

**PHOTON DOSE RATES FROM SPENT FUEL ASSEMBLIES
WITH RELATION TO SELF-PROTECTION (Rev. 1)**

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ABSTRACT

Photon dose rates as a function of fission product decay times have been calculated for spent fuel assemblies typical of MTR-type research and test reactors. Based upon these dose rates, the length of time that a spent fuel assembly will be self-protecting (dose rate greater than 100 rem/h at 1 m in air) can be estimated knowing the mass of fuel burned, the fraction of fuel burned, and the fuel assembly specific power density.

The calculated dose rates cover 20 years of fission product decay, spent fuel with up to 80% ^{235}U burnup and assembly power densities ranging from 0.089 to 2.857 MW/kg ^{235}U . Most of the results are unshielded dose rates at 1 m in air with some shielded dose rates at 40 cm in water. Dose rate sensitivity estimates have been evaluated for a variety of MTR fuel assembly designs and for uncertainties in both the physical and analytical models of the fuel assemblies.

The Monte Carlo dose rates reported in this paper are a revision of the dose rates reported in Ref. 1. These dose rates are the result of a revised energy group structure used for the MCNP calculations, and the fluence-to-dose conversion factors for the new group structure. The photon dose rates in air (water) are a factor 0.85 (0.98) of the old photon dose rates. In addition, this paper presents an analytical method for estimating the photon dose rate of spent fuel, assuming a line-source model of the fission products in a fuel assembly.

INTRODUCTION

The photon dose rate from spent nuclear fuel is a factor, which in combination with the material form and quantity, determines the physical protection requirements^[2] for the spent fuel material. In this paper, the photon dose rate from spent fuel assemblies is calculated for the purpose of estimating the radiation level. The dose rate data are evaluated as functions of specific power density and burnup in the fuel assembly, and as a function of fission product decay time in the spent fuel.

A fuel assembly is considered to be self-protecting when the dose rate is greater than 100 rem/h at a distance of 1 m in air. It is important to know if spent nuclear fuel is self-protecting or when self-protection is lost since significant and costly additional physical protection requirements could be necessary.

Because of the high-radiation fields, it is not always convenient to measure fuel assembly dose rates in an unshielded configuration. As an aid to assess spent fuel radiation levels, dose rates in air have been calculated to cover a broad range of MTR-type fuel assembly designs and burnup histories. Some dose rate calculations also have been made in a shielded configuration. These data could be useful in correlating dose rates in water to corresponding dose rates in air.

FUEL ASSEMBLY MODEL

A functional model of an MTR-type fuel assembly and the locations where dose rates were calculated is shown in Fig. 1. The fuel assembly has aluminum-clad fuel plates and aluminum side plates. The locations are typical of where dose rate measurements could be made relative to the fuel assembly. In the criterion for radiation protection, the location is unspecified other than at 1 m. The dose rate in general will be a function of the fuel assembly orientation.

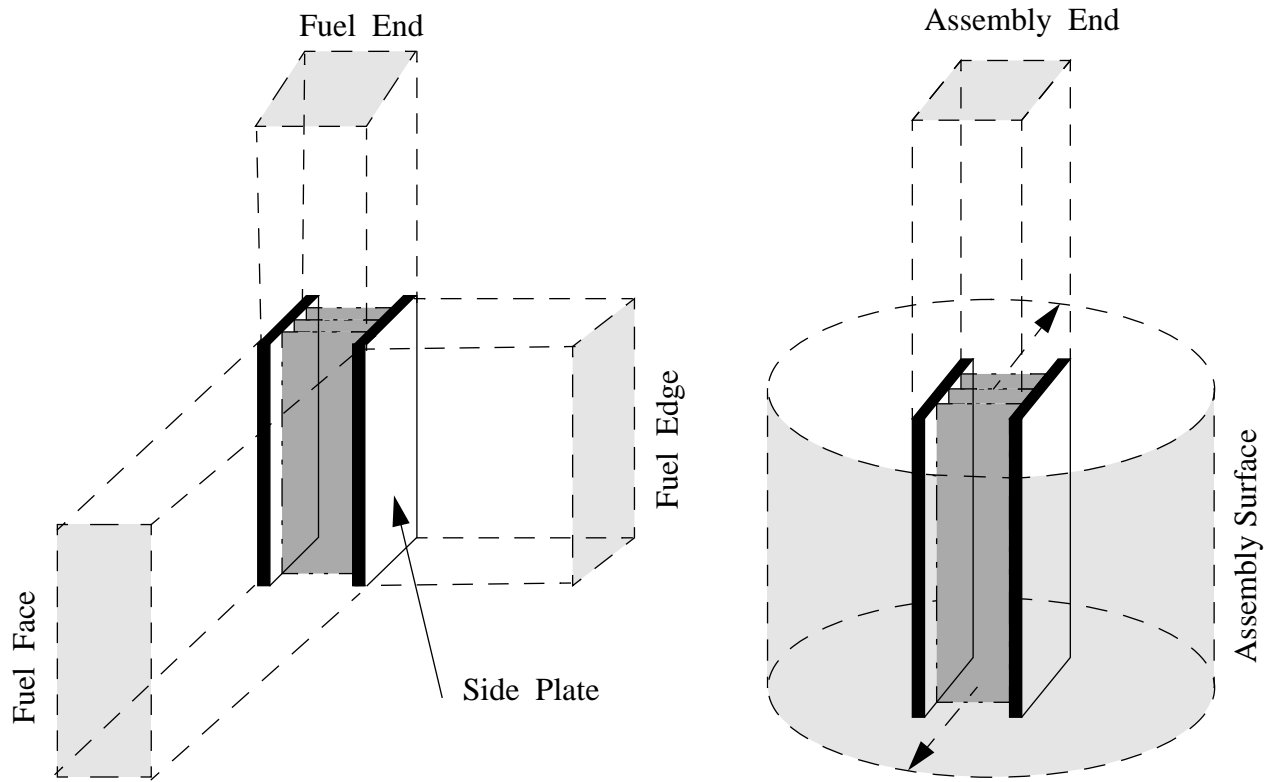


Figure 1. Model of Dose Rate Locations Relative to a Fuel Assembly

The orientation of the fuel assembly in Fig. 1 represent locations towards the fuel-plate face, the fuel-plate edge, and the assembly end - each 1 m from the side plates of the fuel assembly. The cylindrical surface is located 1 m from the axial center line and represents an average dose rate around the fuel assembly. In all cases the dose rates were averaged over the indicated surface areas as defined by the fuel assembly dimensions.

Three MTR fuel assemblies with 16-, 19- and 23-fuel plates were modeled (Fig. 2). The assembly with 16 fuel plates has two additional outside aluminum plates. These three models represent most standard and control, plate-type fuel assemblies. In all cases the fuel assembly was modeled as 60-cm long without end pieces.

CALCULATIONAL MODEL

The fission product photon source was calculated using the isotope generation and depletion code, ORIGEN^[3]. The source was calculated for a ^{235}U mass with up to 80% burnup and for six power densities from 0.089 to 2.857 MW/kg ^{235}U . Burnup in these calculations is equal to the product of the fuel assembly specific power density and the exposure time in days, times the constant $1.25\text{e-}3$ kg ^{235}U burned per MWd. At each burnup level, the fission product photon source was calculated in yearly increments through 20 years of fission product decay.

These photon source data as functions of ^{235}U burnup, assembly specific power density, and fission product decay time were then introduced into an MCNP^[4] Monte Carlo model to calculate the photon flux in select regions around the fuel assembly. The photon source in all cases was uniformly distributed in the fuel meat of all fuel plates in a fuel assembly. The fuel, clad, side plates, gaps, etc. were all modeled so as to fully account for photon transport in the fuel assembly.

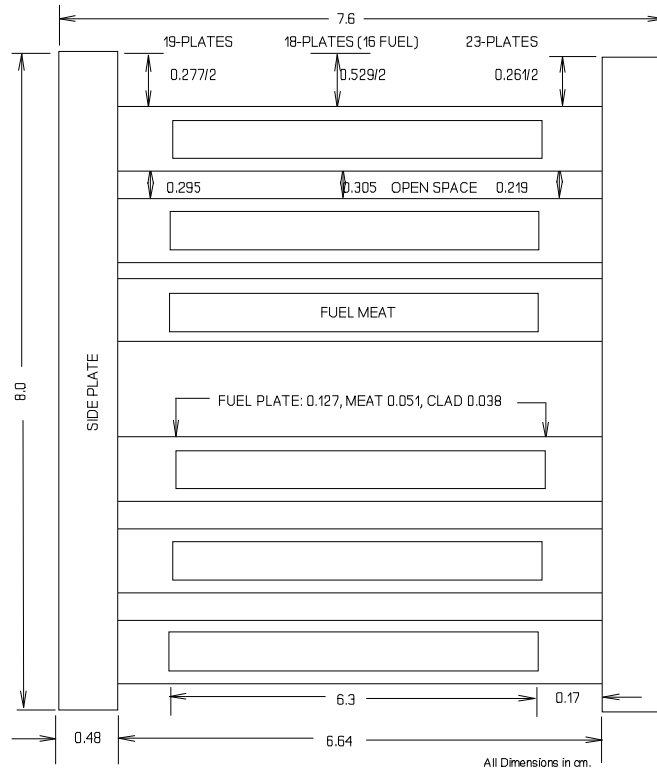


Figure 2. Cross Section of 18-, 19- and 23-Plate Fuel Assemblies

Based upon the calculated photon flux per unit photon source, the dose rates in rem/h were then calculated using the American National Standard fluence-to-dose factors given in Ref. 5.

DOSE RATE SENSITIVITIES

MTR Fuel Assembly Design Variation

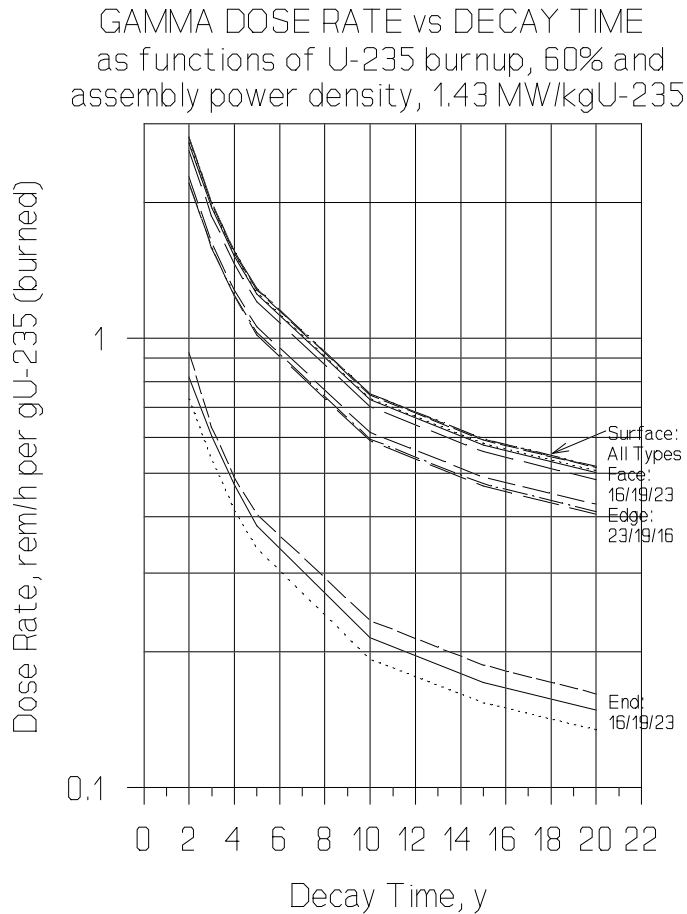


Figure 3. Surface-, Face-, Edge- and End-Location Dose Rates for Three Fuel Assembly Types

Shown in Fig. 3 are dose rates per $g^{235}\text{U}$ burned for the three MTR fuel assembly designs shown in Fig. 2 and for the four "1-meter" regions represented in Fig. 1. Figure 3 shows the variation between the dose rates at 1 m from the face, edge and end of an assembly. It also shows the dose rate variation as a function of the number of fuel plates in an assembly.

Figure 3 also shows that as a function of fission product decay time, the radial surface area dose rate is, within statistics, the same for all fuel assembly designs. Since the same total photon source is assumed in each fuel assembly design, the average dose rate around an assembly is approximately the

same. The insensitivity of the dose rate to the fuel assembly design is the basis for presenting all calculated dose rate data as assembly surface area dose rates.

Constant Factors Relating Dose Rate And Photon Source

To a good approximation the calculated surface area dose rate correlates by a constant to the calculated fission product photon source. This simple correlation exists because the major contribution to both the dose rate and the photon source comes from the fission product ^{137m}Ba with minor contributions from its parent ^{137}Cs , and from ^{90}Y and its parent ^{90}Sr . The photon source from these fission product chains includes bremsstrahlung radiation for beta decay in uranium.

Table 1. Ratio of Photon Dose Rate (rem/h) to Photon Source Rate (ph/s)

Assembly Power Density, MW/kg ^{235}U	Average Ratio at 1 m in Air	
	1 to 20% Burnup 2 to 4 y Decay	20 to 80% Burnup 2 to 20 y Decay
2.857	6.75-12 \pm 4.9%	7.44-12 \pm 4.4%
1.429	6.78-12 \pm 4.9%	7.51-12 \pm 3.5% ^a
0.714	6.81-12 \pm 5.1%	7.51-12 \pm 3.7%
0.357	6.83-12 \pm 5.4%	7.53-12 \pm 3.2%
0.179	6.92-12 \pm 5.2%	7.58-12 \pm 2.5%
0.089	7.04-12 \pm 5.2%	7.61-12 \pm 1.9%
Average	6.86-12 \pm 5.1%	7.53-12 \pm 3.3%

^a Average ratio in water at 1 m, 42.5 cm and 40 cm: 4.70-14 \pm 8.0%, 7.60-12 \pm 3.2% and 9.61-12 \pm 3.2%, respectively.

Table 1 shows the ratio of the calculated dose rate in rem/h per g ^{235}U burned to the calculated fission product photon source in ph/s per g ^{235}U burned. The ratio is tabulated as a function of the fuel assembly specific power density for two groupings of burnup and decay time. Ratios for water at a power density of 1.429 MW/kg ^{235}U are given in the footnote.

The use of these dose-to-source ratios can replace the time-consuming task of Monte Carlo dose rate calculations by simple, photon source calculations. The ratios are fairly insensitive to the fuel assembly specific power densities in Table 1. The approximate 4% uncertainty in the average ratio constant will translate into a similar uncertainty in the dose rate.

Line Source Modeling Of Fuel Assembly

The dose rate calculated for a simple, 60-cm long photon line source correlates very well to the radial surface area dose rate calculated for a photon source that is uniformly distributed in an MTR-type fuel assembly. At 1 m in air, the ratio of dose rates for a fuel-assembly model relative to a line-source model is approximately 0.92.

The smaller fuel-assembly model dose rate is due in part to

the shielding of photons by the fuel assembly. Appendix B describes a method to make dose rate calculations using a line-source model.

