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# Calculated, Two-Dimensional Dose Rates from a PWR Fuel Assembly

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# CALCULATED, TWO-DIMENSIONAL DOSE RATES FROM A PWR FUEL ASSEMBLY

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Contents

Page

Abstract . . . . . 1

1. Introduction . . . . . 1

1.1 Description of the Fuel Assembly . . . . . 2

1.2 Limitations . . . . . 2

2. Dose Rates from a PWR Fuel Assembly . . . . . 7

2.1 Calculational Model and Methods . . . . . 8

2.2 Fission Product Dose Rate from a PWR Fuel Assembly . . . . . 14

2.3 Dose Rate from a PWR Fuel Assembly Containing a $^{60}Co$ Spike . . . . . 18

2.4 Discussion . . . . . 22

4. References . . . . . 23

## Tables

Page

Table 1. Physical characteristics of a PWR fuel assembly . . . . . 3

Table 2. Material compositions in dose rate calculations for a PWR fuel assembly . . . . . 9

Table 3. Fission product gamma-ray intensity from a 10-year-old PWR fuel assembly . . . . . 13

## Figures

Fig. 1. Schematic of a PWR fuel element. (Reproduced from ref. 2). . . . . 4

Fig. 2. Schematic of a PWR fuel assembly. (Reproduced from ref. 2). . . . . 5

Fig. 3. Photograph of a PWR fuel assembly. (Supplied by courtesy of the Westinghouse Electric Corporation) . . . . . 6

Fig. 4. PWR fuel assembly model used in dose rate calculations . . . 10

Fig. 5. Surfaces where PWR fuel assembly dose rates were calculated . . . . . 12

Fig. 6. Source shape and geometry parameter for a fission product source . . . . . 15

Fig. 7. Decay factor for fission products in a PWR fuel assembly . . 16

Fig. 8. Fission product dose rate from a 10-year-old PWR fuel assembly . . . . . 17

Fig. 9. Source shape and geometry parameter for a uniformly distributed spike . . . . . 19

Fig. 10. Decay of $^{60}$Co spike after fabrication . . . . . 20

Fig. 11. Dose rate from 30 ppm $^{60}$Co in a PWR fuel assembly . . . . . 21

# CALCULATED, TWO-DIMENSIONAL DOSE RATES FROM A PWR FUEL ASSEMBLY

A. G. Croff, O. W. Hermann,* and C. W. Alexander

ABSTRACT

This report first describes the physical characteristics of a PWR fuel assembly. A model of the assembly is then defined, and a two-dimensional gamma-ray shielding code is applied to the model to calculate the dose rate from a bare fuel assembly. The results of the calculations are the dose rates from the assembly at points on the surface of the assembly, on spatial surfaces 1 m from a side and the end piece of the assembly, and on spatial surfaces 2 m from a side and the end piece of the assembly. Dose rates are calculated for as-generated mixed fission products and for a uniformly distributed $^{60}Co$ spike. Factors are given to account for the effect of decay on the dose rates.

## 1. INTRODUCTION

The purpose of this document is to present the dose rate characteristics of spent PWR fuel. The characteristics of a fuel assembly that are considered here are:

1. the dose rate from the fission product gamma rays in two dimensions as functions of decay time and distance, and
2. the dose rate from a uniformly distributed spike in two dimensions as functions of decay time and distance.

The two-dimensional calculations are needed to account for the complex variation in dose rate resulting from the finite length of the fuel assembly, geometric effects, and the nonuniform distribution of the fission products in the fuel.

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## 1.1 Description of the Fuel Assembly

The physical characteristics of a typical PWR fuel assembly are summarized in Table 1. The most fundamental part of the fuel assembly is a uranium dioxide ($UO_2$) fuel pellet, which is a right circular cylinder about 0.8 cm in diameter and 1.35 cm long. Pellets are stacked in a Zircaloy tube to a height of 360 to 380 cm, and the ends of the tube are sealed by welded Zircaloy plugs. The Zircaloy tube containing the fuel pellets is called a fuel element, and is shown schematically in Fig. 1. The fuel elements are then arranged in a square array that is held together by Inconel grid spacers along the length of the fuel elements and with two stainless steel end pieces. This square fuel element array, called a fuel assembly, is shown schematically in Fig. 2. The PWR fuel assembly considered in this report contains a 17 x 17 array of fuel elements; however, only 264 of the maximum possible 289 fuel element locations in the assembly are occupied by fuel elements. The remaining locations are occupied with control-rod guide thimbles and with instrument sheaths. The guide tubes also serve to connect the two stainless steel end pieces. A full-length photograph of a PWR fuel assembly is shown in Fig. 3.

## 1.2 Limitations

The fuel assembly characteristics summarized in this report were developed using a specific set of assumptions concerning the fuel composition and irradiation conditions. As a result, the characteristics are not applicable in detail to all situations. The following items summarize the major limitations of the results presented here:

1. The results were specifically calculated for a $^{235}U$-enriched PWR fuel assembly. These results should also be reasonably accurate for $^{233}U$-enriched and plutonium-enriched PWR fuel assemblies at short decay times and less than actual dose rates for longer decay times.

Table 1. Physical characteristics of a PWR fuel assembly[1]

| | |
|---|---|
| Overall assembly length, m | 4.059 |
| Cross section, cm | 21.4 x 21.4 |
| Fuel element length, m | 3.851 |
| Active fuel height, m | 3.658 |
| Fuel element OD, cm | 0.950 |
| Fuel element array | 17 x 17 |
| Fuel elements per assembly | 264 |
| Total weight of assembly, kg | 657.9 |
| Nominal volume of assembly, $m^3$ | 0.186[a] |
| Composition of assembly, kg | |
| Uranium | 461.4 |
| $UO_2$ | 523.4 |
| Zircaloy | 108.4[b] |
| Hardware | 26.1[c] |
| Total metal | 134.5 |

[a]Based on overall outside dimension.

[b]Includes Zircaloy control-rod guide thimbles.

[c]Includes stainless steel nozzles and Inconel-718 grids.

