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FILE

**PREDICTED DISCHARGE PLUTONIUM
ISOTOPICS FOR LEU TEST PEBBLE
IRRADIATED IN THE AVR**

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GENERAL ATOMICS

MASTER

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1. INTRODUCTION

A Subprogram Plan (Ref. 1) related to the Arbeitsgemeinschaft Versuchsreaktor (AVR) Test Program is in place and describes cooperative work being carried out under the United States/Federal Republic of Germany (US/FRG) Implementing Agreement for Cooperation in Gas-Cooled Reactor Development. The AVR information to be provided as described in the plan will provide a basis for examining the accuracy of computational methods used for performance and safety analysis.

The purpose of the cooperation is to obtain experimental information from the AVR relevant to the performance and safety of modular gas-cooled reactors, and to compare measured results with predictions of analytical tools.

Project Work Statement (PWS) No. AVR-4 of the plan titled "Measurement and Computation of Plutonium Buildup in Selected LEU Elements of AVR" has the objectives of measuring the plutonium buildup in selected low-enriched uranium (LEU) fuel elements and to examine the adequacy of computational tools used to predict the plutonium buildup result.

This report provides a progress report on the prediction of plutonium buildup in LEU fuel in a high-enriched uranium (HEU) core and also describes the method for calculating the U-238 resonance integral (cross section).

The plutonium buildup test was carried out in AVR in which an LEU-fueled pebble (containing 6 g of 16.7% enriched uranium) was inserted into the central region of the reactor, irradiated for 7280 MWd or 159 reactor equivalent full power days (EFPDs) and, after one pass through the core was recovered. Detailed post-irradiation measurements were

subsequently performed on this irradiated pebble to determine the discharge isotopic composition and burnup, in fissions per initial metal atom (FIMA).

The results of the measured discharge isotopic composition of the test pebble have not been made available to GA, but will be made available from AVR after this progress report is issued. This progress report summarizes the prediction of the test pebble discharge isotopics as a function of the test element burnups (FIMA) so that these isotopic results can be compared at the actual burnup value as determined from the detailed measured isotopic analysis result.

The analysis results to date have been based on a point burnup (zero-dimensional) depletion code model with preliminary estimates of the test ball temperature history during its irradiation and preliminary available best estimates regarding the burnup and composition of the mostly HEU/Th recycled pebbles that surrounded the test ball in its traverse through the central, high burnup region, of the AVR.

The plutonium buildup isotopics are sensitive to temperature history cross sections, and to the actual burnup of the surrounding HEU/Th pebbles (i.e., the local flux spectrum distribution). This latter effect is very important as deduced from the results of the LEU test element exposure calculations. Prior to this analysis, the assumption of the LEU test pebble burnup had been based on the assumption that the LEU test element was inserted into a mix of recycled HEU pebbles having a prior burnup of 12% FIMA, based on calculated values obtained from fuel depletion model results calculated at the AVR.

The actual measured burnup values for the recycled HEU/Th pebbles recycled into the central region in parallel with the LEU test element showed a measured FIMA of ~14%.

The average measured LEU pebble FIMA, based on several measurements, was ~3.6% FIMA. Our analyses treated the LEU discharge FIMA as a variable with the plutonium buildup calculated as a function of the test element FIMA. One specific value calculated based on the 14% HEU pebble burnup assumption yielded a predicted LEU test element FIMA of 3.67% and that result is the current best estimate for this progress report result.

2. SUMMARY AND CONCLUSIONS

Plutonium buildup calculations for an LEU test element irradiated in AVR to a burnup of ~3.6% FIMA have been completed based on point depletion calculations. Temperature and composition dependent cross sections in a nine neutron energy broad group structure were utilized in these plutonium buildup measurements.

