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NUCLEAR MASS INVENTORY, PHOTON DOSE RATE AND THERMAL DECAY HEAT OF SPENT RESEARCH REACTOR FUEL ASSEMBLIES (Rev. 1)

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REVISION-1 RECORD

ANL/RERTR/TM-26 (original issue, May 1966) has been revised as follows:

Table 5 has been revised to increase the burnup range of TRIGA single-rod fuel from 35 to 60% ^{235}U burnup.

Alternative thermal decay heat expressions have been included which are expected to give results close to actual heat loads of spent fuel. The previous thermal decay heat expression is expected to overestimate an actual heat load by about a factor of two. An analysis of the parameter constants used in the previous expression would suggest an uncertainty in calculated heat loads of the order of 10%. A decay heat comparison has been made for a typical fuel assembly using the ORIGEN code and the decay heat expressions.

Appendix C has been added which compares mass inventory estimates using the isotope generation and depletion code, ORIGEN and the cross section generation code, WIMS. Both codes solve material transmutation equations to determine material number densities. WIMS, however, solves the equations as a function of material burnup, while ORIGEN does not have a similar capability.

Appendix D has been added to illustrate an example calculation of the nuclear mass inventory, the photon dose rate, and the thermal decay heat for an assumed, spent MTR-type fuel assembly. All fuel assembly parameters necessary for the calculations are described.

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SUMMARY

This document has been prepared to assist research reactor operators possessing spent fuel containing enriched uranium of United States origin to prepare part of the documentation necessary to ship this fuel to the United States. Data are included on the nuclear mass inventory, photon dose rate, and thermal decay heat of spent research reactor fuel assemblies.

Isotopic masses of U, Np, Pu and Am that are present in spent research reactor fuel are estimated for MTR, TRIGA and DIDO fuel assembly types. The isotopic masses of each fuel assembly type are given as functions of U-235 burnup in the spent fuel, and of initial U-235 enrichment and U-235 mass in the fuel assembly.

Photon dose rates of spent MTR, TRIGA and DIDO-type fuel assemblies are estimated for fuel assemblies with up to 80% U-235 burnup and specific power densities between 0.089 and 2.857 MW/kg^{235}U, and for fission product decay times of up to 20 years.

Thermal decay heat loads are estimated for spent fuel based upon the fuel assembly irradiation history (average assembly power vs. elapsed time) and the spent fuel cooling time.

INTRODUCTION

As part of the Department of Energy's spent nuclear fuel acceptance criteria, the mass of uranium and transuranic elements in spent research reactor fuel must be specified. These data are, however, not always known or readily determined. It is the purpose of this report to provide estimates of these data for some of the more common research reactor fuel assembly types. The specific types considered here are MTR, TRIGA and DIDO fuel assemblies.

The degree of physical protection given to spent fuel assemblies is largely dependent upon the photon dose rate of the spent fuel material. These data also, are not always known or readily determined. Because of a self-protecting dose rate level of radiation (dose rate greater than 100 rem/h at 1 m in air), it is important to know the dose rate of spent fuel assemblies at all time. Estimates of the photon dose rate for spent MTR, TRIGA and DIDO-type fuel assemblies are given in this report.

For safe spent fuel assembly containment, the thermal heat load generated by the decay of fission products in spent fuel material is an important consideration. This heat load can be estimated by a simple analytical expression that is given in this report.

NUCLEAR MASS INVENTORY

The mass inventory of the heavy metals in research reactor fuels has been calculated using the WIMS code[1] for unit-cell models of MTR, TRIGA and DIDO fuel assembly types. Models of each fuel assembly type were neutronically burned for a length of time corresponding to typical fuel-cycle lengths and U-235 burnup[2]. Table 1 summarizes the fuel assembly models for which mass inventory calculations were made.

Table 1. Fuel Assembly Models

Assembly Type	U-235 Burnup, %	U-235 Enrichment, %	U-235 Mass, g
MTR (19 fuel plates)	5, 10, 20, 30, 40, 50, 60, 70, 80	93 45 19.75	100 200 300 400 200 300 400 100 200 300 400 500
TRIGA (single rod)	5, 10, 15, 20, 25, 30, 35, 40, 45, 50, 55, 60	70 (8.5wt% U) 20 (20wt% U) 20 (12wt% U) 20 (8.5wt% U)	133 98 54 38
TRIGA (25 rod cluster)	10, 20, 30, 40, 50, 60	93.1 (10wt% U) 19.7 (45wt% U)	41.4 53.6
DIDO (4 fuel tubes)	10, 20, 30, 40, 50, 60	93 80 60 20	150 150 150 200

Mass inventory calculations for MTR models were made for assemblies with up to 80% U-235 burnup, for 93, 45 and 19.75% U-235 enrichments, and for initial U-235 masses of 100 to 500 g. The specific MTR model was for a 19-fuel plate assembly. (Supplemental mass inventory calculations, shown in Appendix A, indicate that the MTR model in not a strong function of the number of fuel plates or the specific fuel-clad-coolant geometry.)

Similar calculations were made for two TRIGA assembly types – a single rod model and a 25-rod cluster model. The maximum U-235 burnup in these models was 60%. There were four fuel types for the single rod model and two fuel types for the cluster model.

For DIDO fuel assembly types, mass inventory calculations were made for a 4-fuel tube model with up to 60% U-235 burnup, and for four fuel enrichments and assembly masses.

The results of the mass inventory calculations are shown in the following tables:

The tables show the isotopic masses of U, Np, Pu and Am that are present in spent fuel as functions of the fuel assembly U-235 burnup and initial U-235 mass. As will be noted in the tables for most fuel assembly types, the uranium fuel compositions have excluded initial enrichments of U-234 and U-236. In order to account for initial enrichments of U-234 and/or U-236 in the tables, initial U-234 and U-236 masses can be simply added to the spent fuel mass inventory. (See Appendix B for an assessment of the effect of initial enrichments of U-234 and U-236 upon the overall mass inventory of U, Np, Pu and Am in spent fuel.) Within the uncertainty of the calculations, the results in Tables 2–7 can be used to estimate the spent fuel mass inventory in most MTR, TRIGA and DIDO fuel assembly types. (See Appendix C for a comparison of calculational techniques.)

The mass inventories given in Tables 2–7 are at the time of reactor discharge and therefore do not account for decay of Pu-241 to Am-241 for times after discharge. When necessary to estimate mass inventories after discharge, the Pu-241 mass is decreased and the Am-241 mass is increased by an amount $\Delta M = M_0 \cdot (1 - e^{-\lambda t})$ where M_0 is the Pu-241 mass at discharge, $\lambda = 1.32 \cdot 10^{-4}$ d^{-1} (Pu-241 half-life, 14.4 y), and t is the time in days after discharge. No mass inventories are given for U-239 (half-life, 23.5 m) and Np-239 (half-life, 2.355 d) as they are assumed to decay instantaneously to Pu-239.

PHOTON DOSE RATE

Calculated dose rates for MTR-type fuel assemblies are shown in Table 8. These dose rates are from Ref. 3 and are for fuel assemblies with up to 80% U-235 burnup, specific power densities between 0.089 and 2.857 MW/kg^{235}U, and fission product decay times of up to 20 years.

The data in Table 8 are photon dose rates in air that are averaged over a 60-cm long cylindrical surface, located at a radius of 1 m from the fuel assembly axial center line. For MTR-type fuel assemblies, these average dose rates are independent of the assembly rotational orientation and the number of fuel plates in the assembly. These data also can be interpolated for specific decay time, burnup and assembly power density. In all cases, the dose rates must be multiplied by the mass of U-235 burned in the fuel assembly to estimate the fuel assembly dose rate. The mass of U-235 burned per fuel assembly that is necessary for an unshielded, 100 rem/h self-protecting dose rate at 1 m, is shown in Fig. 1.

Additional analyses have shown that the photon dose rates of MTR, TRIGA and DIDO-type fuel assemblies are similar, given the same fuel assembly characteristics of U-235 burnup, fission product decay time, and specific fuel assembly power density. The average dose rates at 1 m in air for TRIGA (25-rod) and DIDO (4-tube) fuel assemblies are respectively, 1.04 and 1.05 times the dose rates given in Table 8 for MTR fuel assemblies. The dose rates of all three fuel assembly types are for fuel assembly models (nominally 8cm by 8cm by 60cm) containing spent fuel in the form of either rods (TRIGA fuel), annuli (DIDO fuel) or plates (MTR fuel). The small difference in the dose rates are due to the different shielding effects of the fuel elements in the fuel assemblies.