DOSE RATE RESULTS

Air-Water Dose Rate Correlation

As an alternative to making photon dose rate measurements of spent fuel assemblies in air, a more convenient and natural medium may be a water environment. Water has the advantage that spent fuel assemblies are usually stored in water which provides reasonable shielding for safe dosimetry. Water also provides a natural attenuation for particle decay that can otherwise affect dose rate measurements made in air.

From the data in Table 1, which list the ratio of the surface dose rate to the photon source rate in both air and water, it is possible to establish the location in water where the shielded dose rate will be the same as the unshielded dose rate at 1 m. A linear interpolation of the water ratio data at 40 and 42.5 cm shows that at 41.2 cm, the shielded dose rate in water should be the same as the unshielded dose rate in air.

Self-Protecting Dose Rates

Based upon the calculated dose rates per g²³⁵U burned, the total mass of ²³⁵U burned per fuel assembly that is required to achieve a dose rate of 100 rem/h can be easily determined. Figure 4 shows the mass of ²³⁵U burned that is necessary for a spent fuel assembly to be self-protecting as a function of fission product decay time. These data are for four fuel assembly burnups from 20 to 80%, and for six fuel assembly power densities from 0.089 to 2.857 MW/kg²³⁵U. Interpolation of these data to other fuel assembly burnups and power densities can be easily made.

The data in Fig. 4 show either the minimum mass of ²³⁵U burned per fuel assembly for a given number of self-protected years or the maximum number of years spent fuel will be self-protecting for a given mass of ²³⁵U burned per fuel assembly. To use these figures, three characteristics of the fuel assembly should be known or estimated: (1) - the mass of ²³⁵U burned, (2) - the percentage of ²³⁵U burned, and (3) - the time-average specific power density.

For example, in a fuel assembly with 40% burnup that initially contained 280 g²³⁵U, the mass of ²³⁵U burned is 112 g. If irradiated at a time-averaged power density of 0.089 MW/kg²³⁵U (0.025 MW), this fuel assembly would be self-protecting for a maximum of 4 years after discharge from the reactor. At 2.857 MW/kg²³⁵U (0.8 MW), the self-protection would increase to approximately 7 years after reactor discharge. To increase the number of self-protecting years at a given power density, the fuel assembly burnup would need to be increased. An increase in burnup to 60% (168 g²³⁵U burned) would increase the minimum number of self-protected years from about 10

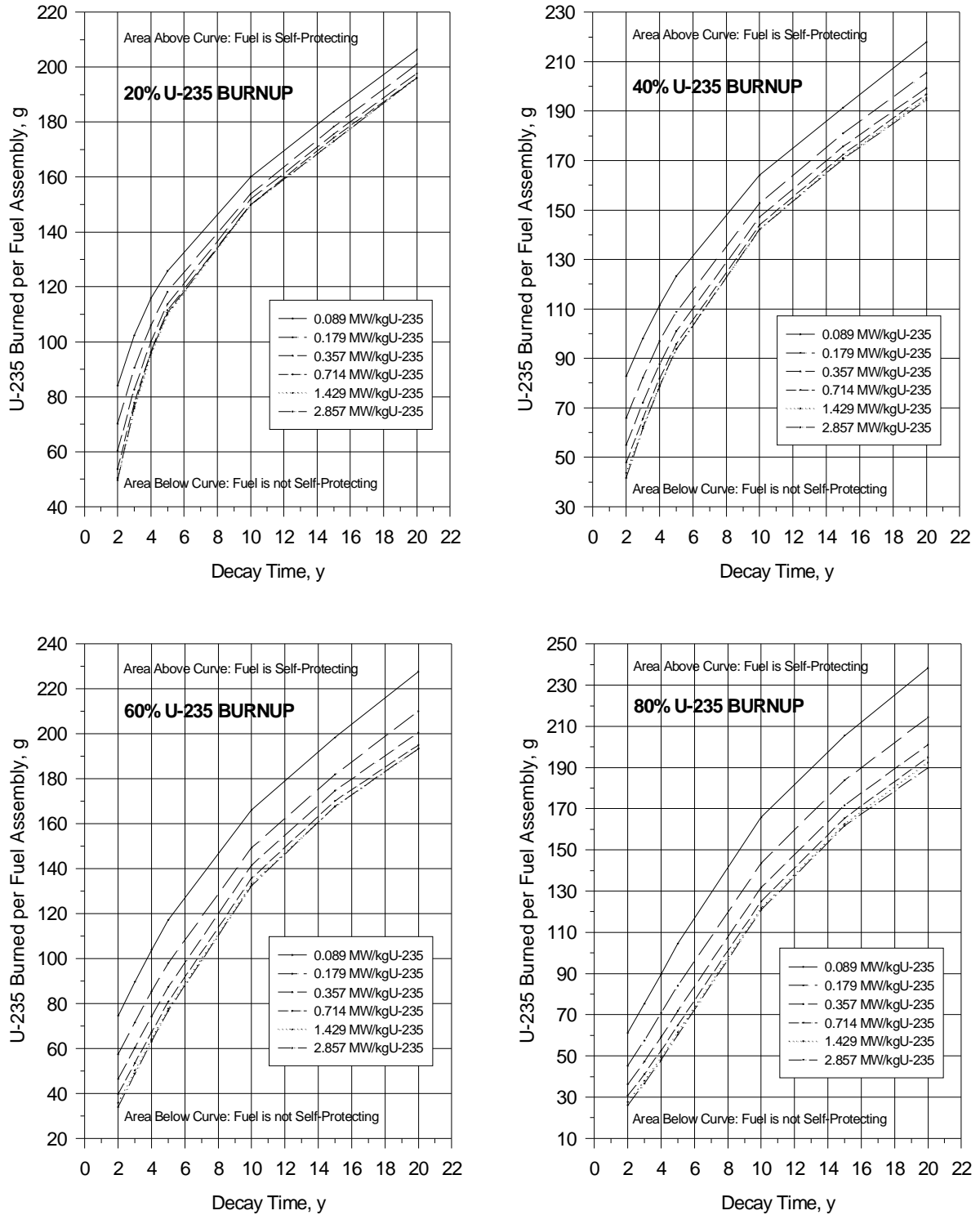


Figure 4. Mass of Burned ^{235}U per Fuel Assembly Necessary for an Unshielded 100rem/h Dose Rate at 1 m for Fuel Assemblies with 20, 40, 60 and 80% ^{235}U Burnup and Power Densities from 0.089 to 2.857 MW/kg ^{235}U

to 15 years for the range of power densities from 0.089 to 2.857 MW/kg²³⁵U.

Throughout this paper the fuel assembly power density has been assumed to be the time-average power density that would be equivalent to the actual fuel assembly irradiation history. This assumption may not be accurate, however, for decay times less than about two years. For decay times less than two years, calculations using actual irradiation histories need to be performed since assemblies retain a "memory" of how they were irradiated. For decay time greater than two years, a fuel assembly loses virtually all memory of its irradiation history and the curves shown in Fig. 4 are independent of the fuel assembly irradiation history.

Revised Dose Rate Results

The Monte Carlo dose rates reported in this paper are a revision of the dose rates reported in Ref. 1. These dose rates are the result of a revised energy group structure used for the MCNP calculations, and the fluence-to-dose conversion factors for the new group structure. The photon dose rates in air (water) are a factor 0.85 (0.98) of the old photon dose rates.

CONCLUSIONS

Based upon the dose rates calculated in this paper, the length of time that a spent fuel assembly will be self-protecting (i.e., having a dose rate that is greater than 100 rem/h at 1 m in air) can be estimated knowing the mass of ²³⁵U burned, the fraction of ²³⁵U burned, and the fuel assembly specific power density. These data for a spent fuel assembly are usually known or can be reasonably estimated. The number of years that a spent fuel assembly will be self-protecting can be read directly from figures that show the g²³⁵U burned as a function of fission product decay time.

These dose rates, however, also can be very sensitive to a number of parameters including such items as the fuel assembly orientation and the location relative to the fuel assembly. Estimates of these dose rate sensitivities have been evaluated for a variety of fuel assembly designs, and shielded (water) and unshielded (air) environments. Of much less importance are dose rate contributions from other sources such as heavy metal formation in the spent fuel, and possible material alloys and impurities in the structure of the fuel assembly. Fission products account for nearly all the dose rate in the spent fuel. These dose rate variations are discussed in Appendix A.

A simplification of dose rate calculations can be made using calculated proportionality factors that relate the spent fuel assembly dose rate to the fission product photon source. This approximation can reduce the dose rate calculation to a photon source calculation times a constant, with a small uncertainty in the result.

Appendix B describes an alternate technique to calculate photon dose rates of spent fuel that is based upon a line-source model of a fuel assembly. The technique is simple, with results that correlate very well to results based upon a detailed model of a fuel assembly and a photon source that is uniformly distributed in the fuel assembly.

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APPENDIX A

DOSE RATE SENSITIVITY DATA

This appendix contains dose rate sensitivity data for various parameters and assumptions inherent in the fuel assembly models. These data also may be used to adjust calculated surface dose rates to compare with measured dose rates that are functions of a fuel assembly design and the fuel assembly orientation.

Face-, Edge- And End-Location Dose Rate Ratios

Table A1 shows the face-, edge- and end-location dose rate ratios relative to the surface area dose rate for three fuel assembly designs. These data may be used to estimate the unshielded, 1-m dose rates at these locations based upon the calculated surface area dose rate.

Table A1. Face-, Edge- and End-Location Dose Rate Ratios

Assembly Type	Dose Rate Ratio ^a with Respect to the Surface Area Dose Rate at 1 m in Air		
	Face	Edge	End
16-Plates	0.972 ± 2%	0.790 ± 2%	0.318 ± 7%
19-Plates	0.976 ± 2%	0.793 ± 2%	0.292 ± 7%
23-Plates	0.940 ± 2%	0.824 ± 2%	0.263 ± 7%

^a For all fuel assembly types at 40 cm in water, the face-, edge- and end-location dose rate ratios with respect to the surface area dose rate are 0.693 ± 2%, 0.686 ± 2% and 0.255 ± 7%, respectively.

Radial Dose Rate Variation

The variation of the surface area dose rate as a function of the distance from the center of the fuel assembly is shown in Table A2. These radial dose rate gradients were calculated for both unshielded and shielded fuel assembly configurations.

Table A2. Radial Dose Rate Gradient

Medium	Percent Change per cm Relative to the Surface Area Dose Rate ^a
Air (1 m)	-1.9 ± 1.0%
Water (40 cm)	-9.8 ± 0.5%

^a Applicable over a radial distance of ± 5 cm relative to a radius of 1 m in air and 40 cm in water.

Axial Dose Rate Variation

Table A3 shows the axial dose rate relative to the average surface dose rate, over five axial sections of the 60-cm active length of a fuel assembly. These results show that the axial dose rate distribution at 1-m radius is fairly flat with only about $\pm 2\%$ variation. At 40-cm radius in water, the central section (-5 to +5 cm) dose rate is about 16% larger than the average dose rate, with similarly large (+10% and -12%) differences in the two other axial sections.

Table A3. Axial Dose Rate Distribution

Medium	Percent Difference Relative to the Surface Area Dose Rate		
	-5 to +5 cm	+5 to +15 cm -5 to -15 cm	+15 to +30 cm -15 to -30 cm
Air (1 m)	2.9 \pm 0.3	1.8 \pm 0.2	-2.2 \pm 0.2
Water (40 cm)	15.8 \pm 0.3	10.2 \pm 0.3	-12.0 \pm 0.2

Aluminum Alloy Materials And Impurities

A number of aluminum alloys containing different alloying materials and impurities are used to manufacture MTR-type fuel assemblies. Table A4 lists a representative sample of the material alloys and impurities commonly found in Al-6061 which is used here as an example. A typical fuel assembly contains about 3 kg of aluminum alloys in the side plates and cladding of the active portion of the fuel plates.

Table A4. Al-6061 Material Alloys and Impurities

Mat'l ppm	Mat'l ppm	Mat'l ppm	Mat'l ppm
Li 10	Ti 300	Co 10	Ga 100
B 10	Cr 1700	Ni 500	Zr 10
Mg 8800	Mn 300	Cu 2600	Cd 10
Si 7000	Fe 2400	Zn 500	Sn 500

Results of dose rate calculations for Al-6061 shows that the major dose rate contributor is ^{60}Co with minor contributions from ^{65}Zn and ^{54}Mn . The latter materials have half-lives of less than a year which leaves ^{60}Co , with a 5.3 y half-life, the only major component. Relative to the dose rate of fission products per gram of ^{235}U burned, the dose rate of cobalt per kilogram of Al-6061 is an order of magnitude smaller.

Heavy Metals

Dose rate calculations were made for the heavy metals generated in the burning of 93% enriched uranium fuel. These data showed that the contribution of heavy metals to the dose rate per gram of ^{235}U burned was five orders of magnitude smaller than the contribution of fission products. Similar dose rate calculations for LEU fuel showed that the heavy metal contribution was larger, but still three orders of magnitude smaller than the contribution of fission products.

APPENDIX B**ANALYTICAL LINE-SOURCE MODEL**

This appendix contains the description of a technique that estimates the photon dose rate of spent fuel, assuming a line-source model of the fission products in a fuel assembly. A comparison of line-source model dose rates to corresponding fuel-assembly model dose rates are also made.

Introduction

Photon dose rate estimates are based upon generating a photon source for the fission products in spent reactor fuel, calculating the photon flux at a point of observation, and correlating the photon flux to a radiation dose. Various measures of the effective radiation dose may be used depending upon recommended biological exposure criteria. For purposes of this work, the most conservative fluence-to-dose estimates from Ref. 5 have been chosen. The purpose of this appendix is to present a simple technique to generate the photon source and to calculate the photon flux. All dose rates are estimated at 1 m in air from the photon source.

The technique generates the photon source of the primary fission products in spent fuel and calculates the photon flux using a Sievert integral. This technique compares to the photon source generated by ORIGEN and the photon flux calculated using MCNP. By modeling the actual, distributed fission product source in the fuel assembly by a line source, the photon flux can be calculated using a much simpler technique. A further simplification of generating only the primary fission products is justified since the majority of all fission products, in the time interval between two and 20 years of fission product decay, are accounted for by just 10 fission products.

Calculations show that the ratio of dose rates of the fuel-assembly model/Monte Carlo technique relative to the line-source model/Sievert integral technique is approximately 0.92. This difference in dose rate is largely the effect of photon shielding by the fuel assembly. Analyses otherwise show good agreement of photon dose rates calculated for a line-source model using either Sievert integral or Monte Carlo techniques.

A Fortran computer program (PHDOSE) is attached to this appendix which may be used to calculate the photon dose rate of spent fuel. The program calculates the photon source for the 10 primary fission products, the Sievert-integral photon flux, and the photon dose rate, given flux-to-dose conversion factors for three photon energy groups. Input to the program requires the fuel assembly irradiation time, specific power density and power, and the fission product decay time in the spent fuel.