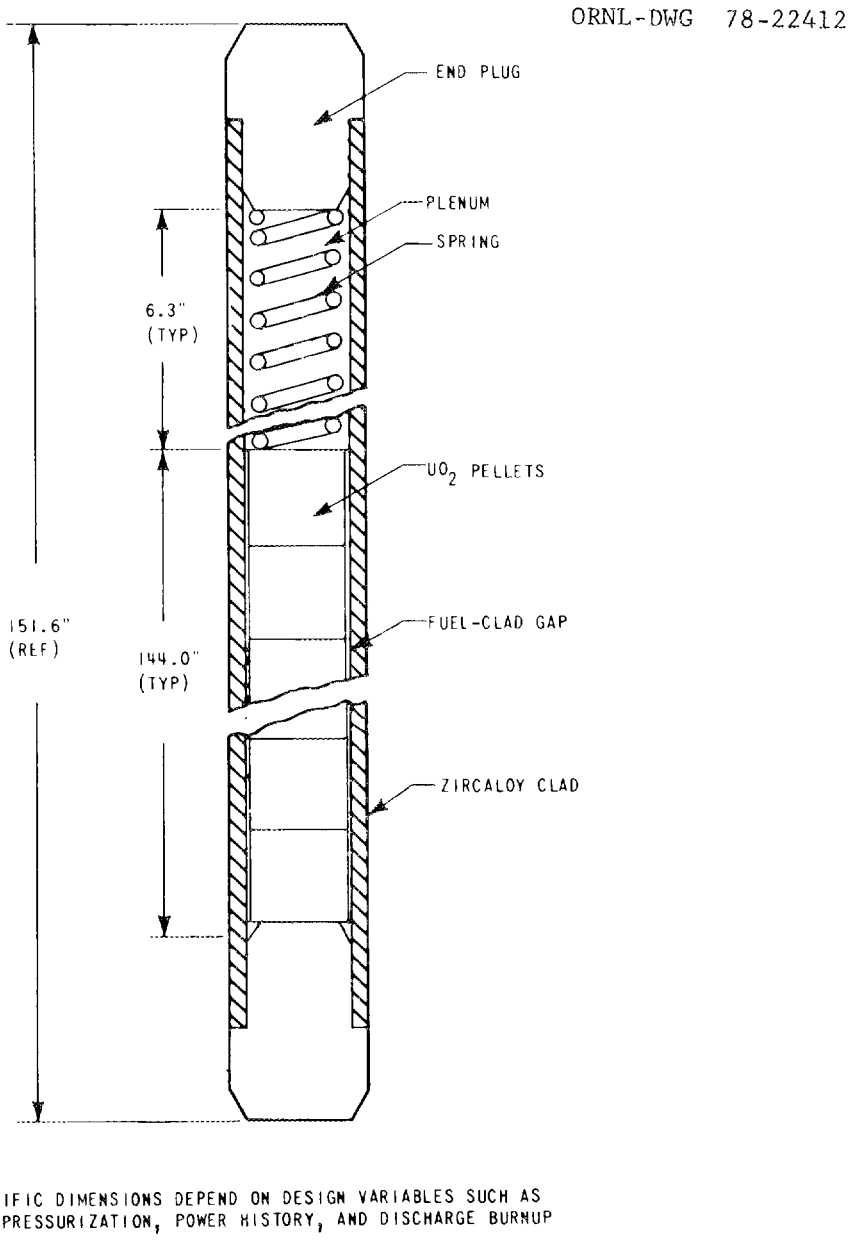


Fig. 1. Schematic of a PWR fuel element. (Reproduced from ref. 2)

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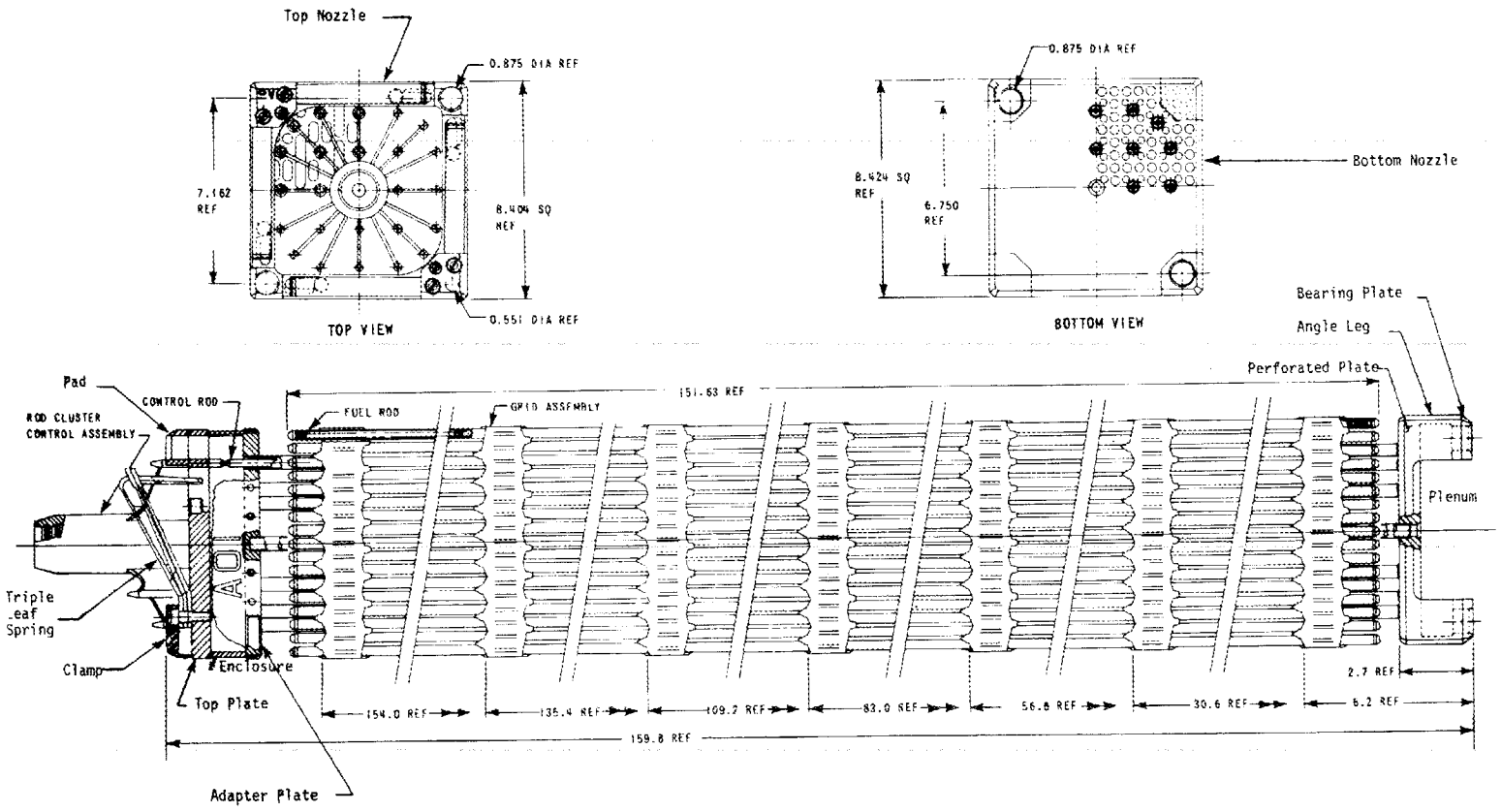


