.3	230.8	27.1	363.3	34.9	120.6	649.7	867.2
20	8633.6	409.8	440.0	210.8	28.1	346.4	28.6	98.2	677.3	884.6
21	7763.9	368.6	395.8	218.8	25.9	343.4	32.1	112.8	645.4	852.0
22	9115.7	444.1	476.9	199.0	26.0	323.0	29.0	101.0	698.1	900.9
23	8198.7	356.9	383.3	230.8	27.1	363.3	34.9	120.6	649.7	867.2
24	8633.6	409.8	440.0	210.8	28.1	346.4	28.6	98.2	677.3	884.6
25	7763.9	368.6	395.8	218.8	25.9	343.4	32.1	112.8	645.4	852.0
26	9115.7	444.1	476.9	199.0	26.0	323.0	29.0	101.0	698.1	900.9
27	8198.7	356.9	383.3	230.8	27.1	363.3	34.9	120.6	649.7	867.2
28	8633.6	409.8	440.0	210.8	28.1	346.4	28.6	98.2	677.3	884.6
29	7763.9	368.6	395.8	218.8	25.9	343.4	32.1	112.8	645.4	852.0
30	9115.7	444.1	476.9	199.0	26.0	323.0	29.0	101.0	698.1	900.9

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Table 2-b. Mass balance information for the 1160 MW(e) standard HTGR - GA calculations

Annual Discharge (kg)

Year	Thorium	Bred uranium			Makeup uranium		Once recycled makeup uranium (Retired at discharge)		Total uranium discharged for recycle		Plutonium	
		U233	U235	U	U235	U	U235	U	U-fissile	U	Pu-fissile	Pu
1	9942.2	141.5	0.8	152.9	209.4	283.2	—	—	351.7	456.1	0.6	2.1
2	8769.8	180.8	3.1	207.7	90.5	169.6	—	—	274.5	377.3	0.8	2.2
3	9051.1	215.0	6.8	261.3	47.7	133.5	—	—	259.5	394.8	1.0	4.0
4	7995.5	203.3	9.6	260.0	21.9	97.4	—	—	234.3	357.4	1.1	5.3
5	8436.5	220.9	10.0	280.4	47.3	193.7	—	—	278.2	474.1	2.1	9.8
6	7595.2	205.9	13.0	276.2	24.1	93.1	14.8	101.0	243.0	369.3	2.2	10.6
7	8031.2	219.0	16.3	303.6	29.9	112.3	6.8	83.6	255.2	415.9	2.1	9.8
8	7136.8	199.9	18.7	292.4	28.7	106.6	3.7	82.6	247.3	399.0	2.0	8.7
9	8450.3	232.6	21.3	338.2	33.1	122.8	1.8	69.5	287.0	461.0	2.0	8.7
10	7631.1	211.7	21.1	314.4	27.4	100.9	3.9	135.7	250.2	415.3	2.4	9.9
11	8005.2	222.0	22.6	331.9	31.3	116.3	2.0	64.3	276.4	448.2	1.9	8.4
12	7199.9	202.1	23.7	315.3	28.9	104.1	2.5	77.1	254.7	419.4	1.9	8.5
13	8455.0	234.8	25.6	359.1	34.9	124.8	2.4	73.0	295.3	483.9	2.1	9.5
14	7605.5	214.3	26.9	342.8	28.0	99.3	2.8	84.0	269.2	442.1	2.0	8.9
15	8010.1	224.4	26.5	352.2	33.0	115.7	2.4	68.9	283.9	467.9	2.0	9.1
16	7204.1	204.1	26.7	331.3	29.7	103.6	2.3	79.3	260.5	434.9	2.0	9.1
17	8459.5	236.8	27.8	372.6	35.8	123.7	2.5	70.8	300.4	496.3	2.1	9.9
18	7609.5	216.2	28.9	355.3	29.3	100.7	3.1	84.7	274.4	456.0	2.1	9.6
19	8014.0	226.1	29.2	368.6	33.9	115.8	2.5	67.2	289.2	484.4	2.1	9.8
20	7207.2	205.5	28.5	343.1	30.5	103.5	2.9	78.1	264.5	446.6	2.1	9.6
21	8459.5	236.8	27.8	372.6	35.8	123.7	2.5	70.8	300.4	496.3	2.1	9.9
22	7609.5	216.2	28.9	355.3	29.3	100.7	3.1	84.7	274.4	456.0	2.1	9.6
23	8014.0	226.1	29.2	368.6	33.9	115.8	2.5	67.2	289.2	484.4	2.1	9.8
24	7207.2	205.5	28.5	343.1	30.5	103.5	2.9	78.1	264.5	446.6	2.1	9.6
25	8459.5	236.8	27.8	372.6	35.8	123.7	2.5	70.8	300.4	496.3	2.1	9.9
26	7609.5	216.2	28.9	355.3	29.3	100.7	3.1	84.7	274.4	456.0	2.1	9.6
27	8014.0	226.1	29.2	368.6	33.9	115.8	2.5	67.2	289.2	484.4	2.1	9.8
28	7207.2	205.5	28.5	343.1	30.5	103.5	2.9	78.1	264.5	446.6	2.1	9.6
29	8459.5	236.8	27.8	372.6	35.8	123.7	2.5	70.8	300.4	496.3	2.1	9.9
30	7609.5	216.2	28.9	355.3	29.3	100.7	3.1	84.7	274.4	456.0	2.1	9.6