Primary Fission Products

The equations for the formation of six fission product pairs $^{90}\text{Sr} - ^{90}\text{Y}$, $^{95}\text{Zr} - ^{95}\text{Nb}$, $^{106}\text{Ru} - ^{106}\text{Rh}$, $^{133}\text{Cs} - ^{134}\text{Cs}$, $^{137}\text{Cs} - ^{137\text{m}}\text{Ba}$, and $^{144}\text{Ce} - ^{144}\text{Pr}$, together with the equation for the formation of ^{60}Co are given below. The equation for ^{60}Co , which is not a fission product, is given since it is a possible photon source in spent fuel arising from the activation of ^{59}Co . If cobalt is present in a fuel assembly, it is usually small and with its fairly short half-life, will not compete with the primary fission products. The two fission product pairs $^{90}\text{Sr} - ^{90}\text{Y}$ and $^{137}\text{Cs} - ^{137\text{m}}\text{Ba}$ are the primary dose rate contributors after a few years of fission product decay. Of the 12 fission products, only 10 actually contribute to the dose rate. While not photon emitters, equations for ^{106}Ru and ^{133}Cs are included since they are the precursors of ^{106}Rh and ^{134}Cs ; direct fission product production of ^{106}Rh or ^{134}Cs is small.

Table B1 lists the physics data that have been used in the equations to estimate the number of atoms of each fission product and cobalt. The number of atoms at any time (t) due to radioactive decay of a material that can neglect radioactive precursors, is given by:

$${}^mN = {}^mN_0 e^{-\lambda_m t}$$

where for material m, λ_m is the decay constant and mN_0 is the number of atoms initially present. When a radioactive parent is present, the number of atoms of a radioactive daughter is given by:

$$dN = \frac{\lambda_p {}^pN_0}{(\lambda_d - \lambda_p)} (e^{-\lambda_p t} - e^{-\lambda_d t}) + dN_0 e^{-\lambda_d t} \approx \frac{\lambda_p {}^pN}{\lambda_d} \quad \text{when } \lambda_d \gg \lambda_p \quad \text{and } e^{-\lambda_d t} \approx 0$$

where the sub- and super-scripts refer to the daughter (d) and parent (p) materials. The number of atoms initially present is given by equations that depend upon the material, the specific power density, and the irradiation time.

Production Of ^{90}Sr , ^{95}Zr , ^{106}Ru , ^{133}Cs , ^{137}Cs And ^{144}Ce

The number of atoms (mN_0) of ^{90}Sr , ^{95}Zr , ^{106}Ru , ^{133}Cs , ^{137}Cs or ^{144}Ce is given by the following equation where during the irradiation time (t_i), the net production rate is proportional to the fission product yield (Y_m) and power, minus the fission product decay (λ_m) and capture (σ_m) rates. The average neutron flux in the equation is proportional to the time-average specific power density during the irradiation time. Units for the power and power density are indicated in square brackets.

$${}^mN_0 = \frac{R_m}{K_m} (1 - e^{-K_m t_i}) \quad \text{with } K_m = (\lambda + \sigma \bar{\phi})_m$$

where $R_m = 3.121e16 \text{ fission/s} \cdot Y_m \text{ atoms/fission} \cdot P [MW]$
and $\bar{\phi} = 2.952e13 \text{ n/cm}^2\text{-s} \cdot \bar{P}_d [MW/kg^{235}U]$

The fission product formation rate (R_m) is based upon a value of 200 MeV per fission and the neutron flux, which is inversely proportional to Σ_f , assumes a value of the ^{235}U fission cross section, $\sigma_f = 412.6 \text{ b}$. The average flux is calculated using the average mass of ^{235}U in determining the specific power density.

Production Of ^{90}Y , ^{95}Nb , ^{106}Rh , ^{137m}Ba And ^{144}Pr

The number of atoms (dN_0) of ^{90}Y , ^{95}Nb , ^{106}Rh , ^{137m}Ba or ^{144}Pr , which are the fission product daughters of ^{90}Sr , ^{95}Zr , ^{106}Ru , ^{137}Cs and ^{144}Ce , are related to the number of atoms (pN_0) of the parents by their decay constants. Because the half-lives of all the daughters (except ^{95}Nb) are very short compared to the parents, the decay rates are in equilibrium. The number of atoms is given by:

$${}^dN_0 = \frac{\lambda_p R_p}{K_p K_d} (1 - e^{-K_d t_i}) - \frac{\lambda_p R_p}{K_p (K_d - K_p)} (e^{-K_p t_i} - e^{-K_d t_i}) \quad \text{with } K_m = (\lambda + \sigma \bar{\phi})_m$$

$${}^dN_0 \approx \frac{\lambda_p {}^pN_0}{K_d} \approx \frac{\lambda_p {}^pN_0}{\lambda_d} \quad \text{when } K_d \approx \lambda_d \gg K_p \quad \text{and } e^{-K_d t_i} \approx 0$$

Production Of ^{134}Cs

The number of atoms of ^{134}Cs is given by the following equation which is based upon the number of ^{133}Cs atoms and a $^{133}Cs(n, \gamma)^{134}Cs$ reaction. The direct fission product production of ^{134}Cs is small and is neglected. The ^{133}Cs production rate R_{133} is given above.

$${}^{134}CS_0 = \frac{R_{133}}{K_{134}} (1 - e^{-K_{134} t_i}) - \frac{R_{133}}{(K_{134} - K_{133})} (e^{-K_{133} t_i} - e^{-K_{134} t_i}) \quad \text{with } K_m = (\lambda + \sigma \bar{\phi})_m$$

Production Of ^{60}Co

The number of atoms of ^{60}Co is given by the following equation which is based upon the number of ^{59}Co atoms and a $^{59}Co(n, \gamma)^{60}Co$ reaction. In the equation ^{59}Co is specified as a fractional part of the fuel assembly (FA) mass.

$${}^{60}\text{Co}_0 = \frac{R_{59}}{K_{60}} (1 - e^{-K_{60}t_i}) \quad \text{with } K_{60} = (\lambda + \sigma\bar{\phi})_{60}$$

where $R_{59} = 1.197e10$ atoms/s per kg FA mass per ppm ${}^{59}\text{Co} \cdot \bar{P}_d$ [MW/kg²³⁵U]

The cobalt reaction rate R_{59} is based upon a ${}^{59}\text{Co}$ capture cross section, $\sigma_c = 39.67$ b.

Photon Source Rate

For material m , the disintegration rate (mD) and the photon source rate (mS_g) as a function of energy, are given by:

$${}^mD = \lambda_m {}^mN \quad \text{and} \quad {}^mS_g = {}^mC_g {}^mD$$

where mC_g is the number of photons emitted per disintegration in energy group g . The energy dependent photon emission constants for each material are included in Table B1. Bremsstrahlung photons from fission product beta decay in UO_2 are included in the emission constants for ${}^{90}\text{Sr}$, ${}^{90}\text{Y}$, ${}^{106}\text{Rh}$, ${}^{137}\text{Cs}$ and ${}^{144}\text{Pr}$.

Photon Dose Rate

If the material m photon source rate (mS_g) is assumed to be distributed in a line source 60-cm long, the photon flux (${}^m\phi_g$) at a perpendicular distance of 1 m in air at the mid-length of the line source, is given by^[7]:

$${}^m\phi_g [\text{ph}/\text{cm}^2\text{-s}] = {}^mS_g [\text{ph}/\text{s}] \cdot F(\theta, \mu t) / (2\pi \cdot 60\text{cm} \cdot 100\text{cm})$$

The function $F(\theta, \mu t)$ is the secant or Sievert integral^[8] which is equal to approximately 0.29 when $\theta = \arctan(30\text{cm}/100\text{cm}) = 16.7^\circ$ and $\mu t = 0.0$ at 1 m in air. (The attenuation coefficient (μ) for air is of the order of $3.35e-5$ cm^{-1} and for $t = 100$ cm, $\mu t = 0.003$). The variation of the attenuation coefficient for air is small over the energy range between 0.3 and 1.1 MeV, and therefore, as a function of energy, the group photon flux is proportional to the photon emission (mC_g). With these data the photon flux is equal to $7.692e-6$ times the photon source rate (mS_g).

The photon dose rate corresponding to a photon flux is given in Table B2 as a function of the photon energy. The data in the third and fourth columns of Table B2 are from Ref. 5, and the flux-to-dose conversion factors in the second column are linear-interpolated values for the photon energies in the first column. For the three photon energies of interest (0.30, 0.63 and 1.10 MeV), the flux-to-dose conversion factors are $5.616e-7$, $1.122e-6$ and $1.774e-6$ rem/h per $\text{ph}/\text{cm}^2\text{-s}$.

ORIGEN Vs. Program-Calculated Photon Source Rates

Total photon source rates generated by ORIGEN and by the above equations for the 10 photon emitting fission products are tabulated in Tables B3 and B4, respectively. These source rates are given in units of ph/s per g²³⁵U burned and as functions of burnup, specific power density, and decay time. The total photon source rate is the sum of the photon source rate at all photon energies. In case of the ORIGEN total source rate, the sum is over all energies from 27.5 keV to 5.25 MeV (18 groups), and for the program-calculated (PHDOSE) total source rate, the sum is over energies from 0.3 to 1.1 MeV (3 groups).

Tables B3 and B4 show that the program-calculated source rates tend to be underestimated for short-decay times and overestimated for long-decay times. The overall agreement of the ORIGEN and the PHDOSE source rates are within a few percent.

Line-Source Model Photon Dose Rates

Calculated dose rates with the ORIGEN and PHDOSE source rates are tabulated in Tables B5 and B6, respectively. These dose rates are based upon line-source model Sievert integrals with 18-energy groups for ORIGEN distributions and with 3-energy groups for PHDOSE distributions. These line-source model dose rates, like the photon source rates, are in similar agreement.

To illustrate the relative agreement of these line-source model/Sievert integral dose rates, dose rates for a fuel assembly with 40% ²³⁵U burnup and a power density of 0.089 MW/kg²³⁵U are plotted in Fig. B1. Figure B1 also shows the Monte Carlo (18-group MCNP) line-source model dose rates calculated with the ORIGEN photon source rate distribution. The MCNP and the Sievert integral dose rates are in near agreement for all decay times. The small difference in dose rate is due in part to the difference in the photon flux calculated by the Monte Carlo and the Sievert integral techniques. The Sievert integral dose rates are slightly larger than the MCNP dose rate. This difference is consistent since the Sievert integral dose rate is for a ring at the mid-point of the 60-cm long line source while the MCNP dose rate is an average over the fuel assembly surface area.

The diamond-shaped data points shown in Fig. B1 at 5 and 10 years, are MCNP calculated dose rates in vacuum, at 1 m from the line source. Within statistics (1σ) the Monte Carlo dose rates in air or in vacuum are the same. Based upon this result, air acts like a void with no photon attenuation or buildup.

Cobalt Photon Dose Rates

Dose rates for ⁶⁰Co calculated with the PHDOSE program are tabulated in Table B7. The dose rates are calculated for the same irradiation conditions as fuel, and therefore, can be compared on a relative basis to the fission product dose rates tabulated in Table B6. The ⁶⁰Co dose rates are per kilogram of assembly mass per

ppm of ^{59}Co .

To illustrate the use of these data, the ^{60}Co dose rate is 0.65 rem/h and the fission product dose rate is 112 rem/h, assuming a 3 kg assembly mass, 10 ppm ^{59}Co and 100 g ^{235}U burned. These dose rates are calculated from data in Tables B6 and B7 for a fuel assembly with 40% ^{235}U burnup, a power density of 0.089 MW/kg ^{235}U , and for 3 years of fission product and ^{60}Co decay. In this example and for most situations, it is expected that the fission product dose rate will exceed the dose rate of ^{60}Co .

Fuel-Assembly Model Photon Dose Rates

Fuel-assembly model dose rates are tabulated in Tables B8 and B9. The dose rates in Table B8 are based upon 23-group MCNP calculations reported in Ref. 1 and the dose rates in Table B9 are based upon new 18-group MCNP calculations. The difference in dose rates is primarily due to the flux-to-dose conversion factors appropriate for the different energy group structures. (The group structure of Ref. 1 is no longer used and the Ref. 1 dose rates should therefore be revised.) The ratio of the new dose rates relative to the Ref. 1 dose rates is 0.85. Figure B2 shows for example, the MCNP calculated dose rates for a fuel assembly with 40% ^{235}U burnup and a power density of 0.089 MW/kg ^{235}U .

For purpose of correlating fuel-assembly model dose rates with line-source model dose rates, the ratio of the MCNP-FA Model dose rates (Fig. B2) relative to the MCNP-LS Model dose rates (Fig. B1) is approximately 0.92.

Self-Protecting Fuel Assembly Masses

Self-protecting fuel assembly masses corresponding to the revised fuel-assembly model dose rates (Table B9) are tabulated in Table B10 and shown in Figs. B3 - B6 for 20 - 80% ^{235}U burnup. Table B11 are self-protecting fuel assembly masses corresponding to the line-source model dose rates (Table B6) calculated with the PHDOSE program. These latter masses have been adjusted by a factor of 0.92 in order to estimate equivalent, fuel-assembly model self-protecting masses. A comparison of these self-protection fuel assembly masses indicate that dose rate estimates using the PHDOSE program are in reasonable agreement with the Monte Carlo dose rate results.

Summary

Equations have been developed that will estimate the photon dose rate of specific fission products in spent (uranium) fuel. The equations are based upon the natural laws of radioactive growth and decay, and material constants (Table B1) applicable to the specific fission products. The fission products include ^{90}Sr , ^{90}Y , ^{95}Zr , ^{95}Nb , ^{106}Rh , ^{134}Cs , ^{137}Cs , $^{137\text{m}}\text{Ba}$, ^{144}Ce and ^{144}Pr which are the 10 primary fission products in spent fuel that have short- and long-term dose rate effects.

The dose rates are calculated at 1 m in air, from a 60-cm long

line source that represent the fission products in a spent fuel assembly. The photon inventory of the primary fission products is calculated as a function of the fuel assembly irradiation time, the specific fuel assembly power density, the initial ^{235}U mass, and the fission product decay time.

The line-source model photon source and dose rates, and self-protecting fuel assembly masses are tabulated in Tables B4, B6 and B11, respectively. These results can be compared to corresponding fuel-assembly model results that are tabulated in Tables B3, B9 and B10, respectively. The ratio of fuel-assembly model dose rates to line-source model dose rates is approximately 0.92.