Fig. 2. Schematic of a PWR fuel assembly. (Reproduced from ref. 2)

Fig. 3. Photograph of a PWR fuel assembly. (Supplied by courtesy of the Westinghouse Electric Corporation)

2. The results are based on a fuel assembly burnup of 33,000 MWd(t) per metric ton of initial uranium as metal. The dose rate from the assembly is roughly proportional to burnup at decay times longer than a few years and is directly dependent on the fuel specific power (i.e., kW/kg fuel) at discharge. At intermediate times, the dose rate depends on both burnup and specific power.

3. These results are not applicable to BWR fuel assemblies because of their case-specific, nonsymmetrical burnup distribution and much smaller amount of uranium per assembly as compared with the PWR.

4. Neutrons are not considered in the dose calculations. The dose rate from neutrons emitted by spent $^{235}$U-enriched and $^{233}$U-enriched PWR fuel is small compared with that from the gamma rays. The dose rate from neutrons emitted by spent plutonium-enriched fuel is significant at longer decay times, as noted in item 1 above.

## 2. DOSE RATES FROM A PWR FUEL ASSEMBLY

This section describes shielding calculations for a PWR fuel assembly which result in the prediction of the dose rate as a function of position relative to the assembly. Most previous dose-rate calculations have treated the fuel assembly as an infinitely long source of radiation emitting a specified number of gamma rays per unit length. The dose rate was then calculated as a function of one dimension, the distance from the line source. However, this type of analysis is inaccurate because it ignores the decrease in dose that results from the finite length of the assembly and the lower burnup at the ends of the fuel assembly. Furthermore, a one-dimensional calculation cannot predict the dose rates near the corners and on the end of the fuel assembly. Thus, a two-dimensional calculation is required to realistically determine the dose rate at multiple points surrounding the spent fuel assembly.

Section 2.1 describes the two-dimensional PWR fuel assembly model and the calculational tools used in determining the dose rates. Section 2.2 presents the calculated dose rates for the case where the radiation source is nonuniformly distributed fission products. Section 2.3

presents the calculated dose rates for the case where the radiation source is a uniformly distributed spike ($^{60}Co$).

## 2.1 Calculational Model and Methods

The PWR fuel assembly described in Sect. 1.1 was much too detailed for use in the two-dimensional shielding calculations discussed herein. Thus, it was necessary to simplify the assembly design to make the shielding calculation feasible. The first step was to make the fuel assembly symmetrical in the third (circumferential) dimension by approximating it as a right circular cylinder with about the same cross-sectional area as the square assembly. Next, the normally heterogeneous nature of the fuel assembly (i.e., air, cladding, and fuel) was eliminated by uniformly mixing all of the fuel assembly constituents within the assembly volume. Since no Zircaloy cross-section data were readily available, stainless steel was substituted for the Zircaloy in the fuel assembly on a gram-for-gram basis. Finally, minor approximations concerning the sizes of the fuel assembly end pieces and gas plenum made the fuel assembly symmetrical, so that only one-fourth of it had to be considered in the calculations. The result of these three approximations was a much simpler calculational model of the fuel assembly that very closely approximated the actual dose characteristics of the fuel assembly.

The atom densities of the materials considered in the two-dimensional shielding calculation are given in Table 2. Only those materials that would have a significant impact on the shielding properties of the fuel assembly were included. A schematic diagram of the geometric model used in the calculation is given in Fig. 4. The cylindrical fuel assembly was assumed to have 360-cm-long fuel elements and identical end pieces 12 cm in length. The diameter of the model assembly was taken to be 24 cm. However, as noted above, symmetry considerations resulted in the calculation having to account for only one-quarter of the fuel assembly. The portion of the assembly considered, as indicated by the dotted lines in Fig. 4, consisted of 180 cm of fuel element and 12 cm of end piece with a diameter of 12 cm. The dose rates were calculated on the side and the end surfaces of the fuel assembly, on spatial surfaces 1 m from

Table 2. Material compositions in dose rate calculations for a PWR fuel assembly

| Element or isotope | Concentration (atoms $barn^{-1}$ $cm^{-1}$) |
|---|---|
| | Fuel assembly |
| $^{235}U$ | $6.8 \times 10^{-5}$ |
| $^{238}U$ | $6.7 \times 10^{-3}$ |
| O | $1.6 \times 10^{-2}$ |
| Cr[a] | $1.2 \times 10^{-3}$ |
| Fe[a] | $4.5 \times 10^{-3}$ |
| Ni[a] | $6.2 \times 10^{-4}$ |
| | Air |
| O | $1.1 \times 10^{-5}$ |
| N | $3.4 \times 10^{-5}$ |

[a]Stainless steel (Cr, Fe, Ni) substituted for Zircaloy on a gram-for-gram basis.