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Table 3-a. Mass balance information for the 1160 MW(e) standard
HTGR - ORNL calculations with variable capacity factor

Annual Charge (kg)

Year	Capacity factor	Thorium	Makeup fuel		Recycle bred fuel			Recycle spent makeup fuel		Total uranium loaded	
			U235	U	U235	U235	U	U235	U	U-fissile	U
1	0.600	28122.7	1325.7	1424.2	—	—	—	—	—	1325.7	1424.2
2	0.660	6952.2	641.8	689.5	—	—	—	—	—	641.8	689.5
3	0.720	7584.2	351.9	378.0	122.8	0.7	132.1	173.6	228.4	649.0	738.5
4	0.720	7584.2	351.9	378.0	179.6	3.0	204.2	103.1	174.6	637.6	756.8
5	0.720	7584.2	351.9	378.0	207.6	6.3	248.6	63.2	143.8	629.0	770.4
6	0.720	7584.2	351.9	378.0	221.7	10.2	278.3	39.2	124.2	622.9	780.5
7	0.720	7584.2	368.9	396.3	201.7	8.8	251.4	70.7	205.1	650.2	852.8
8	0.720	7584.2	368.9	396.3	211.8	13.0	277.2	38.3	106.0	631.9	779.5
9	0.720	7584.2	368.9	396.3	217.7	16.8	297.1	40.2	108.1	643.7	801.5
10	0.720	7584.2	336.9	361.8	221.1	20.3	312.5	41.5	109.4	619.7	783.7
11	0.720	7584.2	306.6	329.4	223.5	23.2	325.1	42.6	110.5	595.9	765.0
12	0.720	7584.2	331.1	355.7	222.3	21.5	317.9	45.5	116.9	620.4	790.5
13	0.720	7584.2	331.1	355.7	222.5	24.3	328.2	45.5	116.8	623.3	800.7
14	0.720	7584.2	335.7	360.6	222.7	26.6	337.0	45.3	116.7	630.2	814.3
15	0.720	7584.2	340.7	366.0	222.8	28.4	344.3	41.0	106.2	633.0	816.6
16	0.720	7584.2	340.7	365.9	223.3	30.0	350.8	37.1	96.5	631.1	813.2
17	0.704	7415.7	333.2	357.8	218.9	28.4	340.1	39.6	102.2	620.1	800.1
18	0.689	7257.7	326.1	350.2	214.9	29.2	339.2	39.3	100.5	609.4	789.9
19	0.673	7089.1	318.5	342.1	210.5	29.7	336.6	39.5	100.2	598.1	778.8
20	0.657	6920.6	310.9	333.9	206.1	29.8	333.0	39.6	99.8	586.4	766.7
21	0.641	6752.1	303.4	325.8	201.5	29.8	328.8	39.1	97.8	573.8	752.4
22	0.626	6594.0	274.8	295.2	197.3	28.8	319.8	38.6	95.9	539.5	710.9
23	0.610	6425.5	276.4	296.9	192.8	28.7	315.1	38.0	93.9	535.8	705.9
24	0.594	6257.0	269.2	289.1	187.6	28.4	309.5	37.1	91.6	522.3	690.2
25	0.579	6099.0	266.0	285.7	182.8	27.9	303.7	36.3	89.3	513.0	678.7
26	0.563	5930.4	262.6	282.0	177.7	27.4	297.1	35.1	86.8	502.8	665.8
27	0.547	5761.9	255.1	274.0	172.6	26.5	287.7	31.5	78.0	485.8	639.6
28	0.531	5593.4	247.6	266.0	167.9	26.0	281.2	31.7	78.3	473.2	625.5
29	0.516	5435.4	240.6	258.5	163.4	25.5	275.0	30.9	76.2	460.4	609.6
30	0.500	5266.8	233.2	250.4	158.5	24.8	267.8	30.6	75.1	447.1	593.3