GAMMA DOSE RATE vs DECAY TIME
as functions of U-235 burnup, 40% and
assembly power density, 0.089 MW/kgU-235

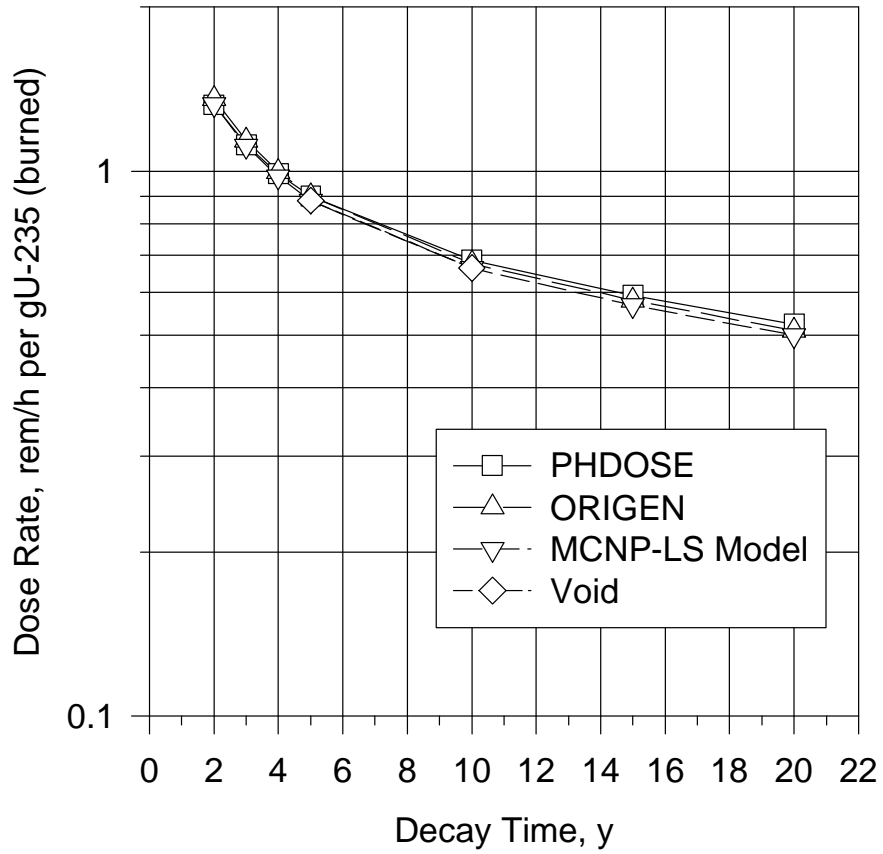


Figure B1. Line-Source Model Photon Dose Rates

GAMMA DOSE RATE vs DECAY TIME
as functions of U-235 burnup, 40% and
assembly power density, 0.089 MW/kgU-235

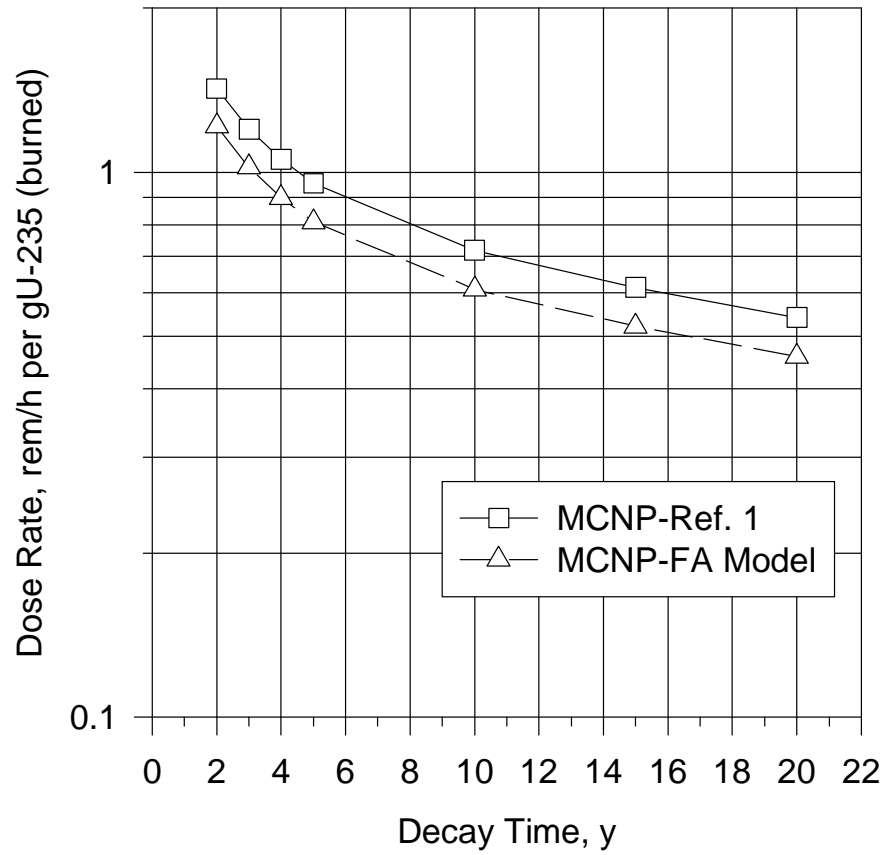


Figure B2. Fuel-Assembly Model Photon Dose Rates

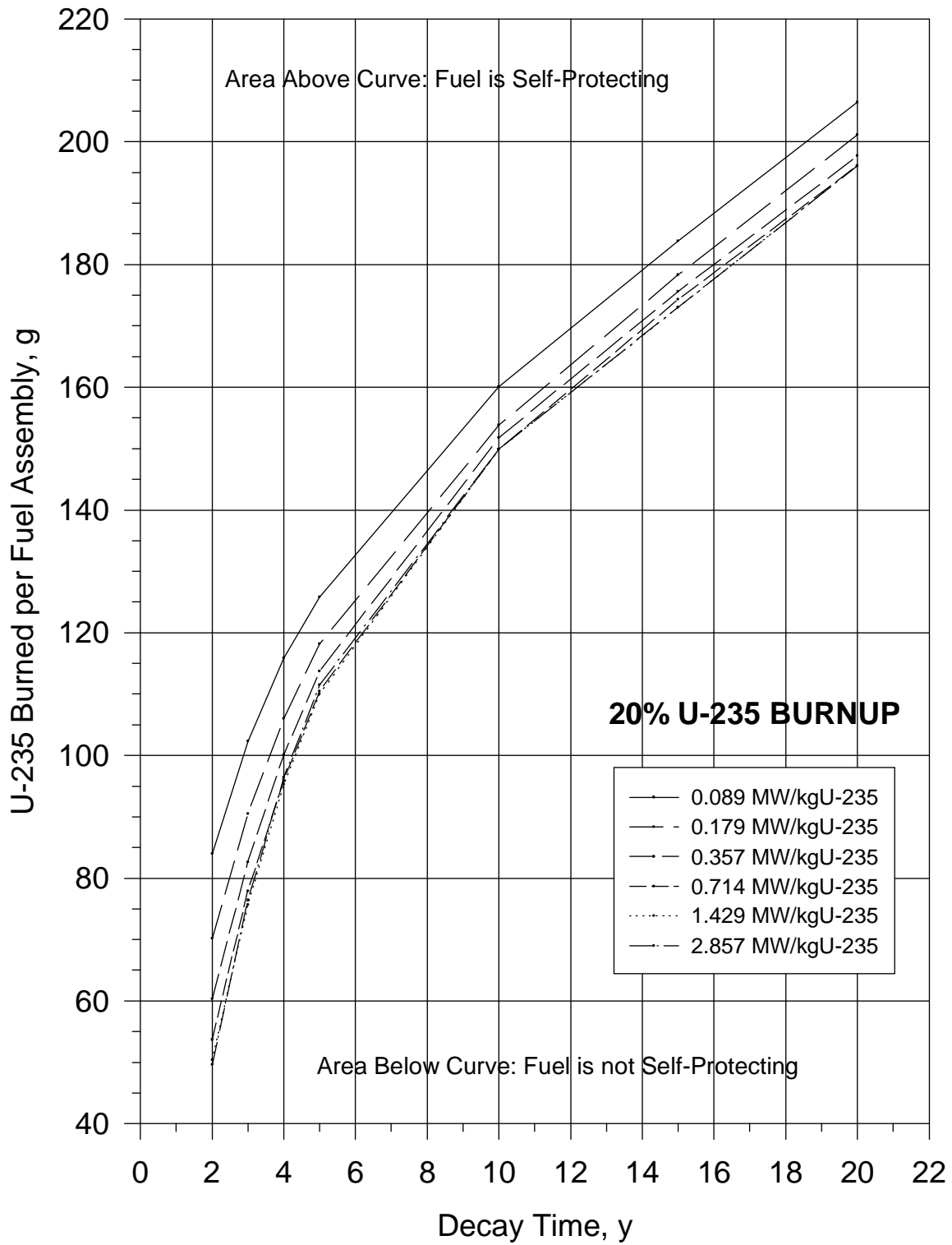


Figure B3. Mass of Burned ^{235}U per Fuel Assembly Necessary for an Unshielded 100 rem/h Dose Rate at 1 m for a Fuel Assembly with 20% ^{235}U Burnup and Power Densities from 0.089 to 2.857 MW/kg ^{235}U

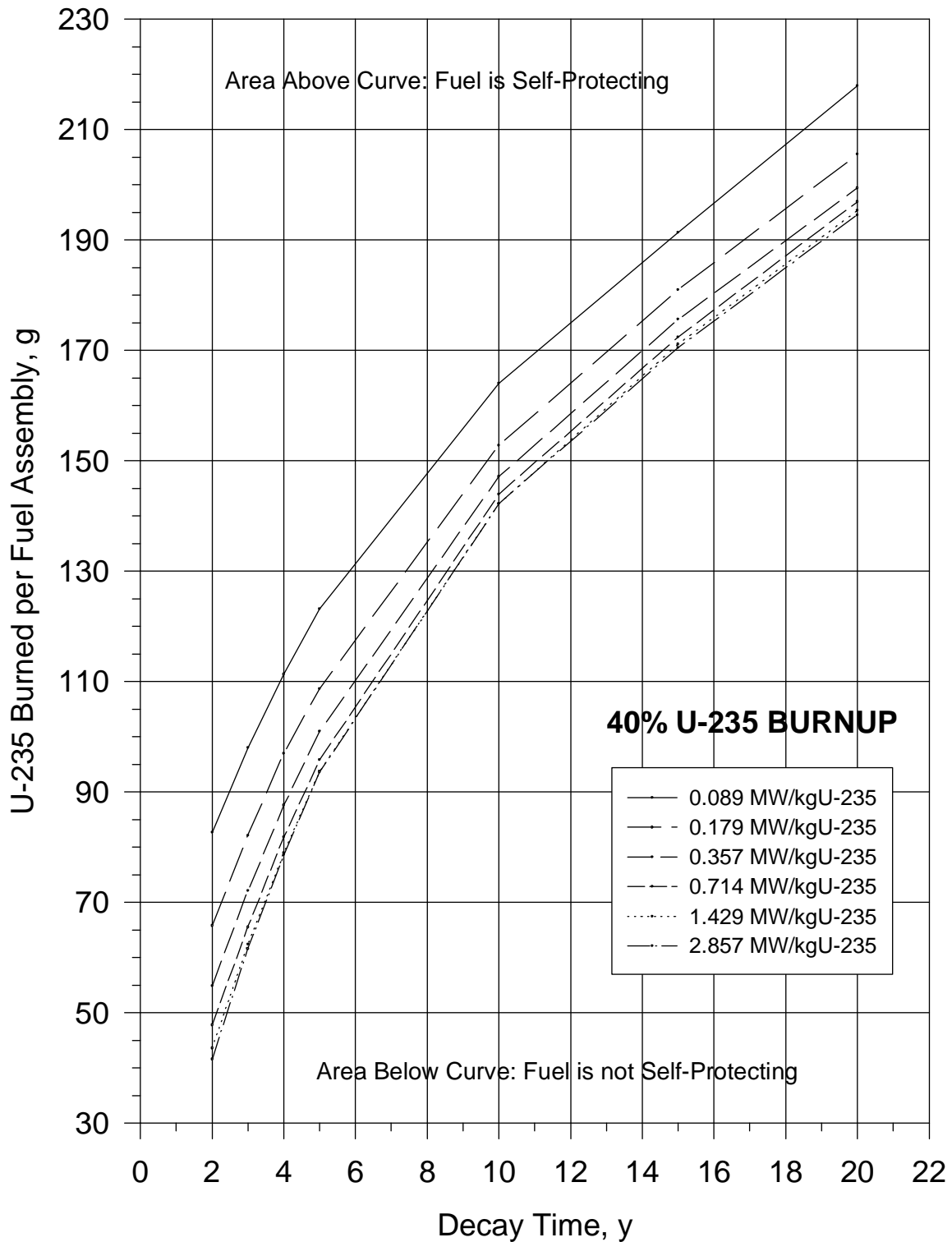


Figure B4. Mass of Burned ^{235}U per Fuel Assembly Necessary for an Unshielded 100 rem/h Dose Rate at 1 m for a Fuel Assembly with 40% ^{235}U Burnup and Power Densities from 0.089 to 2.857 MW/kg ^{235}U

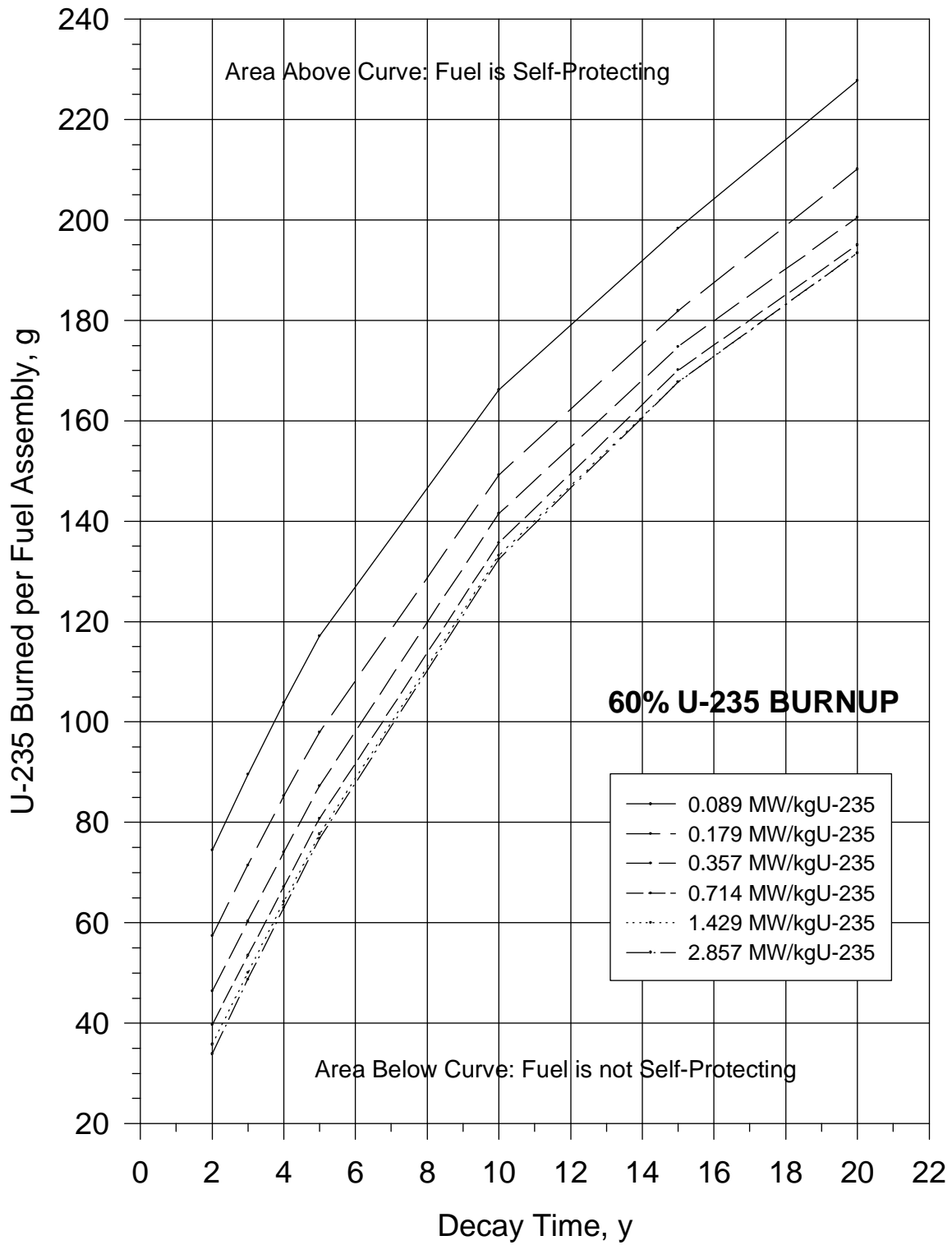


Figure B5. Mass of Burned ^{235}U per Fuel Assembly Necessary for an Unshielded 100 rem/h Dose Rate at 1 m for a Fuel Assembly with 60% ^{235}U Burnup and Power Densities from 0.089 to 2.857 MW/kg ^{235}U