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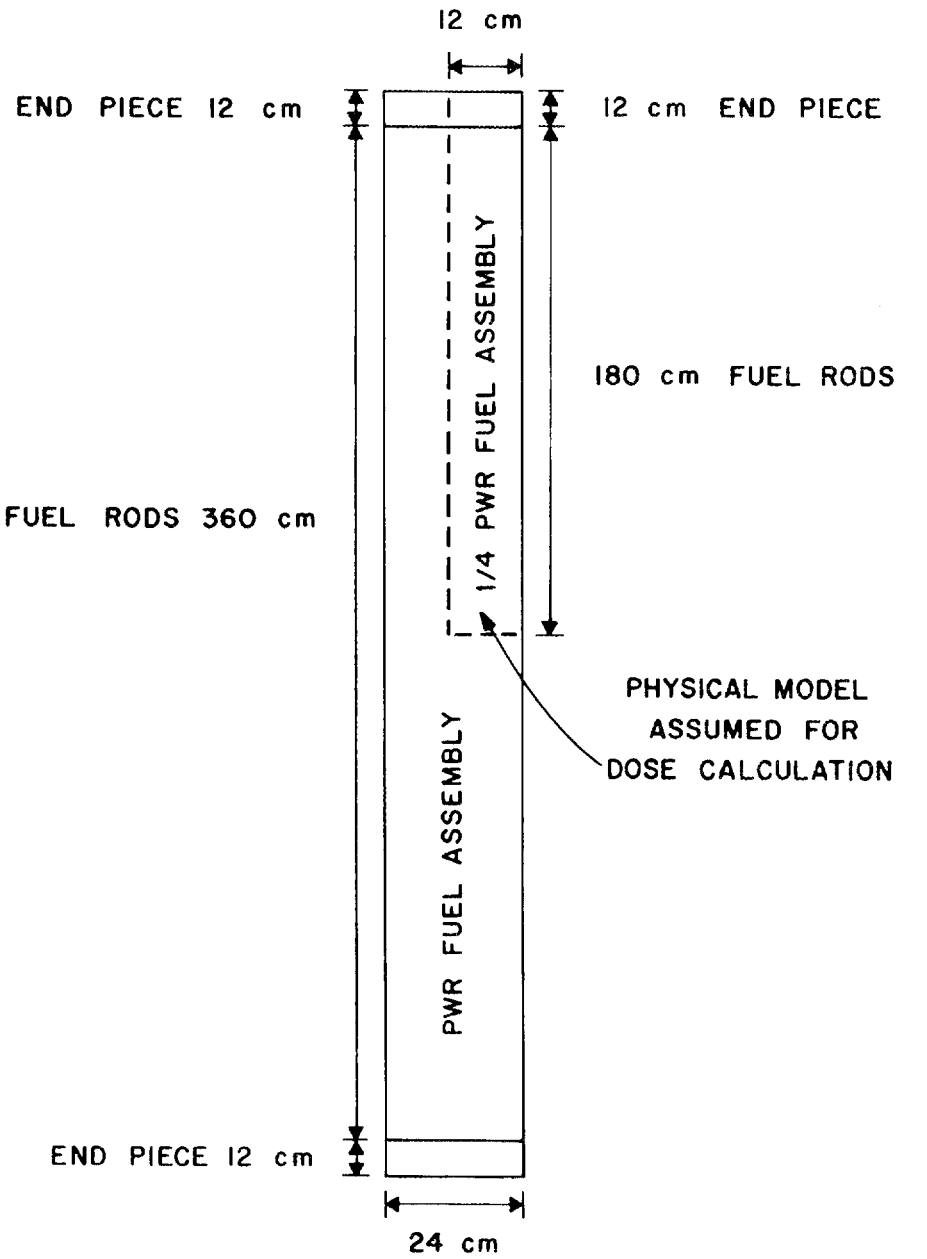


Fig. 4. PWR fuel assembly model used in dose rate calculations.

the side and the end of the fuel assembly, and on spatial surfaces 2 m from the side and the end of the fuel assembly. These six surfaces are depicted schematically in Fig. 5.

The calculations were performed using the DOT-IV[3] and FALSTF[4] computer codes. DOT is a two-dimensional discrete-ordinates computer code that was used to calculate the transport of the fuel assembly radiation within the assembly itself. An RZ geometry was used since the assembly was being approximated as a cylinder. The code uses the multigroup gamma-ray spectrum given in Table 3 as a cosine-distributed source and the fuel assembly model given in Fig. 5 to calculate the gamma-ray flux in many volume intervals within and near the surface of the fuel assembly. The gamma-ray fluxes were converted to dose rates using Henderson's[5] conversion factors, which are a function of the gamma-ray energy. The FALSTF code was then used to calculate the dose rates at points away from the fuel assembly where the use of DOT would require a prohibitively large amount of computer time. FALSTF uses the effective gamma-ray source calculated by DOT and the fuel assembly geometry to calculate the gamma-ray flux and the dose rate at many points within a few meters of the assembly. As with the DOT results, the gamma-ray intensities were converted to dose rates using Henderson's[5] conversion factors.

The absolute dose rates after a decay time of 10 years were calculated with DOT and FALSTF and were based on the fission product gamma-ray spectrum given in Table 3, which was calculated with a modified version of ORIGEN.[6,7] This version of ORIGEN uses fission product decay data taken from the Evaluated Nuclear Data File, Part B, Version IV (ENDF/B-IV).[8]

The calculations required to obtain the variation in gamma-ray dose rate as a function of time were performed using the one-dimensional shielding code ANISN.[9] The absolute dose rates calculated by ANISN were converted to relative values by normalizing them to the calculated dose rate after a decay time of 10 years.