Table 3-b. Mass balance information for the 1160 MW(e) standard
HTGR - ORNL calculations with variable capacity factor

Annual Discharge (kg)

Year	Capacity factor	Thorium	Bred uranium			Makeup uranium		Once recycled makeup uranium (Retained at discharge)		Total uranium discharged for recycle		Plutonium	
			U233	U235	U	U235	U	U235	U	U-fissile	U	Pu-fissile	Pu
1	0.600	6889.4	103.4	0.6	111.3	146.1	192.2	—	—	250.1	303.5	0.5	0.8
2	0.660	7431.5	166.2	2.7	189.1	95.5	161.6	—	—	264.5	350.7	0.7	2.1
3	0.720	7952.9	209.7	6.4	251.1	63.8	145.3	—	—	279.9	396.4	1.1	4.5
4	0.720	7803.2	225.9	10.3	281.2	39.5	125.5	—	—	273.7	406.6	1.4	7.1
5	0.720	7024.4	203.8	8.9	253.9	71.5	207.2	—	—	284.1	461.1	2.2	10.9
6	0.720	7029.4	214.0	13.1	280.1	38.7	107.1	19.3	82.9	265.8	387.2	2.3	12.0
7	0.720	7032.8	220.0	17.0	300.2	40.6	109.2	12.2	77.7	277.6	409.3	2.3	12.9
8	0.720	7035.1	223.4	20.4	315.7	41.9	110.6	7.8	74.3	285.8	426.3	2.4	13.5
9	0.720	7037.4	225.7	23.5	328.4	43.0	111.7	5.0	71.4	292.2	440.1	2.4	13.9
10	0.720	7039.1	224.6	21.8	321.2	45.9	118.0	9.3	115.4	292.2	439.2	3.3	18.8
11	0.720	7033.9	224.8	24.6	331.7	45.9	118.1	5.0	59.0	295.3	449.7	2.3	12.6
12	0.720	7038.5	225.0	26.8	340.4	45.7	117.9	5.2	59.5	297.5	458.3	2.3	12.9
13	0.720	7037.6	225.1	28.7	347.8	41.3	107.3	5.3	59.9	295.1	455.0	2.2	12.7
14	0.720	7037.2	225.6	30.2	354.4	37.5	97.4	5.4	60.1	293.4	451.8	2.2	12.5
15	0.720	7038.2	226.2	29.4	351.4	40.9	105.6	5.9	63.4	296.5	456.9	2.3	13.1
16	0.720	7039.4	226.9	30.9	358.6	41.4	106.1	5.9	63.5	299.2	464.1	2.3	13.3
17	0.704	6884.3	222.5	31.3	355.7	41.7	105.9	5.8	62.2	295.5	461.6	2.3	13.4
18	0.689	6733.9	218.2	31.6	352.9	41.9	105.7	5.3	55.6	291.8	458.5	2.2	12.7
19	0.673	6583.4	213.8	31.6	348.6	41.5	103.7	4.7	49.4	286.9	452.3	2.1	12.1
20	0.657	6427.8	209.2	30.6	338.8	40.9	101.7	5.1	52.2	280.6	440.5	2.1	12.2
21	0.641	6272.1	204.6	30.4	334.5	40.3	99.7	5.0	51.1	275.4	434.2	2.1	12.0
22	0.626	6125.6	199.8	30.2	329.4	39.5	97.5	5.1	50.9	269.5	426.9	2.0	12.1
23	0.610	5969.2	194.6	29.7	323.2	38.6	95.0	5.1	50.4	262.9	418.2	2.0	12.0
24	0.594	5812.4	189.4	29.3	316.6	37.4	92.4	5.0	49.2	256.1	409.0	2.0	11.9
25	0.579	5665.2	184.6	28.3	307.6	33.7	83.4	4.9	48.1	246.7	391.0	1.9	11.2
26	0.563	5503.9	179.8	27.8	301.2	35.9	83.8	4.9	46.9	241.5	385.0	1.8	11.2
27	0.547	5352.7	175.0	27.3	294.4	33.1	81.6	4.7	45.6	235.3	376.0	1.8	11.0
28	0.531	5195.6	170.1	26.6	287.3	32.8	80.5	4.6	44.3	229.5	367.9	1.8	10.8
29	0.516	5059.2	165.4	26.1	280.5	32.6	79.7	4.5	43.1	224.1	360.2	1.7	10.7
30	0.500	20126.9	648.0	100.4	1091.1	311.6	478.3	40.8	197.6	1059.9	1569.3	4.9	23.2