Table 8. Photon Dose Rates At 1 M In Air, rem/h per g^{235}U burned

Decay Time, y	Burnup, %^{235}U	Assembly Power Density, MW/kg^{235}U					
		2.857	1.429	0.714	0.357	0.179	0.089
2	1%	1.84+0	1.84+0	1.83+0	1.80+0	1.77+0	1.70+0
3		1.13+0	1.13+0	1.13+0	1.13+0	1.11+0	1.11+0
4		9.01-1	9.01-1	9.01-1	9.01-1	9.01-1	8.92-1
2	10%	1.89+0	1.87+0	1.80+0	1.64+0	1.50+0	1.28+0
3		1.19+0	1.20+0	1.20+0	1.16+0	1.09+0	9.95-1
4		9.52-1	9.61-1	9.61-1	9.44-1	9.10-1	8.59-1
2	20%	2.01+0	1.98+0	1.86+0	1.66+0	1.42+0	1.19+0
3		1.31+0	1.32+0	1.28+0	1.21+0	1.11+0	9.78-1
4		1.04+0	1.05+0	1.04+0	9.99-1	9.44-1	8.63-1
5		8.97-1	9.10-1	9.05-1	8.80-1	8.46-1	7.95-1
10		6.67-1	6.67-1	6.67-1	6.59-1	6.50-1	6.25-1
15		5.78-1	5.78-1	5.74-1	5.70-1	5.61-1	5.44-1
20		5.10-1	5.10-1	5.10-1	5.06-1	4.97-1	4.85-1
2	40%	2.40+0	2.30+0	2.09+0	1.82+0	1.52+0	1.21+0
3		1.62+0	1.60+0	1.53+0	1.39+0	1.22+0	1.02+0
4		1.27+0	1.27+0	1.22+0	1.14+0	1.03+0	8.99-1
5		1.07+0	1.07+0	1.04+0	9.90-1	9.20-1	8.12-1
10		7.03-1	7.03-1	6.95-1	6.80-1	6.55-1	6.10-1
15		5.87-1	5.84-1	5.80-1	5.70-1	5.53-1	5.23-1
20		5.14-1	5.12-1	5.08-1	5.02-1	4.87-1	4.59-1
2	60%	2.95+0	2.79+0	2.52+0	2.15+0	1.74+0	1.34+0
3		2.05+0	2.00+0	1.87+0	1.66+0	1.40+0	1.12+0
4		1.59+0	1.56+0	1.49+0	1.35+0	1.17+0	9.63-1
5		1.30+0	1.29+0	1.24+0	1.15+0	1.02+0	8.54-1
10		7.55-1	7.51-1	7.37-1	7.07-1	6.70-1	6.02-1
15		5.96-1	5.96-1	5.88-1	5.72-1	5.50-1	5.04-1
20		5.17-1	5.17-1	5.13-1	4.99-1	4.76-1	4.39-1
2	80%	3.85+0	3.62+0	3.26+0	2.76+0	2.21+0	1.64+0
3		2.73+0	2.64+0	2.43+0	2.11+0	1.74+0	1.33+0
4		2.08+0	2.03+0	1.90+0	1.69+0	1.41+0	1.12+0
5		1.66+0	1.63+0	1.54+0	1.39+0	1.19+0	9.57-1
10		8.28-1	8.21-1	8.00-1	7.59-1	6.97-1	6.04-1
15		6.18-1	6.15-1	6.05-1	5.82-1	5.44-1	4.87-1
20		5.27-1	5.20-1	5.13-1	4.97-1	4.66-1	4.20-1

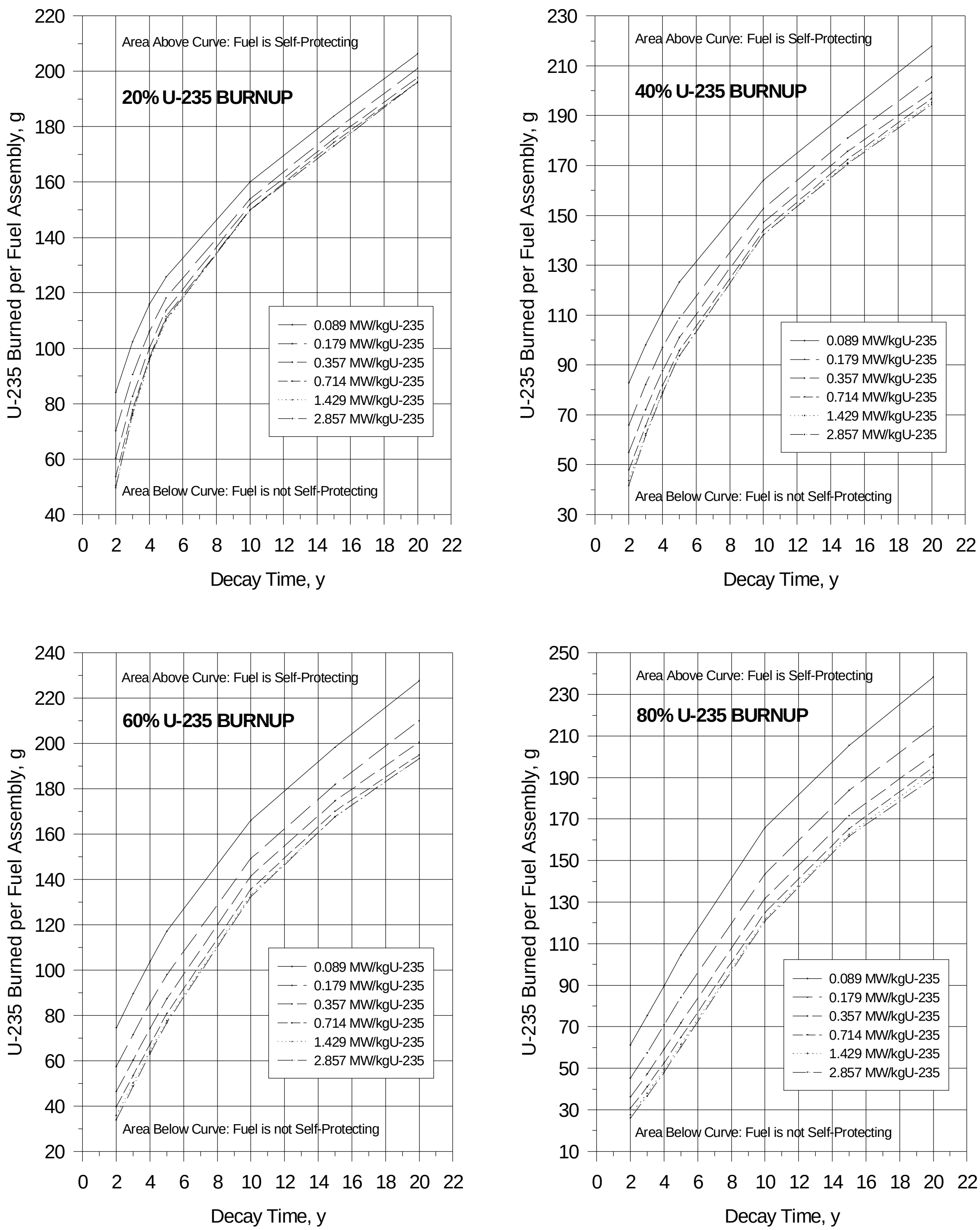

Figure 1. Mass of Burned ^{235}U per Fuel Assembly Necessary for an Unshielded 100 rem/h Dose Rate at 1 m for Fuel Assemblies with 20, 40, 60 and 80% ^{235}U Burnup and Power Densities from 0.089 to 2.857 MW/kg^{235}U

THERMAL DECAY HEAT

The textbook development of calculating the thermal decay heat of reactor fuel is based upon integrating empirical emission rates of the beta and gamma radiation from fission products. These results are, however, generally useful only for fission product decay times of the order of a few days. For longer decay times, this heat load estimate is very conservative.

Other analytical expressions have been developed to fit experimental decay heat data for longer decay times. For purposes of determining the heat load of spent fuel which can have cooling times of the order of several hundred days of even years, these latter expressions should be used to calculate the heat load of spent reactor fuel. These analytical expressions also agree very well with ORIGEN decay heat calculations.

Integrated Beta And Gamma Emission Rates

The heat load from decaying fission products in a fuel assembly is proportional to empirical emission rates of beta and gamma radiation. The rates[4] per U-235 fission, and as a function of decay time t_d in days, are

$$\beta(t_d) = 1.50 \cdot 10^{-6} \cdot t_d{}^{-1.2} \text{ MeV/s-f}$$
$$\gamma(t_d) = 1.67 \cdot 10^{-6} \cdot t_d{}^{-1.2} \text{ MeV/s-f}$$

These energy rates are roughly equal for 0.4 MeV mean energy beta particles and 0.7 MeV mean energy gamma-rays.

For a fuel assembly irradiated continuously for t_i days at a constant fuel assembly power (*P*), the heat (*H*) load power per assembly, t_d days after irradiation is

$$H = 6.85 \cdot 10^{-3} \cdot P \cdot (t_d{}^{-0.2} - (t_i + t_d)^{-0.2}) \text{ Watts} \qquad (1)$$

This expression for the heat load is the integral[5] of the above energy rates over the irradiation time, assuming 200 MeV per U-235 fission, and for the fuel assembly power in watts. For a low duty-factor fuel assembly irradiation, the power and irradiation time are replaced by an average power and an elapsed time. With $\overline{P} \cdot t_e = \sum(P \cdot t_i)$ over all irradiation segments, the heat (*H*) load power per assembly is

$$H \cong 6.85 \cdot 10^{-3} \cdot \overline{P} \cdot (t_d{}^{-0.2} - (t_e + t_d)^{-0.2}) \text{ Watts}$$

where $\overline{P}$ is the average fuel assembly power in watts and t_e is the elapsed time in days from the initial through the final irradiation segment.

A convenient estimate for the average power $(\overline{P})$ is

$$\overline{P} = (G / t_e) / 1.25 \cdot 10^{-6} \text{ Watts}$$

where G is the mass of U-235 burned in the fuel assembly in grams, and the constant is g^{235}U burned per Watt-day.