The plutonium isotopics were calculated as a function of the LEU test element FIMA for three different assumed exposure levels (FIMA) for the recycled HEU/Th pebbles surrounding the LEU test pebble. The HEU/Th FIMA values assumed were 11%, 12%, and 14% with 12% being representative of calculations made earlier at AVR with their existing two-dimensional AVR fuel depletion model, which was subsequently determined to underestimate the pebble in core residence time and burnup. The 14% HEU pebble FIMA value represents the measured FIMA value of recycled HEU pebbles inserted about the time of the LEU test pebble insertion into AVR.

The FIMA achieved in the LEU test pebble is sensitive to the burnup of the surrounding HEU recycled pebbles, and for a given LEU test pebble FIMA, the discharged LEU test element plutonium content is also a function of the HEU pebble FIMA since the neutron spectrum is "softer" at higher HEU pebble burnups. For example, at ~3.6% FIMA on the LEU pebble, the Pu-239 at discharge is about 15% lower if 14% FIMA HEU is assumed as compared to 12% FIMA HEU burnup.

The plutonium buildup for a number of different LEU test element FIMAs has been calculated and summarized in this progress report. One value that resulted from these several calculations yielded an LEU test element FIMA of 3.67% for an HEU FIMA assumption of 14%. That estimate is very close to the measured FIMA of 3.6% used as the current reference

prediction. Table 2-1 summarizes the predicted LEU test pebble discharge heavy metal contents for that case.

The current reference plutonium buildup predictions were based on zero-dimensional point depletion estimates and did not include any spatial flux effects and/or leakage effects on the spectrum predicted for the center of the AVR. Two-dimensional models are being prepared to calculate the AVR rod worths and temperature coefficients. When these results become available an assessment will be made as to the accuracy of these zero-dimensional flux spectrum results. If spatial effects and/or leakage effects result in a change in the spectrum from that predicted by zero-dimensional calculation, we will repeat our plutonium buildup estimate.

TABLE 2-1
PREDICTED TEST ELEMENT HEAVY METAL
LOADING AND FIMA FOR 14% FIMA
HEU/Th PEBBLES AROUND TEST ELEMENT

Test Pebble Heavy Metal Discharge Mass (mg)	
3.67% FIMA	
U-235	748.71
U-236	41.42
U-238	4969.55
Pu-238	0.01
Pu-239	16.21
Pu-240	4.14
Pu-241	0.67
Pu-242	0.06

3. METHOD OF ANALYSIS

3.1. CROSS SECTIONS AND CODE MODELS

The predicted buildup and relative concentration of the various plutonium isotopes at discharge have been calculated based on the standard GA nine neutron group structure (five fast, four thermal groups) that has been used for the Modular High-Temperature Gas-Cooled Reactor (MHTGR) physics analysis. The nine broad group cross sections were generated with the two region MICROX Code (Ref. 2). The point depletion calculations to predict the plutonium buildup were performed with the GARGOYLE (Ref. 3) zero-dimensional burnup code. Both of these codes, with the same nine group cross sections, are also used for MHTGR physics analysis.

The nine neutron group structure is shown in Table 3-1. The major resolved resonances for Th-232 are in group 3 and the resolved resonances for U-238 are in both group 3 and group 4. The thermal group structure has been selected for LEU fuel analysis to adequately handle the thermal cross sections for both Pu-239 and Pu-240.

The cross sections for both the LEU test pebble and the surrounding irradiated HEU/Th pebbles were generated with the two-region flux spectrum MICROX Code. Cross sections were generated based on the GA reference set, which is a mixture of ENDF/B-I, II, and III data, as well as for the current ENDF/B-V data. Most buildup calculations performed utilized the ENDF/B-V cross section base, although selected cases were performed with the GA base set and differences in the plutonium buildup due to differences in the cross section data base were minor as was expected from MHTGR correlations.