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were generated at HEDL's request. Capacity factors which vary annually are used by HEDL to analyze energy systems in their ALPS computer code. Typically, these capacity factors are highest during the middle years of a plant's operation. Lower capacity factors in the first few years supposedly reflect difficulties arising from new equipment while lower capacity factors in the plant's later years reflect the expected increasing frequency of component failure. Mass balances in Table 3 do not represent additional fuel cycle calculations. They were derived from those in Table 1 by applying a simple algorithm suggested by HEDL in examples provided to ORNL, for converting mass balances for a constant capacity factor to mass balances for a variable capacity factor. The resulting mass balances therefore only approximate those which would be obtained from detailed neutronics calculations.

B. Methods and Data Preparation and Verification

Methods are being established for the analysis of denatured uraniuma-thoria fuel cycles in PWRs. The Combustion Engineering System - 80 reactor design has been chosen for the reference PWR design. Core and fuel assembly design information for this reactor has been obtained. The recent EPRI version of the LEOPARD point depletion computer code has been obtained from EPRI and made operational on the ORNL computer. This version of LEOPARD contains fine group microscopic cross section information which has been processed from the ENDF/B Version-IV data files. This code is being used to generate burnup and enrichment dependent four group microscopic cross sections for use in the PDQ-HARMONY neutron diffusion theory code system. Several thoria-uranium fueled critical experiments performed by Babcock and Wilcox are being analyzed with cross sections generated by LEOPARD to provide a benchmark for the PWR methods. Satisfactory agreement between

measured and calculated parameters over a range of experimental conditions will provide confidence in these methods.