A similar heat load expression to Eq. -1, given by Etherington (Ref. 6) and attributed to Way and Wigner (Ref. 7 and 8), is

$$H = 6.22 \cdot 10^{-2} \cdot P \cdot ({t_d}^{-0.2} - (t_i + t_d)^{-0.2}) \text{ Watts}$$

with all times in seconds. With all times in days, this heat load expression is

$$H = 6.40 \cdot 10^{-3} \cdot P \cdot ({t_d}^{-0.2} - (t_i + t_d)^{-0.2}) \text{ Watts}$$

(Note, the Etherington reference to Way and Wigner appears to be incorrect. Reference 7 is Vol. 73 (not Vol. 70)[6] of Phys. Rev.; Ref. 8 may be the intended reference. However neither Ref. 7 or 8 appears to have the formula attributed to Way and Wigner. Reference 8 however, lists the same β and $\beta + \gamma$ emission rates used to develop Eq. -1 which results in a heat load expression constant of $6.85 \cdot 10^{-3}$.) This and other similar heat load expressions, which differ only by the constant, can be readily found in the literature (e.g., Ref. 5, $5.7 \cdot 10^{-2}$ for times in seconds and $5.9 \cdot 10^{-3}$ for times in days).

Fuel assembly decay heat loads calculated with these expressions are expected to be conservative, and within a factor of two or less of measured heat loads. This same conservative heat load estimate also has been found to be true for heat load calculations made with the ORIGEN code[9]. The thermal heat load of a fuel assembly is independent of the fuel assembly type.

The constants used in the above equations are based upon empirical data and therefore, are not necessarily exact; it is not uncommon to find several percent variation in a recommended value. The constants considered here, and their range, are:

1. the beta plus gamma fission product energy rate per fission; $2.7 - 3.2 \cdot 10^{-6}$ MeV/s-f,
2. the total energy release per fission; 190 - 200 MeV/f, and
3. the mass of ^{235}U burned per megawatt-day; 1.2 - 1.3 g^{235}U/MWd.

Depending upon the specific values of the constants that are chosen, the calculated heat load can vary by several percent. In any case, the thermal decay heat is expected to be over predicted.

Decay Heat Curves

An analytical expression given by El-Wakil (Ref. 10), which correlate with the decay heat curves of Ref. 11, estimate heat loads about one-half the heat loads calculated above. This heat load expression is

$$H = 4.95 \cdot 10^{-3} \cdot P \cdot t_d^{-0.06} \cdot (t_d^{-0.2} - (t_i + t_d)^{-0.2}) \text{ Watts} \qquad (2)$$

where all symbols, etc. have the same meaning as above and the times are in days.

The ratio of Eq. -2 to Eq. -1 is

$$4.95 \cdot t_d^{-0.06} / 6.85 = 0.72 \cdot t_d^{-0.06}$$

For decay times (t_d) greater than 1 year, the ratio is approximately 0.5.

Experimental Decay Heat Data

Another analytical expression given by Untermyer and Weills (Ref. 12), has been used to fit experimental decay heat data. This heat load expression is

$$\begin{aligned} H = {} & 0.1 \cdot P \cdot [(t_d + 10)^{-0.2} - (t_i + t_d + 10)^{-0.2}] \\ & - 0.087 \cdot P \cdot [(t_d + 2 \cdot 10^7)^{-0.2} - (t_i + t_d + 2 \cdot 10^7)^{-0.2}] \text{ Watts} \qquad (3) \end{aligned}$$

where the irradiation (t_i) and decay (t_d) times are in seconds.

A plot of the ratio (H / P) for Eqs. -2 and -3 are shown in Fig. 2 as a function of decay time and for an irradiation time of 600 days. The ratio calculated with the ORIGEN code is also shown for comparison.

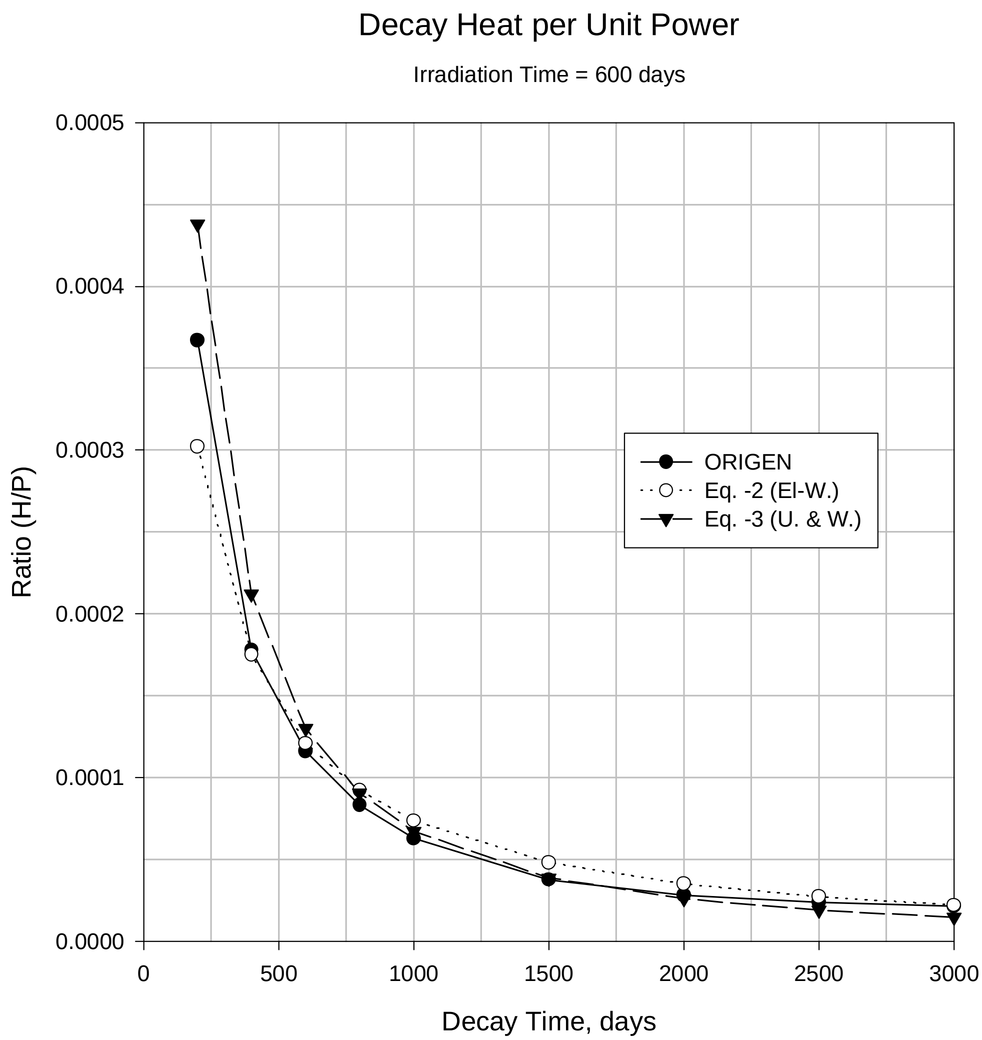

Figure 2. Comparison of Decay Heat Equations-2 and -3 with ORIGEN

These data clearly show the relative decay heat estimated by the decay heat expressions for a typical irradiation time. The ORIGEN ratio is in good agreement with both Eqs. -2 and -3.

CONCLUSIONS

Procedures have been developed to estimate the nuclear mass inventory, the photon dose rate and the thermal decay heat of spent research reactor fuel assemblies. The procedures should provide reasonable estimates based upon known fuel assembly parameters. Estimates for an example spent fuel assembly are given in Appendix D.

Isotopic mass inventories of U, Np, Pu and Am are tabulated in Tables 2–7 for MTR, TRIGA and DIDO fuel assembly types; photon dose rates at 1 m in air are shown in Table 8 for MTR-type fuel assemblies; and analytical expressions are given for the thermal decay heat load of spent uranium fuel. Estimates of TRIGA and DIDO fuel assembly dose rates are respectively, factors of 1.04 and 1.05 times the dose rate for MTR-type fuel assemblies with similar spent fuel material characteristics.

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Table 2. MTR Fuel 93% Enrichment

MTR Fuel	93% Enrichment			100 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	5	10	20	30	40	50	60	70	80
U-234	0	0	0	0	0	0	0	0	0	0
U-235	100	95	90	80	70	60	50	40	30	20
U-236	0	1	2	3	5	6	8	9	11	12
U-238	8	8	8	8	7	7	7	7	7	7
U	108	103	99	91	82	74	65	56	48	39
Np-237	0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.2	0.3
Np	0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.2	0.3
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	0.1	0.1
Pu-240	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	0.2	0.2
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

MTR Fuel	93% Enrichment			200 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	10	20	40	60	80	100	120	140	160
U-234	0	0	0	0	0	0	0	0	0	0
U-235	200	190	180	160	140	120	100	80	60	40
U-236	0	2	3	6	10	13	16	19	21	24
U-238	15	15	15	15	15	15	15	15	14	14
U	215	207	198	181	164	147	130	113	96	78
Np-237	0	0.0	0.0	0.0	0.1	0.2	0.3	0.4	0.6	0.8
Np	0	0.0	0.0	0.0	0.1	0.2	0.3	0.4	0.6	0.8
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1
Pu-239	0	0.0	0.1	0.1	0.2	0.2	0.2	0.3	0.3	0.3
Pu-240	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.0	0.1	0.2	0.2	0.3	0.3	0.4	0.5	0.6
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 2. MTR Fuel 93% Enrichment (conti.)