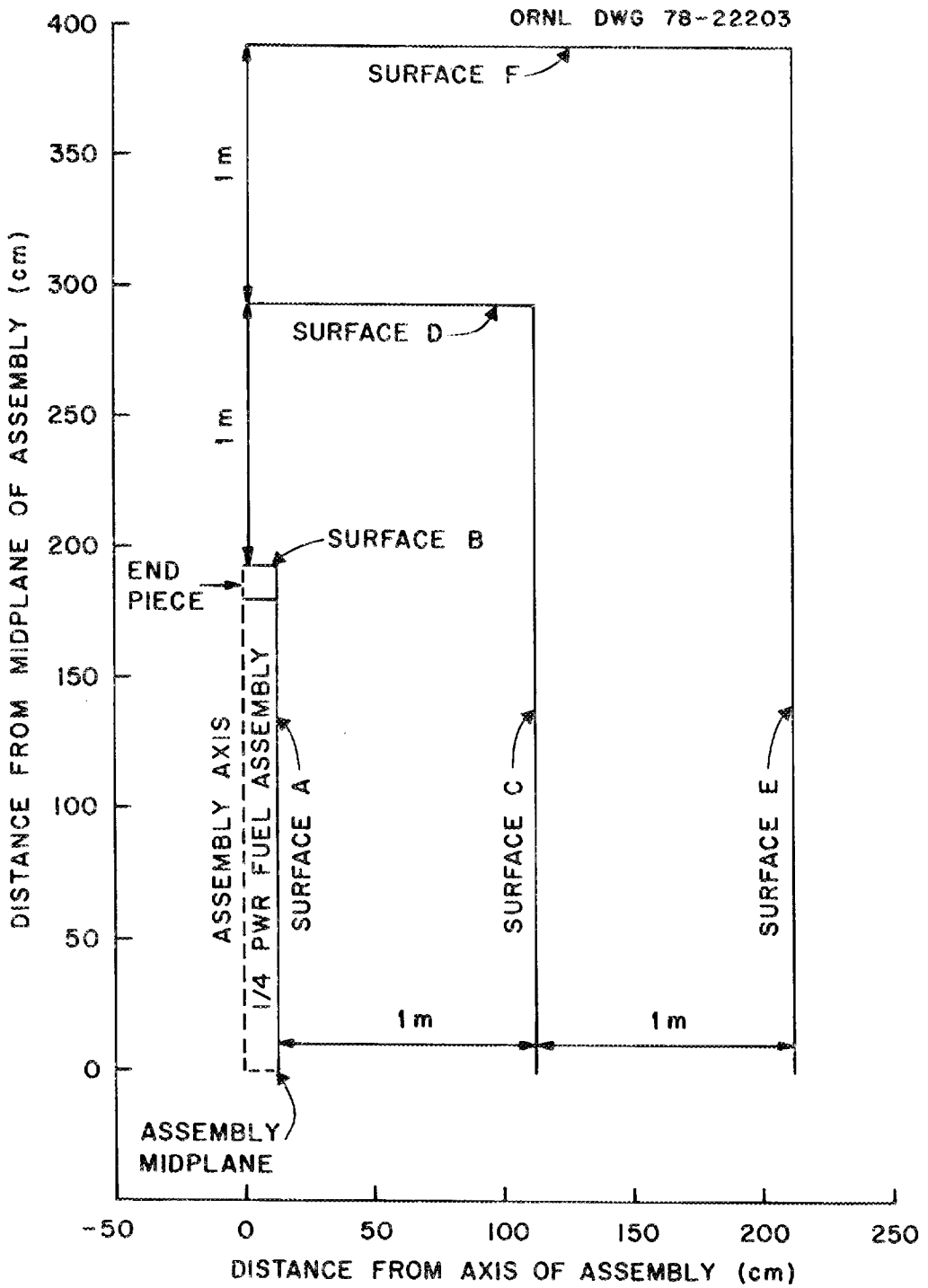


Fig. 5. Surfaces where PWR fuel assembly dose rates were calculated.

Table 3. Fission product gamma-ray intensity from a 10-year-old PWR fuel assembly

| Energy interval boundaries (MeV) | | Gamma-ray intensity (photons $sec^{-1}$ $assembly^{-1}$) |
|---|---|---|
| Upper | Lower | |
| 4.0 | 3.0 | 2.88+08 |
| 3.0 | 2.5 | 2.22+09 |
| 2.5 | 2.0 | 3.73+10 |
| 2.0 | 1.5 | 1.58+12 |
| 1.5 | 1.0 | 5.11+13 |
| 1.0 | 0.70 | 1.30+14 |
| 0.70 | 0.45 | 1.55+15 |
| 0.45 | 0.30 | 7.36+13 |
| 0.30 | 0 | 8.20+14 |

## 2.2 Fission Product Dose Rate from a PWR Fuel Assembly

The dose rate (in rem/hr) from a PWR fuel assembly has been calculated using the methodology and the model described in Sect. 2.1. The results of the calculations can be summarized by the following equation:

$$\text{Dose rate (rem/hr)} = 23{,}400 \times SG \times DF, \qquad (1)$$

where

23,400 = dose rate on the midplane surface of a 10-year-old PWR fuel assembly, rem/hr;

SG = source shape and geometry parameter from Fig. 6 (this parameter is keyed to the surfaces in Fig. 5);

DF = decay factor from Fig. 7 accounting for fission product decay.

As an example, let us assume that the dose at a point 1 m perpendicular to the end of the fuel elements in the assembly at a decay time of 10 years is of interest. From Fig. 5, the point at which the dose rate is desired is on Surface C, 180 cm from the assembly midplane. The source and shape parameter, SG, is about 0.017 and is obtained from Fig. 6 by using the curve corresponding to Surface C at a distance of 180 cm. The decay factor, DF, of 1.0 for a time interval of 10 years is obtained from Fig. 7. The dose rate is then

$$23{,}400 \times 0.017 \times 1.0 = 398 \text{ rem/hr}.$$

The SG parameter shown in Fig. 6 accounts for both the position of the dose recipient with respect to the fuel assembly and the axial variation in burnup within the fuel assembly. The DF given in Fig. 7 accounts for the decay of the fission products and the change in the gamma-ray spectrum with time.