TABLE 3-1
NEUTRON ENERGY GROUP STRUCTURE

Group No.	Lower Energy Boundary (eV)
1	1.83×10^5
2	961.0
3	17.61
4	3.93
5	2.38
6	1.275
7	0.825
8	0.13
9	0.0

The two region MICROX cross sections were generated and included the grain shielding effects in region 1 (the 5 cm o.d. fueled portion of the overall 6 cm o.d. pebble sphere) for both the LEU test ball and for the surrounding HEU/Th irradiated fuel pebbles at several fuel temperatures. The LEU test pebble cross section was based on a MICROX cell with region 1 corresponding to the 5 cm fueled diameter surrounded by the homogenized equivalent of 20 HEU pebbles plus associated void. A Dancoff factor of $C = 0.0$ was input for this case for determining the U-238 resonance cross section.

An infinite array of depleted HEU/Th fuel pebbles was modeled in MICROX to determine the overall cross sections for the GARGOYLE depletion calculations for the reactor loading except for the single LEU test pebble. In this case, region 1 corresponded to the 5 cm o.d. fueled portion of the HEU/Th pebble and region 2 corresponded to the 0.5 cm thick unfueled coating and the associated void between pebbles. In this case, the Th-232 resonance cross sections, and the low density U-238 in the HEU-fueled pebbles were based on the estimated mutual shadowing among the close-packed HEU pebbles and a Dancoff factor of $C = 0.442$ was calculated and specified in the MICROX model.

3.2. HEU/Th PEBBLE EXPOSURE EFFECTS

Most of the preliminary estimates of the LEU test element FIMA values and resultant plutonium buildup values were based on the assumption, as calculated at AVR, that the surrounding irradiated HEU balls being recycled at the time of the LEU test element insertion had achieved a 12% FIMA.

Calculations performed at AVR indicate that their detailed fuel depletion model used in 1986 underestimates, by about 15%, the total transit time, and thus the pebble burnup, for a typical pebble traverse through the inner core. This was verified when the LEU test pebble took about 15% more time to traverse the core than had been predicted based

on the RZ fuel flow and fuel depletion model. A detailed listing of the measured burnup data of the HEU pebbles recycled to the inner zone at the time of the test pebble insertion on January 18, 1986, was received at GA on June 17, 1988 and indicated that the actual HEU pebble burnup was ~14% FIMA. Additional GARGOYLE depletion calculations were performed for the 14% FIMA HEU pebble exposure assumption.

GARGOYLE depletion calculations were actually performed for 11%, 12%, and most recently, for the measured 14% FIMA HEU pebble burnup assumption. In each case, the reference 159 reactor EFPD LEU pebble exposure was calculated, as well as a lower and a higher EFPD assumption such that the LEU pebble plutonium isotopics were generated at three or more different LEU pebble FIMA values for each of the three calculated HEU burnups for the HEU pebbles circulating around and with the LEU test pebble.

Table 3-2 summarizes the matrix of plutonium buildup calculations performed. The actual FIMA for the LEU pebble cannot be accurately predicted, so the calculated plutonium isotopics for the five LEU FIMA results in the presence of 14% FIMA HEU pebbles, should be plotted against the predicted LEU pebble FIMA and read off the plot at the actual measured value of FIMA for the LEU test pebble. The purpose of these calculations was to predict the plutonium buildup at the measured LEU pebble FIMA value and not to predict the actual FIMA value of the LEU test pebble.

As discussed in Section 2, the measured LEU test element exposure was ~3.6% FIMA (Ref. 4). This value was measured in the AVR discharge measuring device(s). The results of detailed isotopic measurements will be provided by AVR later. If the 3.6% FIMA is the actual value, then our calculated point at 3.67% FIMA, for 14% FIMA HEU exposure, is the best "reference" prediction. If the actual FIMA is different, then the predicted value should be inferred from the plot of the results calculated.

TABLE 3-2
SUMMARY OF PLUTONIUM BUILDUP GARGOYLE CASES

HEU FIMA (%)	EFPDs Exposure				
	84	108	144	159	174
	LEU Pebble FIMA (%)				
11	--	--	(3.1)	(3.5)	(3.8)
12	--	(2.7)	(3.6)	(3.9)	(4.3)
14	(2.9)	(3.7)	(4.9)	(5.3)	(5.8)

Note: AVR core average power density of 2.6 W/cc assumed in these cases.