MTR Fuel	93% Enrichment			300 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	15	30	60	90	120	150	180	210	240
U-234	0	0	0	0	0	0	0	0	0	0
U-235	300	285	270	240	210	180	150	120	90	60
U-236	0	3	5	10	15	19	24	28	33	37
U-238	23	23	22	22	22	22	22	21	21	21
U	323	310	297	272	247	221	196	170	144	118
Np-237	0	0.0	0.0	0.1	0.2	0.4	0.6	0.8	1.1	1.5
Np	0	0.0	0.0	0.1	0.2	0.4	0.6	0.8	1.1	1.5
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-239	0	0.1	0.2	0.3	0.4	0.4	0.5	0.5	0.5	0.5
Pu-240	0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	0.2
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.1	0.2	0.3	0.4	0.5	0.7	0.8	0.9	1.1
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

MTR Fuel	93% Enrichment			400 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	20	40	80	120	160	200	240	280	320
U-234	0	0	0	0	0	0	0	0	0	0
U-235	400	380	360	320	280	240	200	160	120	80
U-236	0	3	7	14	20	26	33	39	44	50
U-238	30	30	30	30	29	29	29	28	28	27
U	430	413	397	363	329	295	261	227	192	157
Np-237	0	0.0	0.0	0.2	0.4	0.6	0.9	1.3	1.7	2.2
Np	0	0.0	0.0	0.2	0.4	0.6	0.9	1.3	1.7	2.2
Pu-238	0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.3	0.5
Pu-239	0	0.1	0.2	0.4	0.6	0.7	0.7	0.7	0.7	0.7
Pu-240	0	0.0	0.0	0.0	0.1	0.1	0.1	0.2	0.2	0.2
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1
Pu	0	0.1	0.3	0.5	0.7	0.9	1.1	1.2	1.4	1.7
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 3. MTR Fuel 45% Enrichment

MTR Fuel	45% Enrichment			200 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	10	20	40	60	80	100	120	140	160
U-234	0	0	0	0	0	0	0	0	0	0
U-235	200	190	180	160	140	120	100	80	60	40
U-236	0	2	3	6	10	13	16	19	21	24
U-238	244	244	244	243	242	241	240	239	237	236
U	444	436	427	409	391	374	356	337	319	300
Np-237	0	0.0	0.0	0.0	0.1	0.2	0.3	0.4	0.6	0.8
Np	0	0.0	0.0	0.0	0.1	0.2	0.3	0.4	0.6	0.8
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2
Pu-239	0	0.4	0.7	1.3	1.7	2.0	2.3	2.4	2.3	2.2
Pu-240	0	0.0	0.0	0.1	0.2	0.3	0.5	0.6	0.8	0.9
Pu-241	0	0.0	0.0	0.0	0.0	0.1	0.2	0.2	0.3	0.4
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.2
Pu	0	0.4	0.7	1.4	2.0	2.5	2.9	3.3	3.6	3.8
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

MTR Fuel	45% Enrichment			300 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	15	30	60	90	120	150	180	210	240
U-234	0	0	0	0	0	0	0	0	0	0
U-235	300	285	270	240	210	180	150	120	90	60
U-236	0	3	5	10	15	19	24	29	33	37
U-238	367	366	365	364	362	361	359	357	355	352
U	667	654	640	614	587	560	533	505	477	449
Np-237	0	0.0	0.0	0.1	0.2	0.4	0.6	0.9	1.2	1.5
Np	0	0.0	0.0	0.1	0.2	0.4	0.6	0.9	1.2	1.5
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-239	0	0.6	1.2	2.2	2.9	3.4	3.8	3.9	3.8	3.6
Pu-240	0	0.0	0.0	0.2	0.3	0.6	0.8	1.0	1.2	1.4
Pu-241	0	0.0	0.0	0.0	0.1	0.2	0.3	0.5	0.6	0.7
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.2	0.3
Pu	0	0.6	1.3	2.4	3.4	4.2	5.0	5.6	6.0	6.3
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 3. MTR Fuel 45% Enrichment (conti.)

MTR Fuel	45% Enrichment			400 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	20	40	80	120	160	200	240	280	320
U-234	0	0	0	0	0	0	0	0	0	0
U-235	400	380	360	320	280	240	200	160	120	80
U-236	0	3	7	14	20	27	33	39	45	50
U-238	489	488	487	485	482	480	477	474	471	467
U	889	871	854	818	782	746	710	673	636	597
Np-237	0	0.0	0.0	0.2	0.4	0.6	1.0	1.4	1.8	2.4
Np	0	0.0	0.0	0.2	0.4	0.6	1.0	1.4	1.8	2.4
Pu-238	0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.3	0.5
Pu-239	0	0.9	1.8	3.2	4.2	4.9	5.4	5.5	5.4	5.0
Pu-240	0	0.0	0.1	0.3	0.5	0.8	1.1	1.4	1.6	1.9
Pu-241	0	0.0	0.0	0.1	0.2	0.3	0.6	0.8	1.0	1.1
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.1	0.2	0.3	0.5
Pu	0	0.9	1.9	3.5	4.9	6.2	7.2	8.1	8.7	9.1
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 4. MTR Fuel 19.75% Enrichment

MTR Fuel	19.75% Enrichment			100 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	5	10	20	30	40	50	60	70	80
U-234	0	0	0	0	0	0	0	0	0	0
U-235	100	95	90	80	70	60	50	40	30	20
U-236	0	1	2	3	5	6	8	9	11	12
U-238	406	406	406	405	404	403	402	401	399	398
U	506	502	497	488	479	469	460	450	440	429
Np-237	0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.2	0.3
Np	0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.2	0.3
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1
Pu-239	0	0.3	0.7	1.2	1.7	2.0	2.3	2.5	2.5	2.5
Pu-240	0	0.0	0.0	0.1	0.2	0.3	0.4	0.6	0.8	1.0
Pu-241	0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.2	0.3
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1
Pu	0	0.3	0.7	1.3	1.9	2.4	2.9	3.3	3.7	4.0
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

MTR Fuel	19.75% Enrichment			200 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	10	20	40	60	80	100	120	140	160
U-234	0	0	0	0	0	0	0	0	0	0
U-235	200	190	180	160	140	120	100	80	60	40
U-236	0	2	3	6	10	13	16	19	22	24
U-238	813	812	811	809	807	805	802	800	796	792
U	1013	1003	994	975	957	937	918	898	878	856
Np-237	0	0.0	0.0	0.0	0.1	0.2	0.3	0.5	0.6	0.9
Np	0	0.0	0.0	0.0	0.1	0.2	0.3	0.5	0.6	0.9
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2
Pu-239	0	0.8	1.5	2.8	3.8	4.6	5.1	5.4	5.5	5.3
Pu-240	0	0.0	0.1	0.2	0.4	0.7	1.0	1.4	1.7	2.1
Pu-241	0	0.0	0.0	0.0	0.1	0.2	0.3	0.5	0.7	0.9
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.2	0.4
Pu	0	0.8	1.6	3.1	4.4	5.5	6.6	7.5	8.2	8.8
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 4. MTR Fuel 19.75% Enrichment (conti.)

MTR Fuel	19.75% Enrichment			300 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	15	30	60	90	120	150	180	210	240
U-234	0	0	0	0	0	0	0	0	0	0
U-235	300	285	270	240	210	180	150	120	90	60
U-236	0	3	5	10	15	20	24	29	33	37
U-238	1219	1218	1216	1213	1209	1205	1201	1197	1191	1184
U	1519	1505	1491	1463	1434	1405	1375	1345	1314	1281
Np-237	0	0.0	0.0	0.1	0.2	0.4	0.6	0.9	1.2	1.6
Np	0	0.0	0.0	0.1	0.2	0.4	0.6	0.9	1.2	1.6
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-239	0	1.3	2.6	4.7	6.3	7.5	8.3	8.7	8.7	8.4
Pu-240	0	0.0	0.1	0.4	0.7	1.2	1.7	2.2	2.7	3.2
Pu-241	0	0.0	0.0	0.1	0.2	0.4	0.7	1.0	1.4	1.6
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.1	0.2	0.4	0.7
Pu	0	1.4	2.7	5.1	7.3	9.2	10.9	12.3	13.4	14.3
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1

MTR Fuel	19.75% Enrichment			400 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	20	40	80	120	160	200	240	280	320
U-234	0	0	0	0	0	0	0	0	0	0
U-235	400	380	360	320	280	240	200	160	120	80
U-236	0	4	7	14	20	27	33	39	45	50
U-238	1625	1623	1621	1616	1611	1605	1599	1592	1584	1574
U	2025	2007	1988	1950	1911	1872	1832	1791	1749	1704
Np-237	0	0.0	0.0	0.2	0.4	0.7	1.0	1.4	1.9	2.5
Np	0	0.0	0.0	0.2	0.4	0.7	1.0	1.4	1.9	2.5
Pu-238	0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.4	0.6
Pu-239	0	2.0	3.8	6.8	9.1	10.8	11.8	12.4	12.3	11.7
Pu-240	0	0.0	0.2	0.6	1.1	1.7	2.4	3.1	3.7	4.3
Pu-241	0	0.0	0.0	0.1	0.4	0.7	1.2	1.7	2.2	2.6
Pu-242	0	0.0	0.0	0.0	0.0	0.1	0.2	0.4	0.7	1.2
Pu	0	2.0	3.9	7.5	10.6	13.4	15.8	17.7	19.3	20.4
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1

Table 4. MTR Fuel 19.75% Enrichment (conti.)