III. Fast Reactor Calculations (T. J. Burns)

A. Denatured LMFBR Feasibility Study

The fast reactor work performed as part of the Thorium Assessment Program (TAP) consisted of two main efforts. Due to the increased awareness concerning nuclear safeguards, the initial effort was directed toward estimating the effects of "denatured" fuel (i.e., ^{233}U diluted with ^{238}U) on LMFBR's. The fissile component of such a fuel would not be chemically separable from the rest of the fuel (unlike fresh Pu- ^{238}U fuel). Further, such a fuel would be subject to enrichment limitations to prevent its direct use as weapons material. This analysis was based on a commercial-sized [1200MW(e)] conceptual LMFBR design developed by GE for benchmark use by the Large Core Code Evaluation Working Group. The standard Pu-fueled (2 zone) core with depleted $^{238}\text{UO}_2$ blankets was taken as the reference case. The denatured fuel was then substituted for the original fuel and various performance parameters calculated. Selected results of these calculations are summarized in Table 4. Also tabulated (in Table 5) are results obtained using a heterogeneous model in which Th radial blanket assemblies were interspersed among core assemblies. This model was considered to assess the possibility of increasing the amount of ^{233}U bred in the denatured reactor. It should be noted that the effects of fission products, control poisons, and isotopic changes due to burnup were neglected in these preliminary calculations. Thus, the performance parameters tabulated represent those projected for a clear BOL core.

TABLE 4. CALCULATIONAL RESULTS FOR LMFBR MODEL (2 ENRICHMENT ZONES)

	<u>Reference</u>	<u>Homogeneous Denatured</u>
I. Model		
a. Fuel	Pu/U-238	U-233/U-238
b. Radial Blanket	U-238	Th-232
c. Core Fissile Inventory (kg)	2933	2512
II. Performance Parameters		
a. k	1.0369	1.0369
b. Core Conversion Ratio	0.917	0.836
c. Breeding Ratio		
1. Core	0.906	0.831
2. Axial Blanket	0.201	0.189
3. Radial Blanket	<u>0.222</u>	<u>0.214</u>
TOTAL	1.329	1.234
d. $\sum_c^{\text{fertile}} \phi / \left(\sum_c^{\text{fertile}} \phi \right)_{\text{ref}}$	1.0	1.020
e. $\sum_a^{\text{fissile}} \phi / \left(\sum_a^{\text{fissile}} \phi \right)_{\text{ref}}$	1.0	1.098

TABLE 5. CALCULATIONAL RESULTS FOR DENATURED LMFBR MODELS

I. Model		Homogeneous	Heterogeneous Model
a.	Fuel	U-233/U-238	U-233/U-238
b.	Radial Blanket	Th-232	Th-232
c.	Internal Blanket	---	Th-232
d.	Core Fissile Inventory (kg)	2512	3241
e.	IB Fraction	---	33.9%
II. Performance Parameters			
a.	k	1.0369	1.0369
b.	Core Assembly Conversion Ratio	0.836	0.355
c.	Breeding Ratio		
1.	Core Assemblies Fu Breeding	0.831 1.020	0.350 0.431
2.	Axial Blankets	<u>0.189</u>	<u>0.081</u>
3.	Radial Blanket U-233 Breeding	0.214	0.310
4.	Internal Blanket	--- <u>0.214</u>	<u>0.473</u> <u>0.783</u>
	TOTAL	1.234	1.214
d.	$\left(\sum_c^{\text{fertile}} \phi \right) / \left(\sum_c^{\text{fertile}} \phi \right)_{\text{ref}}$	1.020	1.045
e.	$\left(\sum_a^{\text{fissile}} \phi \right) / \left(\sum_a^{\text{fissile}} \phi \right)_{\text{ref}}$	1.098	1.144