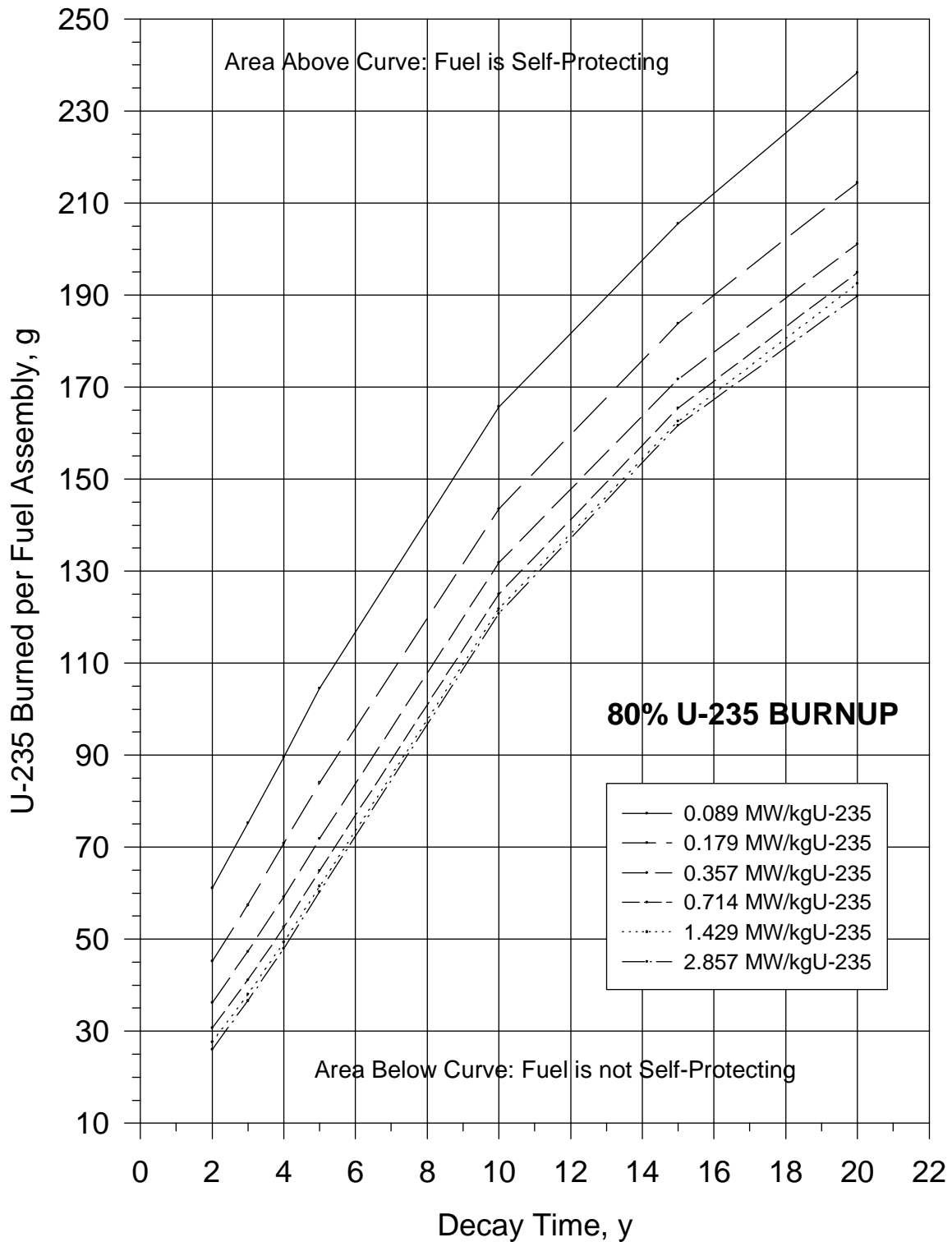


Figure B6. Mass of Burned ^{235}U per Fuel Assembly Necessary for an Unshielded 100 rem/h Dose Rate at 1 m for a Fuel Assembly with 80% ^{235}U Burnup and Power Densities from 0.089 to 2.857 MW/kg ^{235}U

Table B1. Reference^a Material Data

Material	Decay Constant, s ⁻¹	Fission Product Yield, %	Capture Cross Section, b	Energy Group, MeV	Photons/Disintegration
⁹⁰ Sr	7.816-10	5.8	1.028	0.30 0.63	4.69-3 4.39-6
⁹⁰ Y	3.005-6	small	2.740	0.30 0.63 1.10	4.94-2 1.62-2 2.48-3
⁹⁵ Zr	1.225-7	6.50	0.0	0.63	1.15
⁹⁵ Nb	2.285-7	small	3.563	0.63	1.22
¹⁰⁶ Ru	2.174-8	0.401	0.4550		none
¹⁰⁶ Rh	2.318-2	small	0.0	0.30 0.63 1.10	1.53-1 3.43-1 5.20-2
¹³³ Cs	stable	6.69	103.2		none
¹³⁴ Cs	1.066-8	small	107.1	0.63 1.10	2.40 3.10-2
¹³⁷ Cs	7.297-10	6.19	0.1310	0.30 0.63	1.01-2 4.24-5
^{137m} Ba	4.530-3	small	0.0	0.63	1.05
¹⁴⁴ Ce	2.821-8	5.50	1.172	0.30	1.00-1
¹⁴⁴ Pr	6.684-4	small	0.0	0.30 0.63 1.10	7.53-2 4.32-2 8.67-3
⁶⁰ Co	4.176-9	none	4.275	1.10	2.28

^a All data are from Ref. 3, except yield data from Ref. 6.

Table B2. Photon Flux-to-Dose Conversion Vs. Photon Energy

Photon Energy, MeV	Flux-to-Dose Conversion ^a , 10^{-12} Sv-cm ²	Photon Energy, MeV	Flux-to-Dose Conversion ^a , 10^{-12} Sv-cm ²
5.25	15.2	6.0	16.7
4.70	14.04	5.0	14.7
4.22	12.984	4.0	12.5
3.70	11.81	3.0	10.2
3.25	10.775	2.0	7.66
2.75	9.565	1.5	6.24
2.38	8.6252	1.0	4.60
1.99	7.6316	0.8	3.83
1.55	6.382	0.6	2.99
1.10	4.928	0.5	2.54
0.63	3.116	0.4	2.06
0.30	1.56	0.3	1.56
0.20	1.03	0.2	1.03
0.15	0.777	0.15	0.777
0.10	0.533	0.10	0.533
0.0625	0.406375	0.08	0.451
0.0425	0.36975	0.06	0.400
0.0275	0.30625	0.05	0.384
0.0105	0.0715	0.04	0.365
		0.03	0.329
		0.02	0.238
		0.015	0.157
		0.01	0.062
		4.25-3	0.01

^a Sv-cm² = (3.6+5 rem/h)/(ph/cm²-s); Gy \equiv J/kg \equiv 100 rem = Sv.

Table B3. Photon Source Rates, ph/s per g²³⁵U Burned
(ORIGEN: All Fission Products)

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%	2.93e11	2.92e11	2.90e11	2.87e11	2.81e11	2.69e11
3	1.70e11	1.70e11	1.70e11	1.69e11	1.68e11	1.65e11
4	1.28e11	1.28e11	1.28e11	1.28e11	1.28e11	1.26e11
2 10%	2.96e11	2.92e11	2.81e11	2.59e11	2.27e11	1.89e11
3	1.76e11	1.77e11	1.75e11	1.68e11	1.56e11	1.39e11
4	1.34e11	1.34e11	1.34e11	1.31e11	1.25e11	1.17e11
2 20%	3.11e11	3.04e11	2.82e11	2.48e11	2.07e11	1.67e11
3	1.91e11	1.91e11	1.84e11	1.72e11	1.54e11	1.33e11
4	1.44e11	1.45e11	1.43e11	1.36e11	1.27e11	1.15e11
5	1.21e11	1.22e11	1.21e11	1.18e11	1.12e11	1.05e11
10	8.77e10	8.80e10	8.79e10	8.70e10	8.55e10	8.25e10
15	7.59e10	7.59e10	7.57e10	7.52e10	7.41e10	7.20e10
20	6.73e10	6.71e10	6.70e10	6.66e10	6.55e10	6.38e10
2 40%	3.55e11	3.36e11	3.01e11	2.56e11	2.08e11	1.63e11
3	2.29e11	2.23e11	2.10e11	1.88e11	1.63e11	1.35e11
4	1.72e11	1.71e11	1.63e11	1.52e11	1.36e11	1.18e11
5	1.42e11	1.41e11	1.38e11	1.30e11	1.21e11	1.06e11
10	9.20e10	9.20e10	9.11e10	8.93e10	8.59e10	8.02e10
15	7.71e10	7.68e10	7.63e10	7.51e10	7.29e10	6.88e10
20	6.76e10	6.73e10	6.68e10	6.60e10	6.40e10	6.06e10
2 60%	4.20e11	3.92e11	3.48e11	2.90e11	2.32e11	1.76e11
3	2.80e11	2.70e11	2.50e11	2.19e11	1.83e11	1.45e11
4	2.11e11	2.06e11	1.95e11	1.76e11	1.52e11	1.25e11
5	1.70e11	1.68e11	1.61e11	1.49e11	1.32e11	1.11e11
10	9.82e10	9.76e10	9.58e10	9.23e10	8.75e10	7.86e10
15	7.80e10	7.80e10	7.68e10	7.50e10	7.20e10	6.61e10
20	6.79e10	6.79e10	6.73e10	6.55e10	6.25e10	5.77e10
2 80%	5.27e11	4.91e11	4.34e11	3.63e11	2.86e11	2.11e11
3	3.62e11	3.46e11	3.17e11	2.74e11	2.24e11	1.71e11
4	2.71e11	2.63e11	2.45e11	2.17e11	1.82e11	1.43e11
5	2.14e11	2.09e11	1.97e11	1.78e11	1.53e11	1.23e11
10	1.07e11	1.06e11	1.03e11	9.82e10	9.02e10	7.81e10
15	8.04e10	7.99e10	7.86e10	7.59e10	7.10e10	6.34e10
20	6.88e10	6.79e10	6.70e10	6.52e10	6.12e10	5.49e10

Table B4. Photon Source Rates, ph/s per g²³⁵U Burned
 (FP's: ⁹⁰Sr ⁹⁰Y ⁹⁵Zr ⁹⁵Nb ¹⁰⁶Rh ¹³⁴Cs ¹³⁷Cs ^{137m}Ba ¹⁴⁴Ce ¹⁴⁴Pr)

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%	2.90e11	2.89e11	2.87e11	2.83e11	2.77e11	2.65e11
3	1.71e11	1.70e11	1.70e11	1.69e11	1.67e11	1.64e11
4	1.30e11	1.30e11	1.30e11	1.29e11	1.28e11	1.27e11
2 10%	3.01e11	2.92e11	2.78e11	2.56e11	2.24e11	1.87e11
3	1.82e11	1.80e11	1.76e11	1.68e11	1.56e11	1.40e11
4	1.39e11	1.38e11	1.36e11	1.32e11	1.26e11	1.18e11
2 20%	3.16e11	3.01e11	2.78e11	2.45e11	2.05e11	1.66e11
3	1.97e11	1.92e11	1.84e11	1.71e11	1.53e11	1.33e11
4	1.50e11	1.47e11	1.44e11	1.37e11	1.28e11	1.16e11
5	1.26e11	1.25e11	1.23e11	1.19e11	1.14e11	1.06e11
10	9.07e10	9.05e10	9.00e10	8.92e10	8.77e10	8.48e10
15	7.85e10	7.83e10	7.80e10	7.75e10	7.63e10	7.42e10
20	6.95e10	6.94e10	6.91e10	6.87e10	6.77e10	6.58e10
2 40%	3.57e11	3.32e11	2.97e11	2.52e11	2.04e11	1.61e11
3	2.32e11	2.23e11	2.08e11	1.86e11	1.60e11	1.34e11
4	1.76e11	1.71e11	1.63e11	1.51e11	1.36e11	1.18e11
5	1.45e11	1.42e11	1.38e11	1.30e11	1.20e11	1.07e11
10	9.43e10	9.37e10	9.27e10	9.08e10	8.74e10	8.16e10
15	7.90e10	7.87e10	7.81e10	7.69e10	7.46e10	7.04e10
20	6.95e10	6.93e10	6.88e10	6.78e10	6.59e10	6.23e10
2 60%	4.19e11	3.85e11	3.38e11	2.82e11	2.25e11	1.71e11
3	2.81e11	2.66e11	2.44e11	2.13e11	1.78e11	1.42e11
4	2.12e11	2.04e11	1.91e11	1.73e11	1.49e11	1.23e11
5	1.71e11	1.66e11	1.58e11	1.46e11	1.30e11	1.10e11
10	9.91e10	9.81e10	9.63e10	9.30e10	8.75e10	7.90e10
15	7.98e10	7.93e10	7.83e10	7.65e10	7.31e10	6.70e10
20	6.96e10	6.92e10	6.84e10	6.70e10	6.42e10	5.91e10
2 80%	5.14e11	4.70e11	4.12e11	3.42e11	2.69e11	1.98e11
3	3.52e11	3.32e11	3.00e11	2.59e11	2.11e11	1.61e11
4	2.64e11	2.52e11	2.33e11	2.06e11	1.73e11	1.36e11
5	2.09e11	2.01e11	1.89e11	1.70e11	1.46e11	1.18e11
10	1.06e11	1.04e11	1.02e11	9.67e10	8.89e10	7.75e10
15	8.10e10	8.03e10	7.89e10	7.64e10	7.18e10	6.40e10
20	6.97e10	6.92e10	6.81e10	6.62e10	6.26e10	5.61e10

Table B5. Photon Dose Rates in Air, rem/h per g²³⁵U Burned
 (ORIGEN: All Fission Products - Sievert Integral)
 (Line-Source Model Dose Rates)