MTR Fuel	19.75% Enrichment			500 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	25	50	100	150	200	250	300	350	400
U-234	0	0	0	0	0	0	0	0	0	0
U-235	500	475	450	400	350	300	250	200	150	100
U-236	0	4	9	18	26	34	42	50	57	64
U-238	2032	2029	2026	2019	2012	2004	1996	1987	1976	1962
U	2532	2508	2484	2437	2388	2338	2288	2236	2183	2126
Np-237	0	0.0	0.1	0.3	0.6	1.0	1.5	2.1	2.8	3.6
Np	0	0.0	0.1	0.3	0.6	1.0	1.5	2.1	2.8	3.6
Pu-238	0	0.0	0.0	0.0	0.0	0.1	0.2	0.4	0.6	0.9
Pu-239	0	2.6	5.0	9.0	12.1	14.3	15.6	16.2	16.1	15.3
Pu-240	0	0.1	0.2	0.8	1.5	2.3	3.2	4.0	4.7	5.4
Pu-241	0	0.0	0.0	0.2	0.6	1.1	1.8	2.5	3.2	3.6
Pu-242	0	0.0	0.0	0.0	0.0	0.1	0.3	0.6	1.0	1.7
Pu	0	2.7	5.3	10.0	14.2	17.9	21.1	23.7	25.7	27.0
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1

Table 5. TRIGA Fuel Single-Rod Model

TRIGA Fuel 8.5wt% U, 70% Enrichment 133 g U-235

U-235 Burnup, %	0	5	10	15	20	25	30	35	40	45	50	55	60
U-235 Burned, g	0	7	13	20	27	33	40	47	53	60	67	73	80
U-234	0	0	0	0	0	0	0	0	0	0	0	0	0
U-235	133	126	120	113	106	100	93	86	80	73	67	60	53
U-236	0	1	3	4	5	6	7	8	10	11	12	13	14
U-238	57	57	56	56	56	56	55	55	55	55	54	54	54
U	190	184	179	173	167	162	156	150	144	138	132	126	120
Np-237	0	0.0	0.0	0.1	0.1	0.1	0.2	0.3	0.3	0.4	0.5	0.6	0.7
Np	0	0.0	0.0	0.1	0.1	0.1	0.2	0.3	0.3	0.4	0.5	0.6	0.7
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1
Pu-239	0	0.3	0.5	0.7	0.8	0.9	1.0	1.1	1.2	1.2	1.2	1.2	1.2
Pu-240	0	0.0	0.0	0.0	0.1	0.1	0.1	0.2	0.2	0.2	0.3	0.3	0.3
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	0.2	0.2	0.2
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1
Pu	0	0.3	0.5	0.7	0.9	1.1	1.2	1.4	1.5	1.6	1.7	1.8	1.9
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

TRIGA Fuel 20wt% U, 20% Enrichment 98 g U-235

U-235 Burnup, %	0	5	10	15	20	25	30	35	40	45	50	55	60
U-235 Burned, g	0	5	10	15	20	25	29	34	39	44	49	54	59
U-234	0	0	0	0	0	0	0	0	0	0	0	0	0
U-235	98	93	88	83	78	74	69	64	59	54	49	44	39
U-236	0	1	2	3	4	4	5	6	7	8	8	9	10
U-238	392	391	391	390	389	388	388	387	386	385	384	383	382
U	490	485	481	476	471	466	461	457	452	447	442	436	431
Np-237	0	0.0	0.0	0.0	0.1	0.1	0.1	0.2	0.2	0.2	0.3	0.4	0.4
Np	0	0.0	0.0	0.0	0.1	0.1	0.1	0.2	0.2	0.2	0.3	0.4	0.4
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1
Pu-239	0	0.6	1.1	1.6	2.0	2.4	2.7	2.9	3.1	3.3	3.4	3.5	3.5
Pu-240	0	0.0	0.1	0.1	0.2	0.3	0.3	0.4	0.5	0.6	0.7	0.8	0.9
Pu-241	0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.2	0.3	0.4	0.5	0.5
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1
Pu	0	0.6	1.2	1.7	2.3	2.7	3.2	3.6	3.9	4.3	4.6	4.9	5.1
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 5. TRIGA Fuel Single-Rod Model (conti.)

TRIGA Fuel 12wt% U, 20% Enrichment 54 g U-235

U-235 Burnup, %	0	5	10	15	20	25	30	35	40	45	50	55	60
U-235 Burned, g	0	3	5	8	11	14	16	19	22	24	27	30	32
U-234	0	0	0	0	0	0	0	0	0	0	0	0	0
U-235	54	51	49	46	43	41	38	35	32	30	27	24	22
U-236	0	0	1	1	2	2	3	3	4	4	4	5	5
U-238	216	216	215	215	215	215	214	214	213	213	213	212	212
U	270	268	265	262	260	257	255	252	249	247	244	241	239
Np-237	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	0.1	0.2
Np	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	0.1	0.2
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.3	0.5	0.7	0.9	1.1	1.2	1.3	1.4	1.5	1.6	1.6	1.7
Pu-240	0	0.0	0.0	0.0	0.1	0.1	0.1	0.2	0.2	0.3	0.3	0.4	0.4
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	0.2	0.2
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.3	0.5	0.8	1.0	1.2	1.4	1.6	1.8	1.9	2.1	2.2	2.4
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

TRIGA Fuel 8.5wt% U, 20% Enrichment 38 g U-235

U-235 Burnup, %	0	5	10	15	20	25	30	35	40	45	50	55	60
U-235 Burned, g	0	2	4	6	8	10	11	13	15	17	19	21	23
U-234	0	0	0	0	0	0	0	0	0	0	0	0	0
U-235	38	36	34	32	30	29	27	25	23	21	19	17	15
U-236	0	0	1	1	1	2	2	2	2	3	3	3	4
U-238	152	152	152	151	151	151	151	151	150	150	150	150	149
U	190	188	186	185	183	181	179	177	176	174	172	170	168
Np-237	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1
Np	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.2	0.3	0.5	0.6	0.7	0.8	0.9	0.9	1.0	1.1	1.1	1.1
Pu-240	0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.2	0.2	0.2	0.3	0.3
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.2	0.3	0.5	0.6	0.8	0.9	1.0	1.1	1.3	1.4	1.5	1.5
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 6. TRIGA Fuel 25-Rod Cluster Model

TRIGA Fuel	10wt% U, 93.1% Enrichment				41.4 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0.0	4.1	8.3	12.4	16.6	20.7	24.8
U-234	0.4	0.4	0.4	0.4	0.4	0.3	0.3
U-235	41.4	37.2	33.1	29.0	24.8	20.7	16.6
U-236	0.2	1.0	1.7	2.4	3.1	3.8	4.4
U-238	2.4	2.4	2.4	2.3	2.3	2.2	2.2
U	44.5	41.0	37.6	34.1	30.6	27.1	23.5
Np-237	0	0.0	0.0	0.1	0.1	0.2	0.2
Np	0	0.0	0.0	0.1	0.1	0.2	0.2
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.0	0.1	0.1	0.1	0.1	0.1
Pu-240	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.0	0.1	0.1	0.1	0.1	0.2
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

TRIGA Fuel	45wt% U, 19.7% Enrichment				53.6 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0.0	5.4	10.7	16.1	21.4	26.8	32.2
U-234	0.4	0.4	0.4	0.3	0.3	0.3	0.3
U-235	53.6	48.3	42.9	37.5	32.2	26.8	21.4
U-236	0.7	1.7	2.7	3.7	4.6	5.5	6.4
U-238	217.4	216.5	215.6	214.6	213.5	212.3	210.9
U	272.1	266.9	261.6	256.1	250.6	244.9	239.0
Np-237	0	0.0	0.1	0.1	0.2	0.3	0.4
Np	0	0.0	0.1	0.1	0.2	0.3	0.4
Pu-238	0	0.0	0.0	0.0	0.0	0.1	0.1
Pu-239	0	0.7	1.3	1.7	1.9	2.1	2.1
Pu-240	0	0.0	0.1	0.2	0.3	0.4	0.5
Pu-241	0	0.0	0.0	0.1	0.2	0.3	0.4
Pu-242	0	0.0	0.0	0.0	0.0	0.1	0.1
Pu	0	0.8	1.4	2.0	2.5	2.9	3.2
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

Table 7. DIDO Fuel

DIDO Fuel	93% Enrichment				150 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0	15	30	45	60	75	90
U-234	0	0	0	0	0	0	0
U-235	150	135	120	105	90	75	60
U-236	0	2	5	7	9	12	14
U-238	11	11	11	11	11	11	11
U	161	149	136	123	110	98	85
Np-237	0	0.0	0.0	0.1	0.1	0.2	0.3
Np	0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.0	0.1	0.1	0.1	0.1	0.2
Pu-240	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.0	0.1	0.1	0.2	0.2	0.2
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

DIDO Fuel	80% Enrichment				150 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0	15	30	45	60	75	90
U-234	0	0	0	0	0	0	0
U-235	150	135	120	105	90	75	60
U-236	0	2	5	7	9	12	14
U-238	38	37	37	37	37	37	37
U	188	175	162	149	136	123	110
Np-237	0	0.0	0.0	0.1	0.1	0.2	0.3
Np	0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.1	0.2	0.3	0.3	0.4	0.4
Pu-240	0	0.0	0.0	0.0	0.1	0.1	0.1
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.1	0.2	0.3	0.4	0.5	0.6
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

DIDO Fuel	60% Enrichment				150 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0	15	30	45	60	75	90
U-234	0	0	0	0	0	0	0
U-235	150	135	120	105	90	75	60
U-236	0	2	5	7	9	12	14
U-238	100	100	100	99	99	99	98
U	250	237	224	211	198	185	172
Np-237	0	0.0	0.0	0.1	0.1	0.2	0.3
Np	0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.2	0.4	0.6	0.7	0.7	0.8
Pu-240	0	0.0	0.0	0.1	0.1	0.2	0.2
Pu-241	0	0.0	0.0	0.0	0.0	0.1	0.1
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.2	0.5	0.7	0.8	1.0	1.1
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