As indicated by the data in Table 4, the primary penalty involved in using a "denatured" fuel in the reference LMFBR is a significant decrease in the breeding ratio. Although this decrease will adversely affect the reactor's doubling time, it is somewhat compensated by a reduction in the fissile inventory required for the denatured case (~400kg). Calculations have also shown that use of the denatured fuel has a favorable effect on the sodium void coefficient. As indicated above, the heterogeneous case was considered in an effort to increase the ^{233}U production. As shown in Table 5, the heterogeneous concept results in a dramatic increase in ^{233}U production. However, as expected, the heterogeneous model requires a higher fissile inventory. The overall conclusion of this initial effort was that although an LMFBR could operate on denatured fuel, it is probably not possible as a stand-alone reactor. Since even the heterogeneous model requires some fissile feed (i.e., the ^{233}U breeding ratio is <1) a source of ^{233}U must be available to sustain the reactor. Therefore, systems such as that illustrated in Figure 1 appear to merit investigation. The denatured reactors would operate outside the safeguard area in dispersed locations. All fuel reprocessing would be confined to the safeguard area. Also located within the safeguard area would be reactors designed to convert Pu (which is not allowed outside the safeguard area after reprocessing) into ^{233}U (e.g., a Pu-Th LMFBR). Work in this vein is continuing, especially to determine the number of "outside" reactors capable of being supported per "inside" reactor.

B. Methods and Data Preparation and Verification

The second major effort of the fast reactor portion of the TAP was directed at development of a consistent set of nuclear data for the LMFBR and toward automating the various analysis techniques to be utilized. Working from a

pseudo-composition independent library, a 51 group "working" library was generated using LMFBR spectra. This library will be utilized for static ("snapshot") calculations at beginning of life, mid-equilibrium, etc. Additionally, a small computer program was written to produce nuclear data in a format suitable for burnup calculations from the above library; thus maintaining consistency between the static and burnup calculations. Further, due to the multiple types of fuel and/or blanket compositions to be considered in the TAP, modifications were made to the basic computational code being used (CITATION) to automate the required calculations. This effort consisted primarily of developing a procedure to determine the proper equilibrium refuel composition once the type of fuel is selected. Completion of the nuclear data libraries and further code modifications will allow various LMFBR Thorium-based fuel options to be investigated on a production basis.

IV. Scenarios, Objectives and Criteria

The following listing of scenarios, objectives, and criteria was determined on the basis of initial ORNL, ERDA, and BNL considerations and represents a current projection of the scope and purpose of the ORNL portion of the Thorium Assessment Program (TAP). The program will be finalized later, in consultation with ERDA and BNL.

FIGURE 1. FLOW DIAGRAM OF DENATURED LMFBR FUEL CYCLE

SUGGESTED SCENARIOS, OBJECTIVES AND CRITERIA
FOR SAFEGUARDED-RELATED THORIUM ASSESSMENT STUDIES

<u>Restricted Fuel Cycle Scenarios</u>	<u>Objectives</u>	<u>Criteria</u>
1. Throwaway Cycle - HTGR	1. Determine effects of employing HTGR on power cost and fuel resources under no-reprocessing conditions.	1. Comparison of HTGR parameters with LWR parameters.
2. Safeguarded reprocessing and non-denatured cycles with dispersed and denatured cycles.	2. Create a symbiotic system which would maximize resource utilization and power dispersal for the reactor mix considered; determine neutronic, capital cost, and safety implications.	2. Create safeguarded nuclear fuel cycles which compete with coal over the long-term, and which are consistent with the nuclear fuel resource base. Maximize U_{233} component over Pu component. Maximize power production in dispersed reactors.
a. All LWR symbiotic system	a. For current, all LWR	
b. LMFBR + LWR symbiotic systems	b. Introduction of the LMFBR	
c. LMFBR, HTGR, HWR, LWR + LWR, HTGR, HWR symbiotic systems	c. Introduction of HTGR, HWR.	

APPENDIX

Table A-1. Miscellaneous Fuel Cycle Design Parameters

Fuel lifetime (nominal)^{2,3} - 4 yr at 80% capacity factor

Average power density^{2,3} - 8.4 w/cc of active core

Carbon/Thorium ratio:⁴

Initial - 214

Reload - 238

Fresh fuel isotopic composition (wt%)

²³⁵U - 93.15 ± 0.15 (Ref. 5)

²³⁴U - 0.95*

²³⁶U - 0.21*

²³⁸U - 5.69*

Core height - 634.4 cm

Diameter (equivalent) - 845.2 cm

Volume - 355,917.5 l

Volume fractions (cold, reactor averaged)

Fuel element gap** - 0.01396

Fuel compact - 0.19591

Voids around fuel rods† - 0.01546

Coolant - 0.20173

Burnable poison - 0.00634

Fuel handling pickup - 0.00541

Reserve shutdown - 0.00927

Control rod channels - 0.02110

Graphite, main moderator - 0.53082

*Estimates, assumed applicable in 1973.