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%	2.16+0	2.16+0	2.15+0	2.12+0	2.08+0	1.99+0
3	1.30+0	1.30+0	1.30+0	1.30+0	1.29+0	1.27+0
4	1.02+0	1.02+0	1.02+0	1.02+0	1.02+0	1.01+0
2 10%	2.21+0	2.18+0	2.11+0	1.95+0	1.73+0	1.47+0
3	1.36+0	1.37+0	1.37+0	1.32+0	1.24+0	1.12+0
4	1.07+0	1.08+0	1.08+0	1.07+0	1.02+0	9.63-1
2 20%	2.34+0	2.30+0	2.16+0	1.92+0	1.64+0	1.35+0
3	1.49+0	1.50+0	1.46+0	1.38+0	1.25+0	1.10+0
4	1.17+0	1.18+0	1.17+0	1.12+0	1.05+0	9.64-1
5	9.99-1	1.01+0	1.01+0	9.83-1	9.40-1	8.81-1
10	7.37-1	7.40-1	7.38-1	7.32-1	7.19-1	6.94-1
15	6.38-1	6.38-1	6.37-1	6.32-1	6.22-1	6.05-1
20	5.65-1	5.64-1	5.62-1	5.59-1	5.51-1	5.35-1
2 40%	2.76+0	2.63+0	2.39+0	2.07+0	1.71+0	1.36+0
3	1.84+0	1.81+0	1.72+0	1.56+0	1.36+0	1.14+0
4	1.42+0	1.42+0	1.37+0	1.28+0	1.16+0	9.97-1
5	1.19+0	1.18+0	1.16+0	1.10+0	1.02+0	8.99-1
10	7.77-1	7.78-1	7.70-1	7.55-1	7.25-1	6.77-1
15	6.49-1	6.47-1	6.42-1	6.32-1	6.14-1	5.79-1
20	5.68-1	5.66-1	5.62-1	5.55-1	5.39-1	5.10-1
2 60%	3.36+0	3.17+0	2.85+0	2.42+0	1.95+0	1.49+0
3	2.31+0	2.24+0	2.09+0	1.85+0	1.56+0	1.24+0
4	1.77+0	1.74+0	1.65+0	1.50+0	1.30+0	1.07+0
5	1.44+0	1.43+0	1.37+0	1.27+0	1.13+0	9.44-1
10	8.33-1	8.28-1	8.12-1	7.85-1	7.41-1	6.65-1
15	6.61-1	6.61-1	6.50-1	6.34-1	6.07-1	5.58-1
20	5.72-1	5.72-1	5.66-1	5.50-1	5.29-1	4.87-1
2 80%	4.35+0	4.08+0	3.65+0	3.09+0	2.45+0	1.82+0
3	3.05+0	2.93+0	2.70+0	2.35+0	1.93+0	1.47+0
4	2.31+0	2.25+0	2.10+0	1.87+0	1.57+0	1.23+0
5	1.83+0	1.80+0	1.70+0	1.54+0	1.32+0	1.06+0
10	9.15-1	9.05-1	8.81-1	8.39-1	7.70-1	6.68-1
15	6.83-1	6.78-1	6.66-1	6.45-1	6.04-1	5.39-1
20	5.80-1	5.76-1	5.67-1	5.51-1	5.18-1	4.66-1

Table B6. Photon Dose Rates in Air, rem/h per g²³⁵U Burned
 (FP's: ⁹⁰Sr ⁹⁰Y ⁹⁵Zr ⁹⁵Nb ¹⁰⁶Rh ¹³⁴Cs ¹³⁷Cs ^{137m}Ba ¹⁴⁴Ce ¹⁴⁴Pr)
 (Line-Source Model Dose Rates)

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%	2.05+0	2.05+0	2.04+0	2.01+0	1.96+0	1.88+0
3	1.28+0	1.27+0	1.27+0	1.27+0	1.25+0	1.23+0
4	1.03+0	1.02+0	1.02+0	1.02+0	1.02+0	1.01+0
2 10%	2.16+0	2.11+0	2.01+0	1.86+0	1.65+0	1.42+0
3	1.38+0	1.37+0	1.34+0	1.29+0	1.22+0	1.11+0
4	1.10+0	1.10+0	1.08+0	1.06+0	1.02+0	9.69-1
2 20%	2.31+0	2.21+0	2.05+0	1.84+0	1.58+0	1.32+0
3	1.51+0	1.48+0	1.43+0	1.35+0	1.23+0	1.09+0
4	1.20+0	1.19+0	1.16+0	1.12+0	1.05+0	9.67-1
5	1.04+0	1.03+0	1.01+0	9.87-1	9.47-1	8.89-1
10	7.62-1	7.60-1	7.57-1	7.50-1	7.37-1	7.13-1
15	6.59-1	6.58-1	6.56-1	6.51-1	6.41-1	6.23-1
20	5.84-1	5.83-1	5.81-1	5.77-1	5.69-1	5.53-1
2 40%	2.69+0	2.52+0	2.28+0	1.98+0	1.65+0	1.32+0
3	1.83+0	1.77+0	1.67+0	1.51+0	1.33+0	1.12+0
4	1.44+0	1.40+0	1.34+0	1.26+0	1.14+0	9.88-1
5	1.20+0	1.19+0	1.15+0	1.09+0	1.01+0	8.99-1
10	7.93-1	7.89-1	7.80-1	7.64-1	7.35-1	6.87-1
15	6.64-1	6.62-1	6.56-1	6.46-1	6.27-1	5.92-1
20	5.84-1	5.82-1	5.78-1	5.70-1	5.54-1	5.24-1
2 60%	3.25+0	3.02+0	2.70+0	2.30+0	1.86+0	1.43+0
3	2.26+0	2.16+0	2.00+0	1.77+0	1.49+0	1.20+0
4	1.75+0	1.69+0	1.59+0	1.45+0	1.26+0	1.04-1
5	1.43+0	1.40+0	1.33+0	1.23+0	1.10+0	9.26-1
10	8.35-1	8.27-1	8.11-1	7.84-1	7.37-1	6.65-1
15	6.71-1	6.67-1	6.59-1	6.43-1	6.14-1	5.63-1
20	5.85-1	5.81-1	5.75-1	5.63-1	5.39-1	4.97-1
2 80%	4.10+0	3.79+0	3.37+0	2.85+0	2.26+0	1.67+0
3	2.89+0	2.74+0	2.51+0	2.18+0	1.79+0	1.36+0
4	2.21+0	2.12+0	1.97+0	1.75+0	1.47+0	1.15+0
5	1.76+0	1.70+0	1.60+0	1.45+0	1.24+0	9.98-1
10	8.96-1	8.83-1	8.58-1	8.17-1	7.50-1	6.53-1
15	6.81-1	6.75-1	6.64-1	6.42-1	6.04-1	5.38-1
20	5.86-1	5.81-1	5.73-1	5.56-1	5.26-1	4.72-1

Table B7. ^{60}Co Dose Rates in Air, rem/h per kg per ppm ^{59}Co
(Line-Source Model Dose Rates)

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%	8.30-4	8.29-4	8.28-4	8.27-4	8.23-4	8.17-4
3	7.27-4	7.27-4	7.26-4	7.25-4	7.22-4	7.16-4
4	6.38-4	6.37-4	6.37-4	6.35-4	6.33-4	6.28-4
2 10%	8.66-3	8.61-3	8.53-3	8.36-3	8.04-3	7.45-3
3	7.59-3	7.55-3	7.48-3	7.33-3	7.05-3	6.53-3
4	6.65-3	6.62-3	6.55-3	6.43-3	6.18-3	5.73-3
2 20%	1.82-2	1.81-2	1.77-2	1.71-2	1.58-2	1.37-2
3	1.60-2	1.58-2	1.55-2	1.49-2	1.39-2	1.20-2
4	1.40-2	1.39-2	1.36-2	1.31-2	1.22-2	1.05-2
5	1.23-2	1.22-2	1.19-2	1.15-2	1.07-2	9.24-3
10	6.36-3	6.30-3	6.18-3	5.95-3	5.52-3	4.78-3
15	3.29-3	3.26-3	3.20-3	3.08-3	2.86-3	2.48-3
20	1.70-3	1.69-3	1.66-3	1.59-3	1.48-3	1.28-3
2 40%	4.13-2	4.06-2	3.91-2	3.65-2	3.18-2	2.49-2
3	3.62-2	3.56-2	3.43-2	3.20-2	2.79-2	2.18-2
4	3.18-2	3.12-2	3.01-2	2.80-2	2.45-2	1.91-2
5	2.78-2	2.73-2	2.64-2	2.46-2	2.15-2	1.68-2
10	1.44-2	1.41-2	1.36-2	1.27-2	1.11-2	8.68-3
15	7.46-3	7.32-3	7.06-3	6.58-3	5.75-3	4.49-3
20	3.86-3	3.79-3	3.66-3	3.41-3	2.98-3	2.33-3
2 60%	7.34-2	7.16-2	6.81-2	6.19-2	5.18-2	3.81-2
3	6.44-2	6.27-2	5.97-2	5.42-2	4.54-2	3.34-2
4	5.64-2	5.50-2	5.23-2	4.75-2	3.98-2	2.92-2
5	4.95-2	4.82-2	4.59-2	4.17-2	3.49-2	2.56-2
10	2.56-2	2.50-2	2.37-2	2.16-2	1.81-2	1.33-2
15	1.33-2	1.29-2	1.23-2	1.12-2	9.34-3	6.87-3
20	6.86-3	6.69-3	6.36-3	5.78-3	4.84-3	3.56-3
2 80%	1.28-1	1.24-1	1.17-1	1.05-1	8.64-2	6.28-2
3	1.12-1	1.09-1	1.03-1	9.21-2	7.57-2	5.51-2
4	9.81-2	9.52-2	8.99-2	8.07-2	6.64-2	4.83-2
5	8.60-2	8.35-2	7.88-2	7.07-2	5.82-2	4.23-2
10	4.45-2	4.32-2	4.08-2	3.66-2	3.01-2	2.19-2
15	2.30-2	2.24-2	2.11-2	1.90-2	1.56-2	1.13-2
20	1.19-2	1.16-2	1.09-2	9.81-3	8.07-3	5.87-3

Table B8. Photon Dose Rates in Air, rem/h per g²³⁵U Burned
(ORIGEN: All Fission Products - MCNP)
(Fuel-Assembly Model Dose Rates)

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%	2.17+0	2.16+0	2.15+0	2.12+0	2.08+0	2.00+0
3	1.33+0	1.33+0	1.33+0	1.33+0	1.31+0	1.31+0
4	1.06+0	1.06+0	1.06+0	1.06+0	1.06+0	1.05+0
2 10%	2.22+0	2.20+0	2.12+0	1.93+0	1.76+0	1.50+0
3	1.40+0	1.41+0	1.41+0	1.36+0	1.28+0	1.17+0
4	1.12+0	1.13+0	1.13+0	1.11+0	1.07+0	1.01+0
2 20%	2.37+0	2.34+0	2.19+0	1.95+0	1.68+0	1.40+0
3	1.54+0	1.56+0	1.51+0	1.43+0	1.30+0	1.15+0
4	1.22+0	1.24+0	1.23+0	1.18+0	1.11+0	1.02+0
5	1.06+0	1.07+0	1.07+0	1.04+0	9.95-1	9.35-1
10	7.85-1	7.85-1	7.85-1	7.75-1	7.65-1	7.35-1
15	6.80-1	6.80-1	6.75-1	6.70-1	6.60-1	6.40-1
20	6.00-1	6.00-1	6.00-1	5.95-1	5.85-1	5.70-1
2 40%	2.83+0	2.70+0	2.46+0	2.14+0	1.79+0	1.42+0
3	1.91+0	1.89+0	1.80+0	1.63+0	1.43+0	1.20+0
4	1.50+0	1.49+0	1.44+0	1.34+0	1.21+0	1.06+0
5	1.26+0	1.26+0	1.23+0	1.17+0	1.08+0	9.55-1
10	8.28-1	8.28-1	8.18-1	8.00-1	7.70-1	7.18-1
15	6.90-1	6.88-1	6.83-1	6.70-1	6.50-1	6.15-1
20	6.05-1	6.03-1	5.98-1	5.90-1	5.73-1	5.40-1
2 60%	3.47+0	3.28+0	2.97+0	2.53+0	2.05+0	1.58+0
3	2.42+0	2.35+0	2.20+0	1.95+0	1.65+0	1.31+0
4	1.87+0	1.83+0	1.75+0	1.59+0	1.38+0	1.13+0
5	1.53+0	1.51+0	1.46+0	1.35+0	1.20+0	1.00+0
10	8.88-1	8.83-1	8.67-1	8.32-1	7.88-1	7.08-1
15	7.02-1	7.02-1	6.92-1	6.73-1	6.47-1	5.93-1
20	6.08-1	6.08-1	6.03-1	5.87-1	5.60-1	5.17-1
2 80%	4.53+0	4.26+0	3.84+0	3.25+0	2.60+0	1.93+0
3	3.21+0	3.10+0	2.86+0	2.49+0	2.05+0	1.56+0
4	2.45+0	2.39+0	2.24+0	1.99+0	1.66+0	1.31+0
5	1.95+0	1.91+0	1.81+0	1.64+0	1.40+0	1.13+0
10	9.74-1	9.66-1	9.41-1	8.93-1	8.20-1	7.10-1
15	7.28-1	7.24-1	7.11-1	6.85-1	6.40-1	5.73-1
20	6.20-1	6.11-1	6.04-1	5.85-1	5.49-1	4.94-1

Table B9. Photon Dose Rates in Air, rem/h per g²³⁵U Burned
 (ORIGEN: All Fission Products - MCNP)
 (Revised Fuel-Assembly Model Dose Rates)

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

Table B10. Self-Protecting U-235 Mass, $g^{235}\text{U}$ Burned per 100 rem/h
 (ORIGEN: All Fission Products - MCNP)
 (Revised Fuel-Assembly Model Dose Rates)

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%	54.2	54.5	54.7	55.5	56.6	58.8
3	88.5	88.5	88.5	88.5	89.8	89.8
4	111.0	111.0	111.0	111.0	111.0	112.0
2 10%	53.0	53.5	55.5	61.0	66.8	78.4
3	84.0	83.4	83.4	86.5	91.9	100.6
4	105.0	104.1	104.1	106.0	110.0	116.5
2 20%	49.6	50.4	53.7	60.3	70.2	84.0
3	76.4	75.7	77.9	82.6	90.5	102.3
4	96.4	95.3	96.0	100.1	106.0	115.9
5	111.5	110.0	110.5	113.7	118.2	125.8
10	149.9	149.9	149.9	151.8	153.8	160.1
15	173.0	173.0	174.3	175.6	178.3	183.8
20	196.1	196.1	196.1	197.7	201.1	206.4
2 40%	41.6	43.6	47.8	54.9	65.8	82.7
3	61.6	62.4	65.5	72.1	82.1	98.0
4	78.6	79.0	81.8	87.6	97.0	111.3
5	93.6	93.7	95.8	101.0	108.7	123.2
10	142.2	142.2	143.9	147.1	152.8	164.0
15	170.5	171.1	172.4	175.6	181.0	191.3
20	194.5	195.3	196.9	199.4	205.5	217.9
2 60%	33.9	35.8	39.7	46.4	57.4	74.5
3	48.7	50.1	53.5	60.3	71.5	89.6
4	63.0	64.2	67.2	74.1	85.3	103.8
5	76.8	77.7	80.8	87.3	98.0	117.1
10	132.4	133.2	135.7	141.5	149.2	166.1
15	167.7	167.7	170.1	174.7	181.9	198.3
20	193.4	193.4	195.0	200.5	210.1	227.7
2 80%	26.0	27.6	30.7	36.2	45.2	61.1
3	36.6	38.0	41.1	47.3	57.4	75.3
4	48.0	49.3	52.6	59.2	70.8	89.6
5	60.3	61.5	64.9	71.8	84.0	104.5
10	120.8	121.8	125.0	131.8	143.5	165.7
15	161.7	162.6	165.4	171.7	183.8	205.5
20	189.8	192.5	194.9	201.1	214.4	238.3