3.3. FUEL TEMPERATURE ESTIMATES

Another important factor in determining the plutonium buildup is the fuel temperature history in the LEU test pebble as it transverses down the central region of the AVR core. An estimate of the test pebble temperature as a function of axial position was made and is shown in Fig. 3-1. This estimate assumed complete gas mixing in the AVR and was based on the average core power density of 2.6 W/cc for all pebbles, except the LEU test pebble, and the reference average inlet and outlet gas temperatures of 265° and 950°C, respectively. The LEU test element was assumed to have an age peaking factor (APF) of 2.5 times the core average power density in this temperature estimate. This test element temperature estimate will be updated in future calculations, if additional information regarding the expected outlet temperature distribution out of the central radial region of the core becomes available and differs from our assumption of 950°C. This temperature estimate included the localized 2.5 APF estimate for the test pebble based on our earlier assumption of a lower HEU pebble burnup. A slightly higher temperature estimate would result for the case of 14% FIMA in the surrounding HEU recycle pebbles.

The predictions considered two predicted test element fuel temperatures corresponding to 1060°C in the top one-half of the pebble pass, and 582°C in the bottom one-half of the core. The test pebble downward flow is slower in the top one-half of the core. This results in about two-thirds of the exposure at the higher (1060°C) temperature followed by one-third of the exposure at the lower (582°C) temperature. This two temperature time history was assumed in the GARGOYLE calculations, in which the LEU U-238 resonance cross section was changed to the lower temperature value after two-thirds of the assumed LEU pebble time exposure was achieved in the burnup calculation.

The high burnup HEU/Th pebble temperature in the top half of the core was estimated to be ~220°C lower than the test pebble temperature, and ~70°C lower in the bottom half of the core.

Fig. 3-1
TEST PEBBLE TEMPERATURE
AS A FUNCTION OF AXIAL POSITION

4. ANALYSIS RESULTS

4.1. IMPACT OF CROSS SECTION DATA BASE

The nine group cross sections for U-235, U-238, Pu-239, and Pu-240 for the GA reference data set are shown in Table 4-1, and the same cross sections based on the ENDF/B-V data base are shown in Table 4-2. These data were calculated based on an assumed LEU fuel test pebble temperature of 1060°C in both cases.

Table 4-3 shows the plutonium isotopics based on the two cross section sets, and as would be expected from comparing the cross sections given in Tables 4-1 and 4-2, there are only minor differences in the plutonium buildup between the two data base sets. Other than this comparison, the remaining results discussed below are based on the use of the ENDF/B-V data base.

4.2. U-238 RESONANCE CROSS SECTIONS

Table 4-4 summarizes the group 3 and group 4 test element U-238 resonance cross section used in the nine group depletion calculations at the two estimated fuel temperatures discussed earlier. In all buildup calculations, it was assumed that the high temperature value was used for the first two-thirds of the irradiation time, and the low temperature value was used for the last third of the assumed irradiation interval.

4.3. TEST ELEMENT LOADING

The LEU test pebble description and loading data at time of insertion into the AVR is summarized in Table 4-5.

TABLE 4-1
TEST PEBBLE CROSS SECTIONS FROM GA DATA BASE
(1060°C)