DIDO Fuel	20% Enrichment				200 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0	20	40	60	80	100	120
U-234	0	0	0	0	0	0	0
U-235	200	180	160	140	120	100	80
U-236	0	3	7	10	13	16	19
U-238	800	799	797	796	794	793	791
U	1000	982	964	946	927	908	890
Np-237	0	0.0	0.1	0.1	0.2	0.3	0.4
Np	0	0.0	0.1	0.1	0.2	0.3	0.4
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1
Pu-239	0	1.1	2.0	2.7	3.2	3.5	3.7
Pu-240	0	0.0	0.2	0.3	0.6	0.8	1.0
Pu-241	0	0.0	0.0	0.1	0.2	0.3	0.4
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.1
Pu	0	1.2	2.2	3.1	4.0	4.7	5.3
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

Table 7. DIDO Fuel

DIDO Fuel	93% Enrichment				150 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0	15	30	45	60	75	90
U-234	0	0	0	0	0	0	0
U-235	150	135	120	105	90	75	60
U-236	0	2	5	7	9	12	14
U-238	11	11	11	11	11	11	11
U	161	149	136	123	110	98	85
Np-237	0	0.0	0.0	0.1	0.1	0.2	0.3
Np	0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.0	0.1	0.1	0.1	0.1	0.2
Pu-240	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.0	0.1	0.1	0.2	0.2	0.2
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

DIDO Fuel	80% Enrichment				150 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0	15	30	45	60	75	90
U-234	0	0	0	0	0	0	0
U-235	150	135	120	105	90	75	60
U-236	0	2	5	7	9	12	14
U-238	38	37	37	37	37	37	37
U	188	175	162	149	136	123	110
Np-237	0	0.0	0.0	0.1	0.1	0.2	0.3
Np	0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.1	0.2	0.3	0.3	0.4	0.4
Pu-240	0	0.0	0.0	0.0	0.1	0.1	0.1
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.1	0.2	0.3	0.4	0.5	0.6
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

DIDO Fuel	60% Enrichment				150 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0	15	30	45	60	75	90
U-234	0	0	0	0	0	0	0
U-235	150	135	120	105	90	75	60
U-236	0	2	5	7	9	12	14
U-238	100	100	100	99	99	99	98
U	250	237	224	211	198	185	172
Np-237	0	0.0	0.0	0.1	0.1	0.2	0.3
Np	0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.2	0.4	0.6	0.7	0.7	0.8
Pu-240	0	0.0	0.0	0.1	0.1	0.2	0.2
Pu-241	0	0.0	0.0	0.0	0.0	0.1	0.1
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.2	0.5	0.7	0.8	1.0	1.1
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

DIDO Fuel	20% Enrichment				200 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0	20	40	60	80	100	120
U-234	0	0	0	0	0	0	0
U-235	200	180	160	140	120	100	80
U-236	0	3	7	10	13	16	19
U-238	800	799	797	796	794	793	791
U	1000	982	964	946	927	908	890
Np-237	0	0.0	0.1	0.1	0.2	0.3	0.4
Np	0	0.0	0.1	0.1	0.2	0.3	0.4
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1
Pu-239	0	1.1	2.0	2.7	3.2	3.5	3.7
Pu-240	0	0.0	0.2	0.3	0.6	0.8	1.0
Pu-241	0	0.0	0.0	0.1	0.2	0.3	0.4
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.1
Pu	0	1.2	2.2	3.1	4.0	4.7	5.3
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

APPENDIX A

MTR MODEL MASS INVENTORY SENSITIVITY

This appendix examines the sensitivity of MTR-type fuel assemblies to the number of fuel plates in the assembly as well as the fuel element specifications for the fuel, clad and coolant. An examination of many MTR-type fuel assemblies shows that the ratio of the coolant channel thickness to the fuel meat thickness, times the number of fuel plates, is nearly a constant. This constant is also proportional to the H/U-235 atom ratio which can be used to characterize the neutron spectrum in MTR-type fuel assemblies.

Figure A1 shows the H/U-235 atom ratio as a function of the U-235 mass. The upper curve are for 19-plate (0.51mm fuel, 0.38mm clad, 2.95mm coolant) elements and the lower curve are for 23-plate (0.51mm fuel, 0.38mm clad, 2.19mm coolant) elements. Most all MTR-type fuel assemblies as a function of the fuel element specifications are within the range (±6%) of the average H/U-235 ratio.

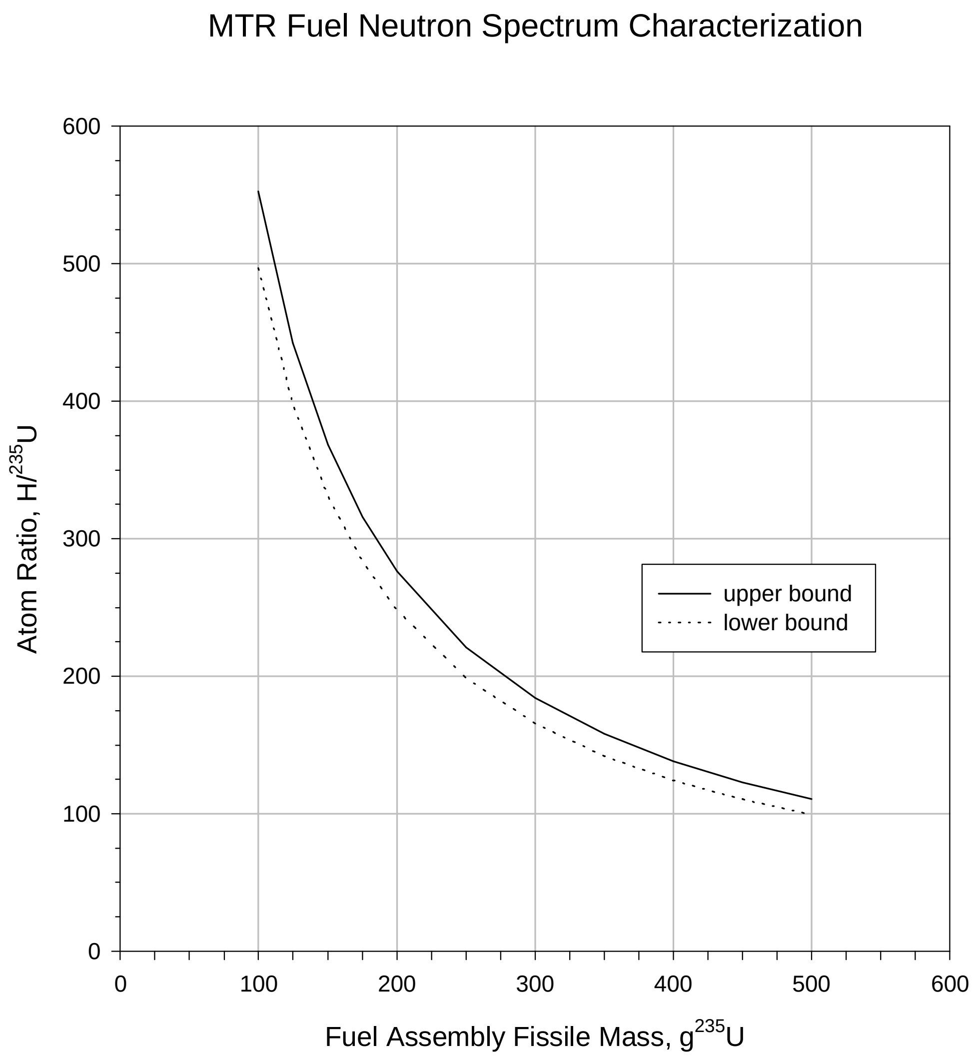

Figure A1. MTR Fuel Assembly Model Sensitivity

Tables A1–A3 show the mass inventory results for MTR fuel assembly types with 300g U-235 and 93, 45 and 19.75% U-235 enrichment. The difference between the upper and lower bound results indicate only small differences in the isotopic masses as a function of fuel element specification.

Table A1. MTR Fuel 93% Enrichment

MTR Upper Bound 93% Enrichment 300 g U-235

U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	15	30	60	90	120	150	180	210	240
U-234	0	0	0	0	0	0	0	0	0	0
U-235	300	285	270	240	210	180	150	120	90	60
U-236	0	3	5	10	15	19	24	28	33	37
U-238	23	23	22	22	22	22	22	21	21	21
U	323	310	297	272	247	221	196	170	144	118
Np-237	0	0.0	0.0	0.1	0.2	0.4	0.6	0.8	1.1	1.5
Np	0	0.0	0.0	0.1	0.2	0.4	0.6	0.8	1.1	1.5
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-239	0	0.1	0.2	0.3	0.4	0.4	0.5	0.5	0.5	0.5
Pu-240	0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	0.2
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.1	0.2	0.3	0.4	0.5	0.7	0.8	0.9	1.1
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

MTR Lower Bound 93% Enrichment 300 g U-235

U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	15	30	60	90	120	150	180	210	240
U-234	0	0	0	0	0	0	0	0	0	0
U-235	300	285	270	240	210	180	150	120	90	60
U-236	0	3	5	10	15	20	24	29	33	37
U-238	23	23	22	22	22	22	22	21	21	21
U	323	310	297	272	247	221	196	170	144	118
Np-237	0	0.0	0.0	0.1	0.2	0.4	0.6	0.9	1.2	1.6
Np	0	0.0	0.0	0.1	0.2	0.4	0.6	0.9	1.2	1.6
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-239	0	0.1	0.2	0.3	0.4	0.4	0.5	0.5	0.5	0.5
Pu-240	0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.2	0.2
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.1	0.2	0.3	0.5	0.6	0.7	0.8	1.0	1.1
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table A2. MTR Fuel 45% Enrichment