Table B11. Self-Protecting U-235 Mass, $g^{235}\text{U}$ Burned per 100 rem/h
 (FP's: ^{90}Sr ^{90}Y ^{95}Zr ^{95}Nb ^{106}Rh ^{134}Cs ^{137}Cs $^{137\text{m}}\text{Ba}$ ^{144}Ce ^{144}Pr)
 (Equivalent Fuel-Assembly Model Dose Rates)

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%	52.9	53.1	53.4	54.1	55.3	57.7
3	85.3	85.3	85.5	85.9	86.6	88.1
4	106.0	106.1	106.2	106.5	107.0	108.0
2 10%	50.2	51.6	54.1	58.5	65.7	76.7
3	78.7	79.5	81.1	84.1	89.4	97.8
4	98.5	99.1	100.2	102.4	106.2	112.2
2 20%	47.1	49.2	53.0	59.2	68.9	82.7
3	71.9	73.3	76.0	80.7	88.5	99.8
4	90.5	91.6	93.6	97.3	103.4	112.4
5	105.0	105.8	107.3	110.1	114.8	122.2
10	142.6	143.0	143.6	145.0	147.5	152.5
15	164.8	165.2	165.8	167.0	169.5	174.5
20	186.1	186.4	187.1	188.4	191.1	196.6
2 40%	40.3	43.1	47.6	54.9	66.1	82.2
3	59.4	61.5	65.3	71.8	82.0	97.2
4	75.7	77.5	80.8	86.5	95.7	110.0
5	90.2	91.7	94.5	99.4	107.7	120.9
10	137.0	137.8	139.3	142.3	147.8	158.3
15	163.6	164.3	165.6	168.2	173.3	183.7
20	186.1	186.8	188.1	190.8	196.3	207.6
2 60%	33.4	36.0	40.3	47.4	58.5	76.0
3	48.0	50.3	54.4	61.3	72.7	90.9
4	62.1	64.2	68.2	75.0	86.3	104.7
5	75.9	77.9	81.6	88.1	99.2	117.4
10	130.2	131.4	134.0	138.7	147.4	163.5
15	162.0	163.0	165.0	169.0	177.0	192.9
20	185.9	187.0	189.0	193.2	201.6	218.9
2 80%	26.5	28.7	32.2	38.2	48.1	65.0
3	37.6	39.6	43.3	49.8	60.8	79.6
4	49.3	51.4	55.3	62.2	74.1	94.4
5	61.7	63.9	67.9	75.1	87.6	108.9
10	121.4	123.2	126.6	133.1	144.9	166.5
15	159.5	160.9	163.8	169.3	180.1	201.9
20	185.6	187.0	189.8	195.4	206.7	230.4

PHDOSE PROGRAM

Purpose To calculate the photon dose rate from spent fuel.

Input (format 7D10.5)

ti Irradiation time (d)
 - if ti = 0.0, ti is calculated from the powden and
 burnup parameters as: $ti = \text{burnup}/\text{powden} \cdot 8$, assum-
 ing 1.25 g²³⁵U burned per MWd.
 - if ti ≥ 0.0, normal output is printed.
 - if ti < 0.0, ti = -ti and optional output is also
 printed.
 - note: if ti ≠ 0.0, ti must be within 1% of the
 calculated irradiation time. (≥ 1%, stop code 101)

power Assembly power (MW)
 powden Assembly power density (MW/kg²³⁵U)
 td Fission product decay time (y)
 kg Structural material mass (kg)
 ppm ⁵⁹Co in structural material (ppm)
 burnup ²³⁵U burnup in fuel (%)

Normal Output

Record #1:

- Echo input (ti, power, powden, td, kg, ppm, burnup)
- Average neutron flux (n/cm²-s)

Record #2:

- ²³⁵U burned (g) [= power/powden*burnup*10]
- Fission product photon source rate (ph/s) per g²³⁵U burned
- ⁶⁰Co photon source rate (ph/s)
- Self-protecting ²³⁵U mass, (g²³⁵U burned)
- Fission product photon dose rate (rem/h) per g²³⁵U burned
- ⁶⁰Co photon dose rate (rem/h)

Fission products (FP):

⁹⁰Sr, ⁹⁰Y, ¹³⁴Cs, ¹³⁷Cs, ^{137m}Ba, ¹⁴⁴Ce, ¹⁴⁴Pr, ¹⁰⁶Rh, ⁹⁵Zr, ⁹⁵Nb

Optional Output

- Isotope formation rates (atoms/s)
- Isotope loss rates (atoms/s)
- Isotope atoms present at time ti (atoms)
- Isotope atoms present at time td (atoms)
- Energy dependent, isotope photon source rates (ph/s)
- Energy dependent, FP and ⁶⁰Co photon fluxes (ph/cm²-s)
- Energy dependent, isotope photon dose rates (rem/h)

Isotope order: ⁹⁰Sr, ⁹⁰Y, ¹³³Cs, ¹³⁴Cs, ¹³⁷Cs, ^{137m}Ba, ¹⁴⁴Ce, ¹⁴⁴Pr, ¹⁰⁶Ru,
¹⁰⁶Rh, ⁹⁵Zr, ⁹⁵Nb and ⁶⁰Co