Group	ν	σ_f	σ_c	σ_{tr}
U-235				
1	2.56650+00	1.35677+00	1.36624-01	5.93809+00
2	2.42512+00	3.56664+00	1.37771+00	1.52079+01
3	2.42302+00	2.66946+01	1.40655+01	5.19514+01
4	2.42300+00	2.69810+01	3.29460+01	7.95258+01
5	2.42300+00	2.25949+01	9.43935+00	4.25247+01
6	2.42300+00	1.61694+01	8.07264+00	3.61356+01
7	2.42300+00	6.47549+01	1.30394+01	9.02597+01
8	2.42300+00	1.47935+02	3.09723+01	1.92956+02
9	2.42300+00	3.05883+02	5.33904+01	3.65077+02
U-238				
1	2.74797+00	1.79222-01	1.08466-01	6.15199+00
2	2.33781+00	9.36788-06	7.98565-01	1.48506+01
3	0.00000	0.00000	1.64184+01	4.45905+01
4	0.00000	0.00000	3.53123+01	4.44864+01
5	0.00000	0.00000	5.75988-01	9.02182+00
6	0.00000	0.00000	4.84983-01	9.12885+00
7	0.00000	0.00000	5.11024-01	9.08951+00
8	0.00000	0.00000	8.38594-01	9.48338+00
9	0.00000	0.00000	1.59153+00	1.00561+01
Pu-239				
1	3.05965+00	1.90049+00	7.89851-02	6.00743+00
2	2.88327+00	2.43402+00	1.53464+00	1.49621+01
3	2.88002+00	2.26520+01	2.12015+01	6.12889+01
4	2.88000+00	6.65703+01	4.71308+01	1.27035+02
5	2.88000+00	9.65850+00	7.36943-01	2.26865+01
6	2.88000+00	1.89220+01	4.68723+00	3.58943+01
7	2.88000+00	3.50753+01	1.37003+01	6.06718+01
8	2.88000+00	1.11332+03	7.39232+02	1.85970+03
9	2.88000+00	5.07223+02	2.33240+02	7.42137+02
Pu-240				
1	3.06662+00	1.04340+00	1.25066-01	5.96542+00
2	2.89287+00	1.15121-01	1.00445+00	1.46819+01
3	2.89002+00	2.78832-01	2.79760+01	6.79075+01
4	2.89000+00	6.72071-04	7.68648-01	1.19623+01
5	2.89000+00	2.05255-03	8.19650+00	2.61007+01
6	2.89000+00	2.97082-02	1.52472+02	1.99526+02
7	2.89000+00	3.30254+00	1.72794+04	1.86559+04
8	2.89000+00	3.60728-02	1.85228+02	1.88128+02
9	2.89000+00	3.67282-02	1.84869+02	1.85287+02

TABLE 4-2
TEST PEBBLE CROSS SECTIONS FROM ENDF/B-V
(1060°C)

Group	ν	σ_f	σ_c	σ_{tr}
U-235				
1	2.57150+00	1.34892+00	1.49051-01	5.87551+00
2	2.43771+00	3.32069+00	1.26972+00	1.53018+01
3	2.43670+00	2.62162+01	1.36093+01	5.19006+01
4	2.43670+00	3.79308+01	3.24322+01	8.07870+01
5	2.43670+00	2.28522+01	9.23787+00	4.34087+01
6	2.43670+00	1.57926+01	7.93749+00	3.59087+01
7	2.43670+00	6.53840+01	1.30722+01	9.10745+01
8	2.43670+00	1.48907+02	3.05469+01	1.92854+02
9	2.43670+00	3.08064+02	5.27337+01	3.65620+02
U-238				
1	2.76025+00	1.86120-01	9.87129-02	6.12867+00
2	2.41850+00	5.37822-05	7.40259-01	1.55888+01
3	0.00000	0.00000	1.65037+01	4.70995+01
4	0.00000	0.00000	3.47186+01	4.43398+01
5	0.00000	0.00000	5.49578-01	8.82977+00
6	0.00000	0.00000	4.65017-01	9.06567+00
7	0.00000	0.00000	4.97603-01	9.03173+00
8	0.00000	0.00000	8.29578-01	9.42393+00
9	0.00000	0.00000	1.58007+00	9.99692+00
Pu-239				
1	3.08152+00	1.88951+00	7.68702-02	6.11081+00
2	2.88217+00	2.49402+00	1.48127+00	1.49839+01
3	2.87976+00	2.32052+01	2.11205+01	5.85640+01
4	2.87975+00	6.76109+01	4.35743+01	1.20340+02
5	2.87975+00	1.11344+01	1.03955+00	2.13071+01
6	2.90557+00	1.96044+01	5.19012+00	3.45133+01
7	2.90557+00	3.54770+01	1.36765+01	5.92594+01
8	2.90557+00	1.10865+03	7.34659+02	1.84968+03
9	2.90557+00	5.07333+02	2.31854+02	7.40255+02
Pu-240				
1	3.07921+00	1.13480+00	1.27981-01	6.04791+00
2	2.80738+00	9.74988-02	1.32360+00	1.48393+01
3	2.80305+00	1.24142-01	3.32572+01	6.79283+01
4	2.80300+00	7.29702-04	8.12154-01	9.66711+00
5	2.80300+00	2.00130-03	8.32359+00	2.26188+01
6	2.80300+00	2.94331-02	1.59922+02	1.99602+02
7	2.80300+00	2.92964+00	1.62553+04	1.73352+04
8	2.80300+00	3.54172-02	1.88577+02	1.89942+02
9	2.80300+00	3.62624-02	1.85571+02	1.84064+02