The above procedure for calculating the fission product dose rate, while reasonably accurate and versatile, is somewhat cumbersome. Therefore, the fission product dose rate from the PWR fuel assembly is also given explicitly in Fig. 8 for a decay time of 10 years and discrete points on the surfaces. The values given in Fig. 8 can be adjusted to other decay times by multiplying them by the decay factor given in Fig. 7.

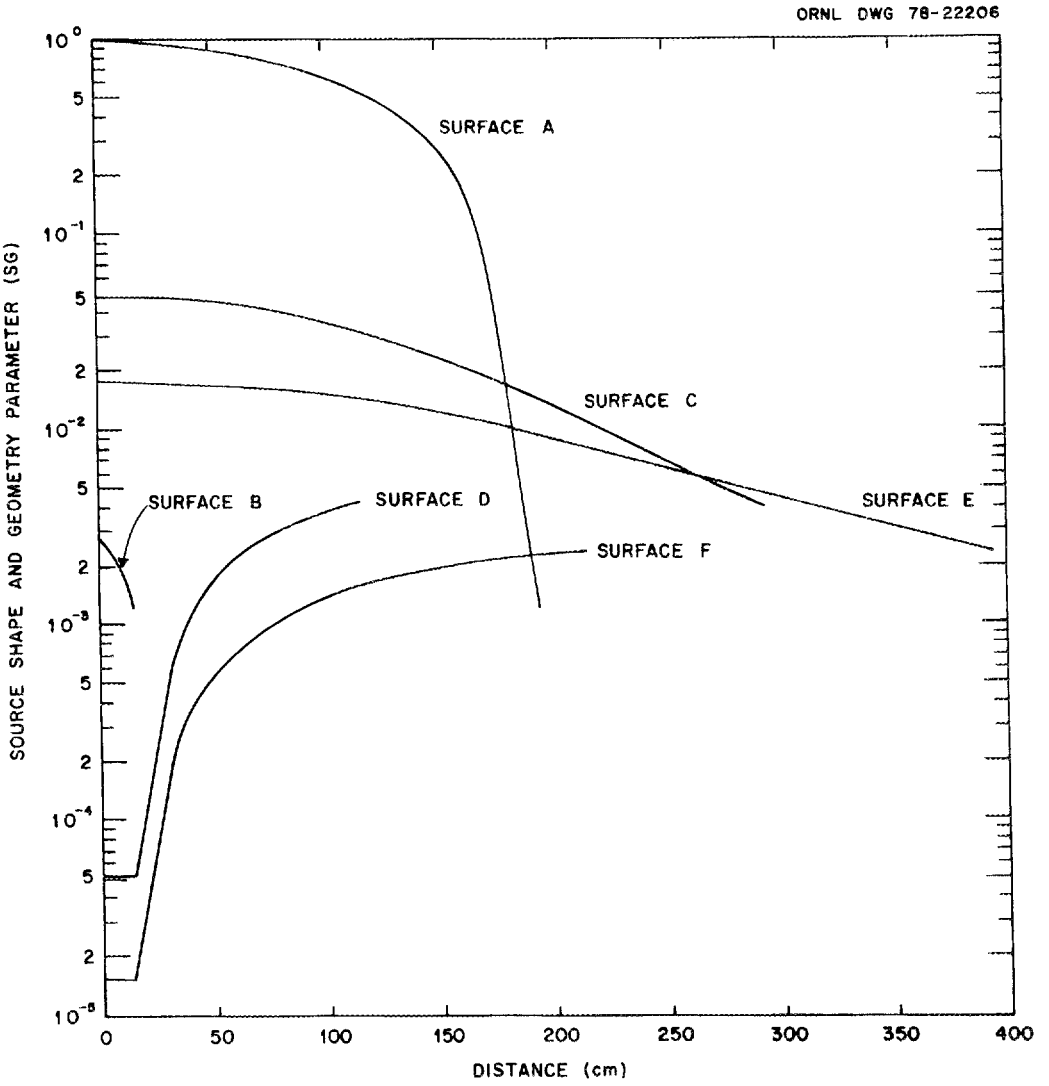


Fig. 6. Source shape and geometry parameter for a fission product source.