Energy order: 0.30, 0.63 and 1.10 MeV

```

program phdose
c hand calculation of photon dose rate
real*8 dk(13),yield(13),cap(13),r(13),k(13),n0(13),n(13),f2d(3)
real*8 gsour(5),gflux(5),gdose(5),ph(20),srate(20),drate(20)
real*8 ti,power,powden,td,kg,ppm,burnup
real*8 bu,time,dm,pflux,nflux,sums,sumd,self
common dk,cap,r,n0,powden,kg,ppm,bu,time
c parameter order in data arrays:
c      material      t(1/2)      ph/dis      groups*
c      #01 Sr-90      28.1y      #01-#02      #1-#2
c      #02 Y-90       2.67d      #03-#05      #1-#3
c      #03 Cs-133
c      #04 Cs-134      2.06y      #06-#07      #2-#3
c      #05 Cs-137      30.1y      #08-#09      #1-#2
c      #06 Ba-137m     2.55m      #10          #2
c      #07 Ce-144      284.d      #11          #1
c      #08 Pr-144      17.3m      #12-#14      #1-#3
c      #09 Ru-106      1.01y
c      #10 Rh-106      29.9s      #15-#17      #1-#3
c      #11 Zr-95       65.5d      #18          #2
c      #12 Nb-95       35.1d      #19          #2
c      #13 Co-60       5.26y      #20          #5
c      *gps, MeV: #1=0.30, #2=0.63, #3=1.10, #4=sum #1-#3, #5=Co-60 (1.10)
c decay constant, s-1
data dk/7.816d-10,3.005d-6,0.d0,1.066d-8,7.297d-10,4.530d-3,
x 2.821d-8,6.684d-4,2.174d-8,2.318d-2,1.225d-7,2.285d-7,4.176d-9/
c fission product yield
data yield/5.8d-2,0.d0,6.69d-2,0.d0,6.19d-2,0.d0,5.5d-2,0.d0,
x 4.01d-3,0.d0,6.5d-2,2*0.d0/
c capture cross section, cm**2
data cap/1.028d-24,2.740d-24,103.2d-24,107.1d-24,0.1310d-24,
x 0.d0,1.172d-24,0.d0,0.4550d-24,2*0.d0,3.563d-24,4.275d-24/
c photons/disintegration
data ph/4.69d-3,4.39d-6,4.94d-2,1.62d-2,2.48d-3,2.4d0,3.1d-2,
x 1.01d-2,4.24d-5,1.05d0,1.0d-1,7.53d-2,4.32d-2,8.67d-3,1.53d-1,
x 3.43d-1,5.2d-2,1.15d0,1.22d0,2.28d0/
c photon flux
data pflux/7.692d-6/
c fluence-to-dose factor, sv-cm**2
data f2d/1.56d-12,3.116d-12,4.928d-12/
c flux-to-dose conversion, rem/h
do 105 i=1,3
105 f2d(i)=f2d(i)*3.6d5
10 format(7d10.5)
11 format(1x,1p6d12.5)
c read input: irradi time (d); power (MW); pow den (MW/kgU-235);
c decay time (y); asby mass (kg); Co-59 (ppm); U-235 burnup (%)
c note: if ti=0, ti is calculated in the 'time' parameter below
100 read(5,10,end=999) ti,power,powden,td,kg,ppm,burnup
c burnup in percent
burnup=1.0d-2*burnup
c calculate irradiation time and average neutron flux
time=burnup/(1.25d-3*powden)
if(ti.eq.0.0d0) ti=time
dm=1.0d0 - 0.5d0*burnup
nflux=2.952d13*powden/dm
c write input data plus nflux
write(6,20)
20 format('0input: irradi time (d), power (MW), pow den (MW/kgU-235),
2decay time (y),'/5x,'asby mass (kg), Co-59 (ppm), U-235 burnup (%)
3, avg flux (n/cm**2-s)')
write(6,11) ti,power,powden,td,kg,ppm,burnup,nflux
c use ti as a flag for additional printout
nskip=0
if(ti.lt.0.0d0) then

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```

        ti=-ti
        nskip=1
    endif
c check if ti is consistent (<1%) with burnup and power density
    if(dabs(ti/time - 1.0d0).ge.1.0d-2) stop 101
c irradiation time (ti) in days and decay time (td) in years
    ti=ti*8.64d4
    td=td*3.1536d7
c initialize arrays
    do 101 i=1,13
101 r(i)=0.0d0
        do 103 i=1,5
            gsour(i)=0.0d0
            gflux(i)=0.0d0
103 gdose(i)=0.0d0
c in subroutines, calculate nflux in 10 steps
    bu=burnup/10.0d0
    time=ti/10.0d0
c #1: sr-90
    r(1)=3.121d16*yield(1)*power
    k(1)=dk(1) + cap(1)*nflux
c    n0(1)=r(1)/k(1)*(1.0d0 - dexp(-k(1)*ti))
    call sr90
    n(1)=n0(1)*dexp(-dk(1)*td)
    srate(1)=dk(1)*n(1)*ph(1)
    srate(2)=dk(1)*n(1)*ph(2)
    drate(1)=f2d(1)*srate(1)*pflux
    drate(2)=f2d(2)*srate(2)*pflux
    gsour(1)=gsour(1) + srate(1)
    gsour(2)=gsour(2) + srate(2)
    gflux(1)=gflux(1) + srate(1)*pflux
    gflux(2)=gflux(2) + srate(2)*pflux
    gdose(1)=gdose(1) + drate(1)
    gdose(2)=gdose(2) + drate(2)
c #2: y-90
    k(2)=dk(2) + cap(2)*nflux
    n0(2)=dk(1)/k(2)*n0(1)
    if(k(2)*ti.le.4.605170186d1) n0(2)=
x dk(1)*r(1)/k(1)*((1.0d0 - dexp(-k(2)*ti))/k(2) -
x (dexp(-k(1)*ti) - dexp(-k(2)*ti))/(k(2)-k(1)))
    n(2)=dk(1)/dk(2)*n(1)
    if(dk(2)*td.le.4.605170186d1) n(2)=
x dk(1)*n0(1)*(dexp(-dk(1)*td) - dexp(-dk(2)*td))/
x (dk(2) - dk(1)) + n0(2)*dexp(-dk(2)*td)
    srate(3)=dk(2)*n(2)*ph(3)
    srate(4)=dk(2)*n(2)*ph(4)
    srate(5)=dk(2)*n(2)*ph(5)
    drate(3)=f2d(1)*srate(3)*pflux
    drate(4)=f2d(2)*srate(4)*pflux
    drate(5)=f2d(3)*srate(5)*pflux
    gsour(1)=gsour(1) + srate(3)
    gsour(2)=gsour(2) + srate(4)
    gsour(3)=gsour(3) + srate(5)
    gflux(1)=gflux(1) + srate(3)*pflux
    gflux(2)=gflux(2) + srate(4)*pflux
    gflux(3)=gflux(3) + srate(5)*pflux
    gdose(1)=gdose(1) + drate(3)
    gdose(2)=gdose(2) + drate(4)
    gdose(3)=gdose(3) + drate(5)
c #3: cs-133
    r(3)=3.121d16*yield(3)*power
    k(3)=dk(3) + cap(3)*nflux
    n0(3)=r(3)/k(3)*(1.0d0 - dexp(-k(3)*ti))
    n(3)=n0(3)*dexp(-dk(3)*td)
c #4: cs-134

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k(4)=dk(4) + cap(4)*nflux
c n0(4)=r(3)*((1.0d0 - dexp(-k(4)*ti))/k(4) -
c x (dexp(-k(3)*ti) - dexp(-k(4)*ti))/(k(4)-k(3)))
call cs134
n(4)=n0(4)*dexp(-dk(4)*td)
srate(6)=dk(4)*n(4)*ph(6)
srate(7)=dk(4)*n(4)*ph(7)
drate(6)=f2d(2)*srate(6)*pflux
drate(7)=f2d(3)*srate(7)*pflux
gsour(2)=gsour(2) + srate(6)
gsour(3)=gsour(3) + srate(7)
gflux(2)=gflux(2) + srate(6)*pflux
gflux(3)=gflux(3) + srate(7)*pflux
gdose(2)=gdose(2) + drate(6)
gdose(3)=gdose(3) + drate(7)
c #5: cs-137
r(5)=3.121d16*yield(5)*power
k(5)=dk(5) + cap(5)*nflux
c n0(5)=r(5)/k(5)*(1.0d0 - dexp(-k(5)*ti))
call cs137
n(5)=n0(5)*dexp(-dk(5)*td)
srate(8)=dk(5)*n(5)*ph(8)
srate(9)=dk(5)*n(5)*ph(9)
drate(8)=f2d(1)*srate(8)*pflux
drate(9)=f2d(2)*srate(9)*pflux
gsour(1)=gsour(1) + srate(8)
gsour(2)=gsour(2) + srate(9)
gflux(1)=gflux(1) + srate(8)*pflux
gflux(2)=gflux(2) + srate(9)*pflux
gdose(1)=gdose(1) + drate(8)
gdose(2)=gdose(2) + drate(9)
c #6: ba-137m
k(6)=dk(6) + cap(6)*nflux
n0(6)=dk(5)/k(6)*n0(5)
if(k(6)*ti.le.4.605170186d1) n0(6)=
x dk(5)*r(5)/k(5)*((1.0d0 - dexp(-k(6)*ti))/k(6) -
x (dexp(-k(5)*ti) - dexp(-k(6)*ti))/(k(6)-k(5)))
n(6)=dk(5)/dk(6)*n(5)
if(dk(6)*td.le.4.605170186d1) n(6)=
x dk(5)*n0(5)*(dexp(-dk(5)*td) - dexp(-dk(6)*td))/
x (dk(6) - dk(5)) + n0(6)*dexp(-dk(6)*td)
srate(10)=dk(6)*n(6)*ph(10)
drate(10)=f2d(2)*srate(10)*pflux
gsour(2)=gsour(2) + srate(10)
gflux(2)=gflux(2) + srate(10)*pflux
gdose(2)=gdose(2) + drate(10)
c #7: ce-144
r(7)=3.121d16*yield(7)*power
k(7)=dk(7) + cap(7)*nflux
c n0(7)=r(7)/k(7)*(1.0d0 - dexp(-k(7)*ti))
call ce144
n(7)=n0(7)*dexp(-dk(7)*td)
srate(11)=dk(7)*n(7)*ph(11)
drate(11)=f2d(1)*srate(11)*pflux
gsour(1)=gsour(1) + srate(11)
gflux(1)=gflux(1) + srate(11)*pflux
gdose(1)=gdose(1) + drate(11)
c #8: pr-144
k(8)=dk(8) + cap(8)*nflux
n0(8)=dk(7)/k(8)*n0(7)
if(k(8)*ti.le.4.605170186d1) n0(8)=
x dk(7)*r(7)/k(7)*((1.0d0 - dexp(-k(8)*ti))/k(8) -
x (dexp(-k(7)*ti) - dexp(-k(8)*ti))/(k(8)-k(7)))
n(8)=dk(7)/dk(8)*n(7)
if(dk(8)*td.le.4.605170186d1) n(8)=

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x dk(7)*n0(7)*(dexp(-dk(7)*td) - dexp(-dk(8)*td))/
x (dk(8) - dk(7)) + n0(8)*dexp(-dk(8)*td)
  srate(12)=dk(8)*n(8)*ph(12)
  srate(13)=dk(8)*n(8)*ph(13)
  srate(14)=dk(8)*n(8)*ph(14)
  drate(12)=f2d(1)*srate(12)*pflux
  drate(13)=f2d(2)*srate(13)*pflux
  drate(14)=f2d(3)*srate(14)*pflux
  gsour(1)=gsour(1) + srate(12)
  gsour(2)=gsour(2) + srate(13)
  gsour(3)=gsour(3) + srate(14)
  gflux(1)=gflux(1) + srate(12)*pflux
  gflux(2)=gflux(2) + srate(13)*pflux
  gflux(3)=gflux(3) + srate(14)*pflux
  gdose(1)=gdose(1) + drate(12)
  gdose(2)=gdose(2) + drate(13)
  gdose(3)=gdose(3) + drate(14)
c #9: ru-106
  r(9)=3.121d16*yield(9)*power
  k(9)=dk(9) + cap(9)*nflux
c
  n0(9)=r(9)/k(9)*(1.0d0 - dexp(-k(9)*ti))
  call rul06
  n(9)=n0(9)*dexp(-dk(9)*td)
c #10: rh-106
  k(10)=dk(10) + cap(10)*nflux
  n0(10)=dk(9)/k(10)*n0(9)
  if(k(10)*ti.le.4.605170186d1) n0(10)=
x dk(9)*r(9)/k(9)*((1.0d0 - dexp(-k(10)*ti))/k(10) -
x (dexp(-k(9)*ti) - dexp(-k(10)*ti))/(k(10)-k(9)))
  n(10)=dk(9)/dk(10)*n(9)
  if(dk(10)*td.le.4.605170186d1) n(10)=
x dk(9)*n0(9)*(dexp(-dk(9)*td) - dexp(-dk(10)*td))/
x (dk(10) - dk(9)) + n0(10)*dexp(-dk(10)*td)
  srate(15)=dk(10)*n(10)*ph(15)
  srate(16)=dk(10)*n(10)*ph(16)
  srate(17)=dk(10)*n(10)*ph(17)
  drate(15)=f2d(1)*srate(15)*pflux
  drate(16)=f2d(2)*srate(16)*pflux
  drate(17)=f2d(3)*srate(17)*pflux
  gsour(1)=gsour(1) + srate(15)
  gsour(2)=gsour(2) + srate(16)
  gsour(3)=gsour(3) + srate(17)
  gflux(1)=gflux(1) + srate(15)*pflux
  gflux(2)=gflux(2) + srate(16)*pflux
  gflux(3)=gflux(3) + srate(17)*pflux
  gdose(1)=gdose(1) + drate(15)
  gdose(2)=gdose(2) + drate(16)
  gdose(3)=gdose(3) + drate(17)
c #11: zr-95
  r(11)=3.121d16*yield(11)*power
  k(11)=dk(11) + cap(11)*nflux
c
  n0(11)=r(11)/k(11)*(1.0d0 - dexp(-k(11)*ti))
  call zr95
  n(11)=n0(11)*dexp(-dk(11)*td)
  srate(18)=dk(11)*n(11)*ph(18)
  drate(18)=f2d(2)*srate(18)*pflux
  gsour(2)=gsour(2) + srate(18)
  gflux(2)=gflux(2) + srate(18)*pflux
  gdose(2)=gdose(2) + drate(18)
c #12: nb-95
  k(12)=dk(12) + cap(12)*nflux
  n0(12)=dk(11)/k(12)*n0(11)
  if(k(12)*ti.le.4.605170186d1) n0(12)=
x dk(11)*r(11)/k(11)*((1.0d0 - dexp(-k(12)*ti))/k(12) -
x (dexp(-k(11)*ti) - dexp(-k(12)*ti))/(k(12)-k(11)))

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```

n(12)=dk(11)/dk(12)*n(11)
if(dk(12)*td.le.4.605170186d1) n(12)=
x dk(11)*n0(11)*(dexp(-dk(11)*td) - dexp(-dk(12)*td))/
x (dk(12) - dk(11)) + n0(12)*dexp(-dk(12)*td)
srate(19)=dk(12)*n(12)*ph(19)
drate(19)=f2d(2)*srate(19)*pflux
gsour(2)=gsour(2) + srate(19)
gflux(2)=gflux(2) + srate(19)*pflux
gdose(2)=gdose(2) + drate(19)
c #13: co-60
r(13)=1.197d10*kg*ppm*powden/dm
k(13)=dk(13) + cap(13)*nflux
c n0(13)=r(13)/k(13)*(1.0d0 - dexp(-k(13)*ti))
call co60
n(13)=n0(13)*dexp(-dk(13)*td)
srate(20)=dk(13)*n(13)*ph(20)
drate(20)=f2d(3)*srate(20)*pflux
gsour(5)=srate(20)
gflux(5)=srate(20)*pflux
gdose(5)=drate(20)
c sum fission product photon source and dose rates
sums=0.0d0
sumd=0.0d0
do 102 i=1,19
sums=sums + srate(i)
102 sumd=sumd + drate(i)
burnup=power/powden*1.0d3*burnup
sums=sums/burnup
sumd=sumd/burnup
c mass of burned U-235 necessary for 100 rem/h dose rate
self=1.0d2/sumd
c write output: U-235 burned, g;
c FP source, ph/s per gU-235 burned; Co-60 source, ph/s;
c self-protected U-235 mass burned, g
c FP dose, rem/h per gU-235 burned; Co-60 dose, rem/h
write(6,21)
21 format(' output: U-235 burned (g), FP source (ph/s per g), Co-60 s
2source (ph/s), '/5x,'self-pro U-235 mass (g), FP dose (rem/h per g),
3 Co-60 dose (rem/h)')
write(6,11) burnup,sums,srate(20),self,sumd,drate(20)
if(nskip.eq.0) go to 100
c write optional data
write(6,22)
22 format(' optional: form & loss rates (atoms/s), initial & current
2no. (atoms), '/5x'source (ph/s), flux (ph/cm**2-s), dose (rem/h)')
c formation rates, atoms/s
write(6,11) r
c loss rates, atoms/s
write(6,11) k
c atoms initially present, atoms (time=ti)
write(6,11) n0
c atoms currently present, atoms (time=td)
write(6,11) n
c photon source rates, ph/s
write(6,11) srate
c photon source rates, ph/s; sum fission product photon source
do 106 i=1,3
106 gsour(4)=gsour(4) + gsour(i)
write(6,11) gsour
c photon fluxes, ph/cm**2-s; sum fission product photon fluxes
do 107 i=1,3
107 gflux(4)=gflux(4) + gflux(i)
write(6,11) gflux
c photon dose rates, rem/h
write(6,11) drate

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c photon dose rates, rem/h; sum fission product photon dose
  do 108 i=1,3
108 gdose(4)=gdose(4) + gdose(i)
  write(6,11) gdose
  go to 100
999 stop
  end
c
  subroutine sr90
  real*8 dk(13),cap(13),r(13),n0(13)
  real*8 powden,kg,ppm
  real*8 bu,time,dm,nflux,t(10),c(10),n(10),k
  common dk,cap,r,n0,powden,kg,ppm,bu,time
  do 101 i=1,10
  t(i)=dfloat(i)*time
  dm=1.0d0-bu*(dfloat(i)-0.5d0)
  nflux=2.952d13*powden/dm
  k=dk(1) + cap(1)*nflux
  if(i.eq.1) c(1)=-r(1)/k
  if(i.ne.1) c(i)=n(i-1)*dexp(k*t(i-1)) - r(1)*dexp(k*t(i-1))/k
101 n(i)=r(1)/k + c(i)*dexp(-k*t(i))
c
  write(6,11) n
  11 format(1x,1p6d12.5)
  n0(1)=n(10)
  return
  end
c
  subroutine cs134
  real*8 dk(13),cap(13),r(13),n0(13)
  real*8 powden,kg,ppm
  real*8 bu,time,dm,nflux,t(10),c(10),n(10),k1,k2,del
  common dk,cap,r,n0,powden,kg,ppm,bu,time
  do 101 i=1,10
  t(i)=dfloat(i)*time
  dm=1.0d0-bu*(dfloat(i)-0.5d0)
  nflux=2.952d13*powden/dm
  k1=dk(3) + cap(3)*nflux
  k2=dk(4) + cap(4)*nflux
  del=k2-k1
  if(i.eq.1) c(1)=r(3)*(-1.0d0/k2 + 1.0d0/del)
  if(i.ne.1) c(i)=n(i-1)*dexp(k2*t(i-1)) -
x r(3)*(dexp(k2*t(i-1))/k2 - dexp(del*t(i-1))/del)
101 n(i)=r(3)*(1.0d0/k2 - dexp(-k1*t(i))/del) + c(i)*dexp(-k2*t(i))
c
  write(6,11) n
  11 format(1x,1p6d12.5)
  n0(4)=n(10)
  return
  end
c
  subroutine cs137
  real*8 dk(13),cap(13),r(13),n0(13)
  real*8 powden,kg,ppm
  real*8 bu,time,dm,nflux,t(10),c(10),n(10),k
  common dk,cap,r,n0,powden,kg,ppm,bu,time
  do 101 i=1,10
  t(i)=dfloat(i)*time
  dm=1.0d0-bu*(dfloat(i)-0.5d0)
  nflux=2.952d13*powden/dm
  k=dk(5) + cap(5)*nflux
  if(i.eq.1) c(1)=-r(5)/k
  if(i.ne.1) c(i)=n(i-1)*dexp(k*t(i-1)) - r(5)*dexp(k*t(i-1))/k
101 n(i)=r(5)/k + c(i)*dexp(-k*t(i))
c
  write(6,11) n
  11 format(1x,1p6d12.5)
  n0(5)=n(10)

```

```

return
end
c
subroutine cel44
real*8 dk(13),cap(13),r(13),n0(13)
real*8 powden,kg,ppm
real*8 bu,time,dm,nflux,t(10),c(10),n(10),k
common dk,cap,r,n0,powden,kg,ppm,bu,time
do 101 i=1,10
t(i)=dfloat(i)*time
dm=1.0d0-bu*(dfloat(i)-0.5d0)
nflux=2.952d13*powden/dm
k=dk(7) + cap(7)*nflux
if(i.eq.1) c(1)=-r(7)/k
if(i.ne.1) c(i)=n(i-1)*dexp(k*t(i-1)) - r(7)*dexp(k*t(i-1))/k
101 n(i)=r(7)/k + c(i)*dexp(-k*t(i))
c
write(6,11) n
11 format(1x,1p6d12.5)
n0(7)=n(10)
return
end
c
subroutine ru106
real*8 dk(13),cap(13),r(13),n0(13)
real*8 powden,kg,ppm
real*8 bu,time,dm,nflux,t(10),c(10),n(10),k
common dk,cap,r,n0,powden,kg,ppm,bu,time
do 101 i=1,10
t(i)=dfloat(i)*time
dm=1.0d0-bu*(dfloat(i)-0.5d0)
nflux=2.952d13*powden/dm
k=dk(9) + cap(9)*nflux
if(i.eq.1) c(1)=-r(9)/k
if(i.ne.1) c(i)=n(i-1)*dexp(k*t(i-1)) - r(9)*dexp(k*t(i-1))/k
101 n(i)=r(9)/k + c(i)*dexp(-k*t(i))
c
write(6,11) n
11 format(1x,1p6d12.5)
n0(9)=n(10)
return
end
c
subroutine zr95
real*8 dk(13),cap(13),r(13),n0(13)
real*8 powden,kg,ppm
real*8 bu,time,dm,nflux,t(10),c(10),n(10),k
common dk,cap,r,n0,powden,kg,ppm,bu,time
do 101 i=1,10
t(i)=dfloat(i)*time
dm=1.0d0-bu*(dfloat(i)-0.5d0)
nflux=2.952d13*powden/dm
k=dk(11) + cap(11)*nflux
if(i.eq.1) c(1)=-r(11)/k
if(i.ne.1) c(i)=n(i-1)*dexp(k*t(i-1)) - r(11)*dexp(k*t(i-1))/k
101 n(i)=r(11)/k + c(i)*dexp(-k*t(i))
c
write(6,11) n
11 format(1x,1p6d12.5)
n0(11)=n(10)
return
end
c
subroutine co60
real*8 dk(13),cap(13),r(13),n0(13)
real*8 powden,kg,ppm
real*8 bu,time,dm,nflux,t(10),c(10),n(10),k1,k2,r59
common dk,cap,r,n0,powden,kg,ppm,bu,time

```



```
do 101 i=1,10
t(i)=dfloat(i)*time
dm=1.0d0-bu*(dfloat(i)-0.5d0)
nflux=2.952d13*powden/dm
k1=      39.67d-24*nflux
k2=dk(13) + cap(13)*nflux
r59=1.022d19*kg*ppm*k1
if(i.eq.1) c(1)=-r59/k2
if(i.ne.1) c(i)=n(i-1)*dexp(k2*t(i-1)) - r59*dexp(k2*t(i-1))/k2
101 n(i)=r59/k2 + c(i)*dexp(-k2*t(i))
c write(6,11) n
11 format(1x,1p6d12.5)
n0(13)=n(10)
return
end
```