TABLE 4-3
COMPARISON OF PLUTONIUM ISOTOPICS
BASED ON ENDF/B-V DATA SET AND
BASED ON GA REFERENCE DATA SET

	Weight (milligrams/ball)	
	ENDF/B-V	GA Reference
Pu-239	19.79	19.85
Pu-240	5.33	5.38
Pu-241	1.04	1.10
Pu-242	0.10	0.10

TABLE 4-4
U-238 RESONANCE CROSS SECTIONS AT
TWO TEST PEBBLE OPERATING
TEMPERATURES

	Sigma Capture, Barns	
	T = 1060°C	T = 582°C
Group 3(a)	16.5	15.1
Group 4(a)	34.7	31.6

(a) Based on ENDF/B-V data.

TABLE 4-5
LEU TEST ELEMENT DESCRIPTION
AND LOADING

Overall diameter, cm	6.0
Fueled region diameter, cm	5.0
Uranium total, gms	6.0
U-235, gms	1.0
U-235 enrichment, %	16.7
Fuel particle type	
Kernel diameter, μm	500.0
Outer coating diameter, μm	922.0

4.4. TEST ELEMENT FIMA RESULTS

The burnup (FIMA) of the LEU test element is a strong function of the burnup (FIMA) of the surrounding irradiated recycle pebbles as was summarized in Table 3-2 earlier. Figure 4-1 shows this relationship as determined from the point burnup calculations assuming burnup for the indicated EFPDs of exposure occurred at the AVR core average power density of 2.6 W/cc.

The FIMA of the test element is seen to increase dramatically for a given EFPDs exposure as the FIMA of the surrounding HEU/Th fuel is increased.

4.5. PLUTONIUM BUILDUP RESULTS

The calculated heavy metal discharge weights (in milligrams) for the various test pebble FIMAs are summarized in Tables 4-6 through 4-8 for the 11%, 12%, and 14% HEU FIMA values, respectively.

The discharge data for the "reference" 14% HEU FIMA burnup assumption is considered to be the reference or best estimate prediction for the plutonium buildup.

The discharge quantities of U-235 and U-238 versus LEU test element FIMA for the 14% HEU FIMA cases are plotted in Fig. 4-2. The data for the plutonium isotopics is plotted in Fig. 4-3.

The sensitivity of the discharge plutonium isotopics to the assumed burnup of the surrounding HEU fuel elements is illustrated in Fig. 4-4, where the Pu-239 and Pu-240 compositions are plotted against the test element FIMA both for 12% and 14% FIMA HEU element burnup assumptions. The assumption of a higher FIMA in the HEU pebbles leads to a softer (more thermalized) neutron spectrum and an appreciable reduction in the plutonium buildup for a given test element FIMA.