MTR Upper Bound	45% Enrichment			300 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	15	30	60	90	120	150	180	210	240
U-234	0	0	0	0	0	0	0	0	0	0
U-235	300	285	270	240	210	180	150	120	90	60
U-236	0	3	5	10	15	19	24	29	33	37
U-238	367	366	365	364	362	361	359	357	355	352
U	667	654	640	614	587	560	533	505	477	449
Np-237	0	0.0	0.0	0.1	0.2	0.4	0.6	0.9	1.2	1.5
Np	0	0.0	0.0	0.1	0.2	0.4	0.6	0.9	1.2	1.5
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-239	0	0.6	1.2	2.2	2.9	3.4	3.8	3.9	3.8	3.6
Pu-240	0	0.0	0.0	0.2	0.3	0.6	0.8	1.0	1.2	1.4
Pu-241	0	0.0	0.0	0.0	0.1	0.2	0.3	0.5	0.6	0.7
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.2	0.3
Pu	0	0.6	1.3	2.4	3.4	4.2	5.0	5.6	6.0	6.3
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

MTR Lower Bound	45% Enrichment			300 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	15	30	60	90	120	150	180	210	240
U-234	0	0	0	0	0	0	0	0	0	0
U-235	300	285	270	240	210	180	150	120	90	60
U-236	0	3	5	10	15	20	24	29	33	37
U-238	367	366	365	364	362	360	358	356	354	351
U	667	654	640	614	587	560	532	505	477	448
Np-237	0	0.0	0.0	0.1	0.2	0.4	0.7	0.9	1.3	1.6
Np	0	0.0	0.0	0.1	0.2	0.4	0.7	0.9	1.3	1.6
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.4
Pu-239	0	0.7	1.3	2.3	3.1	3.7	4.0	4.1	4.0	3.8
Pu-240	0	0.0	0.1	0.2	0.4	0.6	0.8	1.0	1.3	1.4
Pu-241	0	0.0	0.0	0.0	0.1	0.2	0.4	0.5	0.7	0.8
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.4
Pu	0	0.7	1.4	2.6	3.6	4.5	5.3	5.9	6.4	6.7
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table A3. MTR Fuel 19.75% Enrichment

MTR Upper Bound	19.75% Enrichment			300 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	15	30	60	90	120	150	180	210	240
U-234	0	0	0	0	0	0	0	0	0	0
U-235	300	285	270	240	210	180	150	120	90	60
U-236	0	3	5	10	15	20	24	29	33	37
U-238	1219	1218	1216	1213	1209	1205	1201	1197	1191	1184
U	1519	1505	1491	1463	1434	1405	1375	1345	1314	1281
Np-237	0	0.0	0.0	0.1	0.2	0.4	0.6	0.9	1.2	1.6
Np	0	0.0	0.0	0.1	0.2	0.4	0.6	0.9	1.2	1.6
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.3
Pu-239	0	1.3	2.6	4.7	6.3	7.5	8.3	8.7	8.7	8.4
Pu-240	0	0.0	0.1	0.4	0.7	1.2	1.7	2.2	2.7	3.2
Pu-241	0	0.0	0.0	0.1	0.2	0.4	0.7	1.0	1.4	1.6
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.1	0.2	0.4	0.7
Pu	0	1.4	2.7	5.1	7.3	9.2	10.9	12.3	13.4	14.3
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1

MTR Lower Bound	19.75% Enrichment			300 g U-235						
U-235 Burnup, %	0	5	10	20	30	40	50	60	70	80
U-235 Burned, g	0	15	30	60	90	120	150	180	210	240
U-234	0	0	0	0	0	0	0	0	0	0
U-235	300	285	270	240	210	180	150	120	90	60
U-236	0	3	5	10	15	20	24	29	33	37
U-238	1219	1217	1216	1212	1209	1205	1200	1195	1189	1182
U	1519	1505	1491	1462	1433	1404	1374	1344	1312	1279
Np-237	0	0.0	0.0	0.1	0.2	0.4	0.7	1.0	1.3	1.8
Np	0	0.0	0.0	0.1	0.2	0.4	0.7	1.0	1.3	1.8
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.4
Pu-239	0	1.4	2.8	5.0	6.7	8.0	8.8	9.2	9.2	8.8
Pu-240	0	0.0	0.1	0.4	0.8	1.3	1.8	2.3	2.8	3.3
Pu-241	0	0.0	0.0	0.1	0.2	0.5	0.8	1.2	1.5	1.8
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.1	0.2	0.5	0.8
Pu	0	1.5	2.9	5.5	7.8	9.8	11.6	13.1	14.3	15.1
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1
Am	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1

APPENDIX B

U-234 AND U-236 MASS INVENTORY SENSITIVITY

The initial fuel composition of some reactor fuels may contain specifications for U-234 and/or U-236 in addition to the usual specifications for U-235 and U-238. It is the purpose of this appendix to evaluate the effect that U-234 and U-236 have on the overall fuel assembly mass inventory when these isotopes are or are not included in the initial fuel assembly composition.

A comparison of the fuel mass inventory for a HEU and a LEU fuel composition, with and without initial enrichments of U-234 and U-236, are shown in Table B1. Typical enrichments of U-234 and U-236 in research reactor fuels are less than 1%; these specific data are for typical TRIGA fuel compositions.

The upper section of Table B1 shows the mass inventory for HEU and LEU fuels with initial enrichments of U-234 and U-236, and the lower section shows similar data for the same fuels but without initial U-234 and U-236 enrichment. The result of this comparison shows that to first-order, any initial mass of U-234 or U-236 can be simply added to the mass inventory for U-234, U-236 and total U at any burnup level. The mass inventory for Np-237 and Pu-238 which are also functions of the U-236 mass, are not substantially affected by an initial enrichment of U-236.

Table B1. TRIGA Fuel 25-Rod Cluster Model

TRIGA Fuel	10wt% U, 93.1% Enrichment				41.4 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0.0	4.1	8.3	12.4	16.6	20.7	24.8
U-234	0.4	0.4	0.4	0.4	0.4	0.3	0.3
U-235	41.4	37.2	33.1	29.0	24.8	20.7	16.6
U-236	0.2	1.0	1.7	2.4	3.1	3.8	4.4
U-238	2.4	2.4	2.4	2.3	2.3	2.2	2.2
U	44.5	41.0	37.6	34.1	30.6	27.1	23.5
Np-237	0	0.0	0.0	0.1	0.1	0.2	0.2
Np	0	0.0	0.0	0.1	0.1	0.2	0.2
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.0	0.1	0.1	0.1	0.1	0.1
Pu-240	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.0	0.1	0.1	0.1	0.1	0.2
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

TRIGA Fuel	45wt% U, 19.7% Enrichment				53.6 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0.0	5.4	10.7	16.1	21.4	26.8	32.2
U-234	0.4	0.4	0.4	0.3	0.3	0.3	0.3
U-235	53.6	48.3	42.9	37.5	32.2	26.8	21.4
U-236	0.7	1.7	2.7	3.7	4.6	5.5	6.4
U-238	217.4	216.5	215.6	214.6	213.5	212.3	210.9
U	272.1	266.9	261.6	256.1	250.6	244.9	239.0
Np-237	0	0.0	0.1	0.1	0.2	0.3	0.4
Np	0	0.0	0.1	0.1	0.2	0.3	0.4
Pu-238	0	0.0	0.0	0.0	0.0	0.1	0.1
Pu-239	0	0.7	1.3	1.7	1.9	2.1	2.1
Pu-240	0	0.0	0.1	0.2	0.3	0.4	0.5
Pu-241	0	0.0	0.0	0.1	0.2	0.3	0.4
Pu-242	0	0.0	0.0	0.0	0.0	0.1	0.1
Pu	0	0.8	1.4	2.0	2.5	2.9	3.2
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

TRIGA Fuel	10wt% U, 93.1% Enrichment				41.4 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0.0	4.1	8.3	12.4	16.6	20.7	24.8
U-234	0.0	0.0	0.0	0.0	0.0	0.0	0.0
U-235	41.4	37.2	33.1	29.0	24.8	20.7	16.6
U-236	0.0	0.8	1.5	2.3	2.9	3.6	4.2
U-238	2.4	2.4	2.4	2.3	2.3	2.2	2.2
U	43.8	40.4	37.0	33.5	30.1	26.5	23.0
Np-237	0	0.0	0.0	0.1	0.1	0.1	0.2
Np	0	0.0	0.0	0.1	0.1	0.1	0.2
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-239	0	0.0	0.1	0.1	0.1	0.1	0.1
Pu-240	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-242	0	0.0	0.0	0.0	0.0	0.0	0.0
Pu	0	0.0	0.1	0.1	0.1	0.1	0.2
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

TRIGA Fuel	45wt% U, 19.7% Enrichment				53.6 g U-235		
U-235 Burnup, %	0	10	20	30	40	50	60
U-235 Burned, g	0.0	5.4	10.7	16.1	21.4	26.8	32.2
U-234	0.0	0.0	0.0	0.0	0.0	0.0	0.0
U-235	53.6	48.3	42.9	37.5	32.2	26.8	21.4
U-236	0.0	1.1	2.1	3.1	4.0	4.9	5.7
U-238	217.4	216.5	215.6	214.6	213.5	212.3	211.0
U	271.1	265.9	260.6	255.2	249.7	244.0	238.1
Np-237	0	0.0	0.0	0.1	0.2	0.2	0.3
Np	0	0.0	0.0	0.1	0.2	0.2	0.3
Pu-238	0	0.0	0.0	0.0	0.0	0.0	0.1
Pu-239	0	0.7	1.3	1.7	1.9	2.1	2.1
Pu-240	0	0.0	0.1	0.2	0.3	0.4	0.5
Pu-241	0	0.0	0.0	0.1	0.2	0.3	0.4
Pu-242	0	0.0	0.0	0.0	0.0	0.1	0.1
Pu	0	0.8	1.4	2.0	2.5	2.9	3.2
Am-241	0	0.0	0.0	0.0	0.0	0.0	0.0
Am	0	0.0	0.0	0.0	0.0	0.0	0.0

APPENDIX C

MASS INVENTORY ESTIMATE: ORIGEN VS. WIMS

In this paper, the spent fuel nuclear mass inventories are based upon material number densities calculated within the WIMS code using burnup dependent cross sections and fluxes to solve the material transmutation equations. Unit-cell models of MTR, TRIGA and DIDO fuel assemblies with typical fuel compositions used WIMS to generate actinide cross sections and number densities as a function of U-235 burnup.