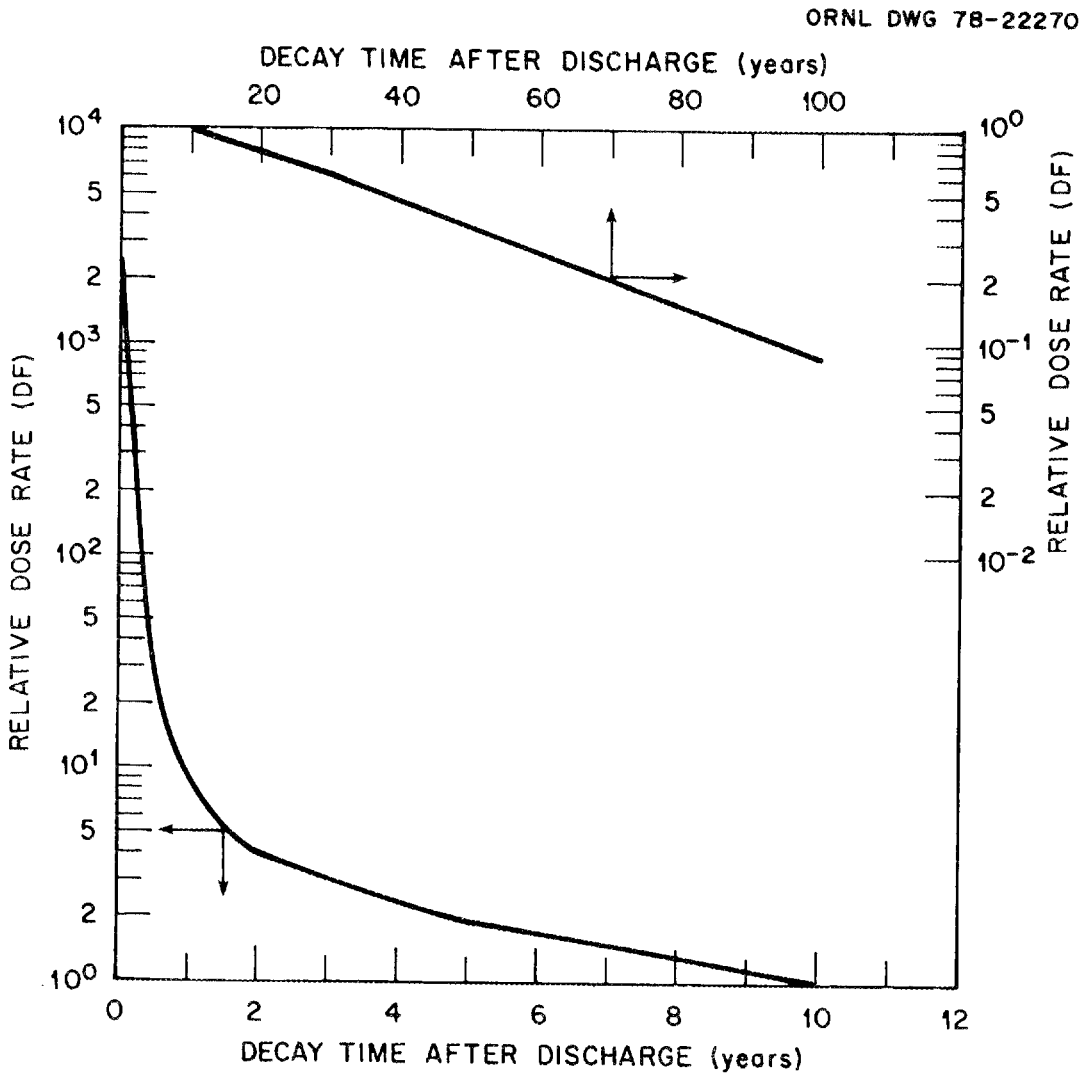


Fig. 7. Decay factor for fission products in a PWR fuel assembly.

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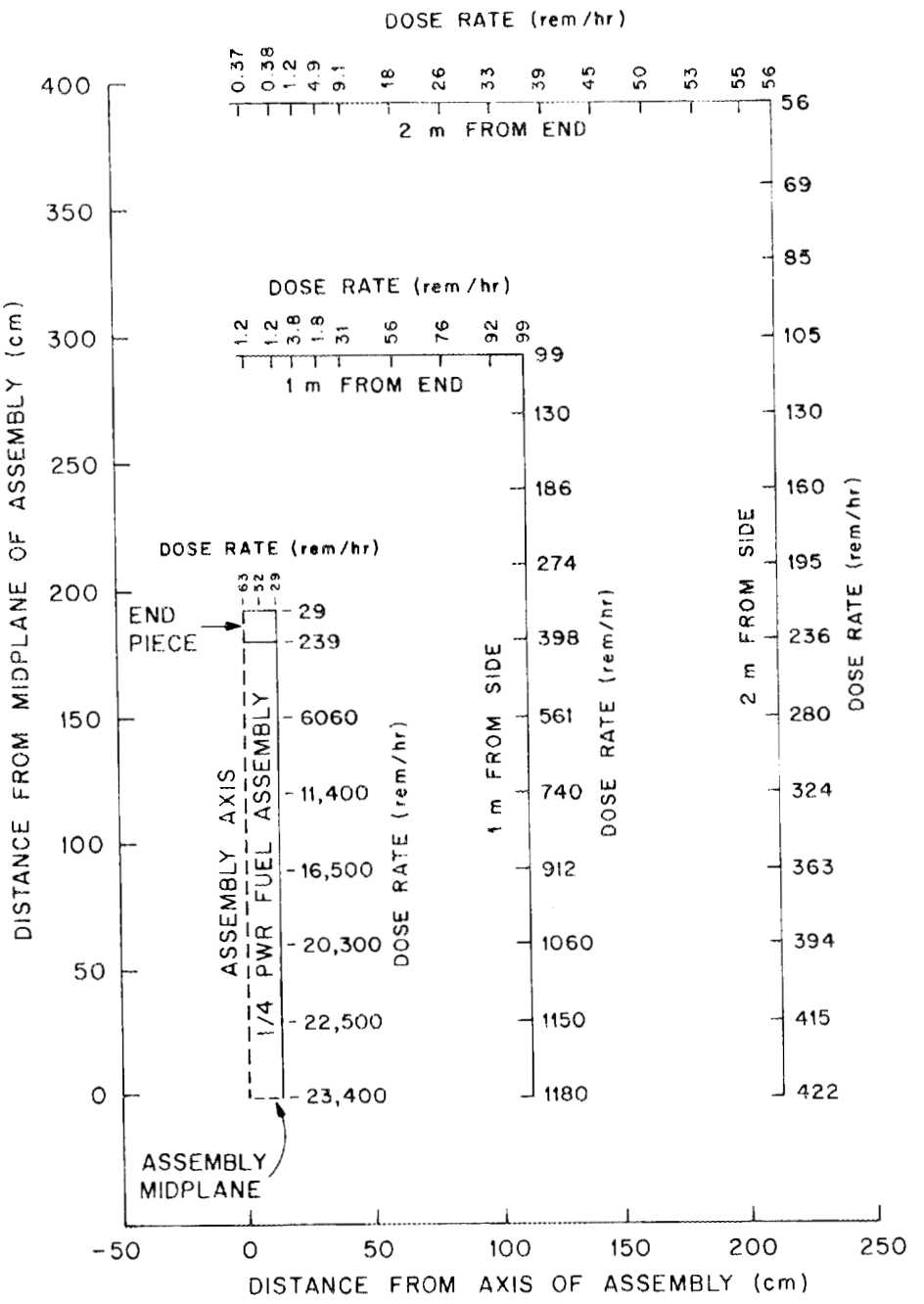


Fig. 8. Fission product dose rate from a 10-year-old PWR fuel assembly.