**Element to element clearance = 0.1 in.

†Clearance plus space between rods and end plug.

Table A-2. Fuel element and control fuel element data⁶

<u>Fuel element information, per element</u>			
Shape - hexagonal, 8.182 in. along sides, 31.22 in. long			
Coolant passages:	<u>Number</u>	<u>Diameter (in.)</u>	
	66	0.828	
	6	0.719	
Burnable poison	6	0.500	
			<u>Length (in.)</u>
Fuel hole:*	10	0.625	26.81
	114	0.625	29.24
<u>Control fuel element information, per element</u>			
Shape - hexagonal, 8.182 in. along sides, 31.22 in. long			
Coolant passages:	<u>Number</u>	<u>Diameter (in.)</u>	
	33	0.828	
	10	0.719	
Reserve shutdown channel:	1	3.75	
Control rod channel:	2	4.00	
Burnable poison:	4	0.500	
			<u>Length (in.)</u>
Fuel hole:*	18	0.625	26.81
	58	0.625	29.24

*Fuel rod OD - cold was 0.620 in.

Fuel handling pickup hole was assumed to have an effective length of 15 in. x 1.59 in. diameter in each element type.

Table A-3. Fuel particle information

<u>I. Fissile Particle (TRISO, UC₂)</u>		
	<u>Diameter/Thickness (μM)^{6,7}</u>	<u>Density (gm/cc)</u>
Kernel (dia)	200	11.18 (as UC ₂) (Ref. 3)
Buffer	85	1.25 (Ref. 8)
Inner Iso. Py. C.	25	1.825 (Ref. 8)
Si C	25	3.19 (Ref. 8)
Outer Iso. Py. C.	35	1.825 (Ref. 8)
 <u>II. Fertile Particle (BISO, TH C₂)</u>		
	<u>Diameter/Thickness (μM)^{6,7}</u>	<u>Density (gm/cc)</u>
Kernel (dia)	450	8.65 (as Th C ₂) (Ref. 3)
Buffer	65	1.25 (Ref. 8)
Outer Iso. Py. C.	75	1.825 (Ref. 8)
 <u>III. Miscellaneous Fuel Compact Data:</u>		
Binder and matrix density - 0.85 gm/cc (Ref. 9)		
Density inert C fill particles - 1.47 gm/cc		

REFERENCES

1. Colby, Dahlberg and Jaye, HTGR Fuel and Fuel Cycle Summary Description, GA-10233 (May 25, 1971).
2. Private communication (computer printout from GAC's GACOST code) R. H. Broglia to E. J. Allen, July 1976.
3. Letter from K. E. Asmussen (GAC) to W. E. Thomas (ORNL), Sept. 22, 1971.
4. Private communication, K. E. Asmussen (GAC) to W. E. Thomas (ORNL), Jan. 26, 1972.
5. Specifications for Coated Fuel Particles for Fort St. Vrain - First Core; Spec No. 18-R-13, Issue H, Jan. 5, 1971.
6. Letter from K. E. Asmussen (GAC) to W. E. Thomas (ORNL), Jan. 5, 1971.
7. Private communication, John Coobs (ORNL) to W. E. Thomas (ORNL), Jan. 21, 1971.
8. Quarterly Progress Report for Period Ending March 31, 1971, HTGR Base Program, General Atomic Company, GA-10501.
9. Quarterly Progress Report for Period Ending March 31, 1971 - Fort St. Vrain R&D Program, General Atomic Company, GA-10560.

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