Spot checks of the mass inventories for two TRIGA fuel compositions were also calculated using the isotope generation and depletion code, ORIGEN. The principal actinide cross sections input to ORIGEN were collapsed one-group, zero-burnup material cross sections calculated by WIMS.

The fuel material mass inventories predicted by ORIGEN and by WIMS for TRIGA fuel materials (133 g HEU and 38 g LEU) with 35% U-235 burnup are shown in Table C1. The uranium isotopes at the 1-gram level and the Np, Pu and Am isotopes at the 0.1-gram level are in reasonably good agreement. The slightly larger ^{237}Np and ^{239}Pu and the slightly smaller ^{238}U inventories are due to the use of zero-burnup cross sections in estimating the 35% U-235 burnup inventories.

Table C1. Gram-Mass Inventory Estimates

	8.5wt% U, 70% Enrichment 133 g U-235		8.5wt% U, 20% Enrichment 38 g U-235	
	ORIGEN	WIMS	ORIGEN	WIMS
U-235 Burnup, %	35	35	35	35
U-235 Burned, g	47	47	13	13
U-234	0	0	0	0
U-235	86	86	25	25
U-236	8	8	2	2
U-238	55	55	150	151
U	150	150	177	177
Np-237	0.4	0.3	0.0	0.0
Np	0.4	0.3	0.0	0.0
Pu-238	0.0	0.0	0.0	0.0
Pu-239	1.3	1.1	0.9	0.9
Pu-240	0.2	0.2	0.1	0.1
Pu-241	0.1	0.1	0.0	0.0
Pu-242	0.0	0.0	0.0	0.0
Pu	1.6	1.4	1.1	1.0
Am-241	0.0	0.0	0.0	0.0
Am	0.0	0.0	0.0	0.0

The mass inventories using the WIMS 35%-burnup material cross sections as input to ORIGEN shows a change in the inventories in the direction of the WIMS results. In particular, the ^{237}Np and ^{239}Pu inventories decrease and the ^{238}U inventory increases. With these cross sections, the ^{237}Np and ^{239}Pu inventories are slightly underestimated compared to WIMS. The cross sections used in ORIGEN are not extremely sensitive to burnup, but they should be for a specific fuel material composition and not simply default library cross sections. The difference between ORIGEN and WIMS inventories would be expected to increase as U-235 burnup increases.

Since the Np and Pu mass inventories calculated above are slightly overestimated using zero-burnup cross sections and slightly underestimated using 35%-burnup cross sections, it is recommended that mid-cycle burnup cross sections be used in any ORIGEN mass inventory calculation. The mid-cycle cross sections would be expected to approximately cancel any over- or under-estimate and give inventory masses of U, Np, Pu and Am closer to the masses calculated with WIMS.

APPENDIX D

EXAMPLE CALCULATION: NUCLEAR MASS INVENTORY, PHOTON DOSE RATE AND THERMAL DECAY HEAT

In this example, a 280 g^{235}U MTR-type fuel assembly has been irradiated at an average fuel assembly power ($\overline{P}$) of 25 kW over an elapsed time (t_e) of 3584 days. The irradiation history of this fuel assembly is such that it can not be described simply, using a constant power (P) and a continuous irradiation time (t_i). It is assumed, however, that

$$\overline{P} \cdot t_e = 89.6\,MWd = \sum (P \cdot t_i)$$

where the sum of ($P \cdot t_i$) traces the fuel assembly irradiation history over all irradiation segments when the fuel assembly power was constant and the irradiation time was continuous. The elapsed time is the calendar time from the first through the last irradiation segment. Assuming 1.25 g^{235}U burned per MWd, this fuel assembly has 112 g^{235}U burned and 40% ^{235}U burnup. The fission product decay time (t_d) or cooling time for this fuel assembly is assumed to be 3 years.

Nuclear Mass Inventory

If the fuel assembly enrichment is 93%, then 300 g^{235}U, 40% ^{235}U burnup data of Table 2 can be prorated to 280 g^{235}U. For enrichments of 45 or 19.75%, similar prorated data from Table 3 or 4, respectively, should be used. Table D1 summarize the spent fuel mass inventory of 280 g^{235}U fuel assemblies which have 40% ^{235}U burnup.

Table D1. Mass Inventory of Spent HEU, MEU and LEU Fuel Assemblies

Isotope	HEU-93%	MEU-45%	LEU-19.75%
U-234	0	0	0
U-235	168	168	168
U-236	18	18	19
U-238	21	337	1125
U	206	523	1311
Np-237	0.4	0.4	0.4
Np	0.4	0.4	0.4
Pu-238	0.0	0.0	0.0
Pu-239	0.4	3.2	7.0
Pu-240	0.1	0.6	1.1
Pu-241	0.0	0.2	0.4
Pu-242	0.0	0.0	0.0
Pu	0.5	3.9	8.6
Am-241	0.0	0.0	0.0
Am	0.0	0.0	0.0

These 280 g^{235}U spent fuel inventory masses could also have been estimated using linear interpolation of the 200 and 300 g^{235}U, 40% ^{235}U burnup data tabulated in Tables 2, 3 and 4. Note, inventory masses for non-tabulated fuel assembly burnup should also use linear interpolation of tabulated data (e.g. 45% ^{235}U burnup, interpolate between 40 and 50% tabulated data).

Photon Dose Rate

The photon dose rate of this fuel assembly is calculated from data presented in Table 8. The assembly power density is 0.089 MW/kg^{235}U (25 kW / 280 g^{235}U), the ^{235}U burnup is 40%, and the decay time is 3 years. With these data, Table 8 estimates that the photon dose rate is 1.02 rem/h per g^{235}U burned. With 112 g^{235}U burned, the dose rate is 114 rem/h at 1 meter from the fuel assembly.

For fuel with 40% burnup and with 112 g^{235}U burned, Fig. 1 estimates that this fuel assembly will be self-protecting (dose rate greater than 100 rem/h) for about 4 years.

The photon dose rate for non-tabulated assembly power densities, ^{235}U burnup and/or decay times can be estimated using linear interpolation of the data in Table 8. Linear interpolation to determine the photon dose rate would be necessary, for example, for a fuel assembly with the following parameters: 3.5 year decay time, 50% ^{235}U burnup and 0.134 MW/kg^{235}U assembly power density. A simple table which interpolates each parameter separately is a useful aid. Table D2 is constructed to determine the photon dose rate for these non-tabulated fuel assembly parameters.

Table D2. Fuel Assembly Parameter Linear Interpolation

Decay Time, y	Burnup, % ^{235}U	Assembly Power Density, MW/kg^{235}U	Photon Dose Rate, rem/h per g^{235}U burned
3	50	0.179	1.31
3	50	0.089	1.07
3	50	0.134	1.19
4	50	0.179	1.10
4	50	0.089	0.931
4	50	0.134	1.0155
3.5	50	0.134	1.10

The bottom line, estimated photon dose rate is 1.10 rem/h per g^{235}U burned.

Thermal Decay Heat

ORIGEN Calculation

The thermal decay heat calculated with the ORIGEN code for this example is about 4.2 Watts.

Integrated Emission Rate Equation

The thermal decay heat of this fuel assembly using the conservative heat load equation based upon Eq. -1

$$H \cong 6.85 \cdot 10^{-3} \cdot \overline{P} \cdot (t_d^{-0.2} - (t_e + t_d)^{-0.2}) \text{ Watts}$$

is about 10.6 W. This result is based upon an average fuel assembly power ($\overline{P}$) of 25,000 Watts, a cooling or decay time (t_d) of 1095 days (3 y) and an elapsed time (t_e) of 3584 days.

El-Wakil Equation

The thermal decay heat with these same data and the heat load equation based upon Eq. -2

$$H \cong 4.95 \cdot 10^{-3} \cdot \overline{P} \cdot t_d^{-0.06} \cdot (t_d^{-0.2} - (t_e + t_d)^{-0.2}) \text{ Watts}$$

is about 5.1 W.

Untermyer and Weills Equation

Similarly, using the heat load equation based upon Eq. -3 with a decay time of $9.46 \cdot 10^7$ seconds (1095 d) and an elapsed time of $3.10 \cdot 10^8$ seconds (3584 d)

$$\begin{aligned} H \cong {} & 0.1 \cdot \overline{P} \cdot [(t_d + 10)^{-0.2} - (t_e + t_d + 10)^{-0.2}] \\ & - 0.087 \cdot \overline{P} \cdot [(t_d + 2 \cdot 10^7)^{-0.2} - (t_e + t_d + 2 \cdot 10^7)^{-0.2}] \text{ Watts} \end{aligned}$$

is about 3.8 W.