Fig. 4-1
TEST PEBBLE FIMA VS. ASSUMED EFPD EXPOSURE

TABLE 4-6
PREDICTED TEST ELEMENT HEAVY METAL LOADING AND FIMA FOR
11% FIMA HEU/Th PEBBLES AROUND TEST ELEMENT

	Test Pebble Heavy Metal Discharge Mass (mg)		
	3.13% FIMA	3.46% FIMA	3.78% FIMA
U-235	786.03	764.76	743.69
U-236	36.02	39.54	43.02
U-238	4967.43	4963.92	4960.38
Pu-238	0.01	0.02	0.02
Pu-239	19.06	20.06	20.93
Pu-240	3.99	4.67	5.37
Pu-241	0.66	0.85	1.06
Pu-242	0.05	0.07	0.10

TABLE 4-7
 PREDICTED TEST ELEMENT HEAVY METAL LOADING AND FIMA FOR
 12% FIMA HEU/Th PEBBLES AROUND TEST ELEMENT

	Test Pebble Heavy Metal Discharge Mass (mg)			
	2.67% FIMA	3.55% FIMA	3.92% FIMA	4.28% FIMA
U-235	816.05	757.76	733.96	710.46
U-236	30.82	40.45	44.36	48.21
U-238	4974.25	4965.36	4961.59	4957.81
Pu-238	0.01	0.02	0.02	0.03
Pu-239	16.13	18.92	19.79	20.53
Pu-240	2.87	4.58	5.33	6.10
Pu-241	0.39	0.81	1.04	1.28
Pu-242	0.02	0.07	0.10	0.14

TABLE 4-8
 PREDICTED TEST ELEMENT HEAVY METAL LOADING AND FIMA FOR
 14% FIMA HEU/Th PEBBLES AROUND TEST ELEMENT

	Test Pebble Heavy Metal Discharge Mass (mg)				
	2.87% FIMA	3.67% FIMA	4.85% FIMA	5.32% FIMA	5.79% FIMA
U-235	801.85	748.71	672.24	641.58	611.65
U-236	32.74	41.42	53.81	58.76	63.55
U-238	4976.54	4969.55	4958.84	4954.29	4949.70
Pu-238	0.01	0.01	0.03	0.05	0.06
Pu-239	14.14	16.21	18.22	18.77	19.19
Pu-240	2.74	4.14	6.34	7.26	8.15
Pu-241	0.35	0.67	1.33	1.66	2.00
Pu-242	0.02	0.06	0.18	0.25	0.34

Fig. 4-2
DISCHARGE URANIUM WEIGHTS
VS. TEST PEBBLE FIMA (14% HEU FIMA)

Fig. 4-3
DISCHARGE PLUTONIUM WEIGHTS
VS. TEST PEBBLE FIMA (14% HEU FIMA)

Fig. 4-4
DISCHARGE PU-239 & PU-240 WEIGHTS
VS. TEST PEBBLE FIMA

5. UNCERTAINTIES IN RESULTS

There are several uncertainties relating to both the experiment and in the predicted results. The actual temperature history of the irradiated test pebble is not known nor is its actual path through the inner core region known. The predicted temperature was based on an earlier, and lower HEU pebble FIMA power assumption, and the assumption that the gas outlet temperature in the central region was 950°C, due to a perfect gas mixing assumption.

This preliminary zero-dimensional analysis also did not include any spectral effects due to net inleakage (or outleakage) from the central test zone. This leakage effect is probably small near the actual center of the inner zone but if the test element traversed at the outer portion of the inner zone there would be strong flux gradients near the LEU-fueled outer region of the core and the irradiation spectrum would be much different than assumed.

The possible uncertainties in the plutonium buildup due to these and other effects, have not been quantified. Such uncertainty analyses were beyond the scope of this task. However, from the results predicted and given in this report, certain trends and conclusions have been identified that could prove useful for understanding the probable cause(s) for differences between the predicted values and the measured values if they differ significantly.

6. REFERENCES

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