## 2.3 Dose Rate from a PWR Fuel Assembly Containing a $^{60}Co$ Spike

The dose rate from a PWR fuel assembly containing a uniformly distributed $^{60}Co$ spike was calculated using the methodology and model described in Sect. 2.1. Cobalt-60 was selected as the spike based on the work of Selle et al.[10,11] The results of the calculations can be summarized by the following equation:

$$\text{Dose rate (rem/hr)} = 32{,}900 \times \frac{C}{30} \times SG \times DF, \qquad (2)$$

where

32,900 = dose rate on the midplane surface of a PWR fuel assembly containing 30 parts of uniformly distributed $^{60}Co$ per million parts of initial uranium as metal,

C = the actual concentration of $^{60}Co$ in the fuel assembly in parts per million parts of initial uranium,

SG = source shape and geometry parameter from Fig. 9 (this parameter is keyed to the surfaces in Fig. 5),

DF = decay factor (taken from Fig. 10) for $^{60}Co$.

As an example, let us determine the dose rate from 30 ppm of $^{60}Co$ immediately after fabrication at the same position that was used in the fission product example in Sect. 2.2. In this example, C = 30, SG = 0.036 (from Fig. 9), and DF = 1.0 (from Fig. 10) since there is no decay after fabrication. Thus, the dose rate is

$$32{,}900 \times (30/30) \times 0.036 \times 1.0 = 1184 \text{ rem/hr}.$$

The source shape and geometry parameter shown in Fig. 9 accounts only for the position of the dose recipient with respect to the fuel assembly since the source distribution within the assembly is uniform. The decay factor given in Fig. 10 accounts only for the decay of $^{60}Co$ after fabrication since the gamma-ray spectrum does not change with time.

As with the fission product source, the dose rates from the 30 ppm of $^{60}Co$ uniformly distributed in the PWR fuel assembly are given in Fig. 11 for a zero decay time and discrete points on the surfaces.

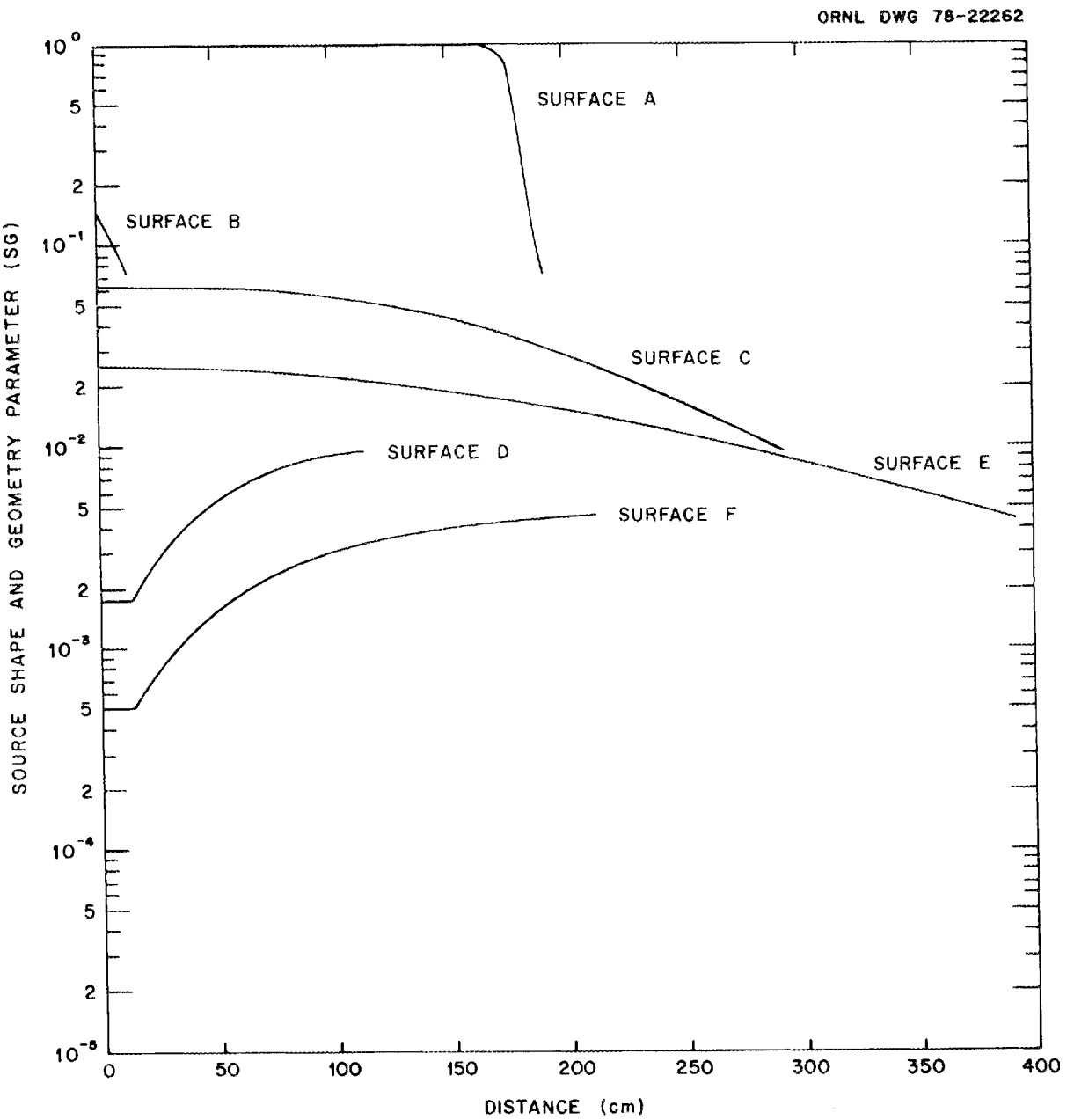


Fig. 9. Source shape and geometry parameter for a uniformly distributed spike.

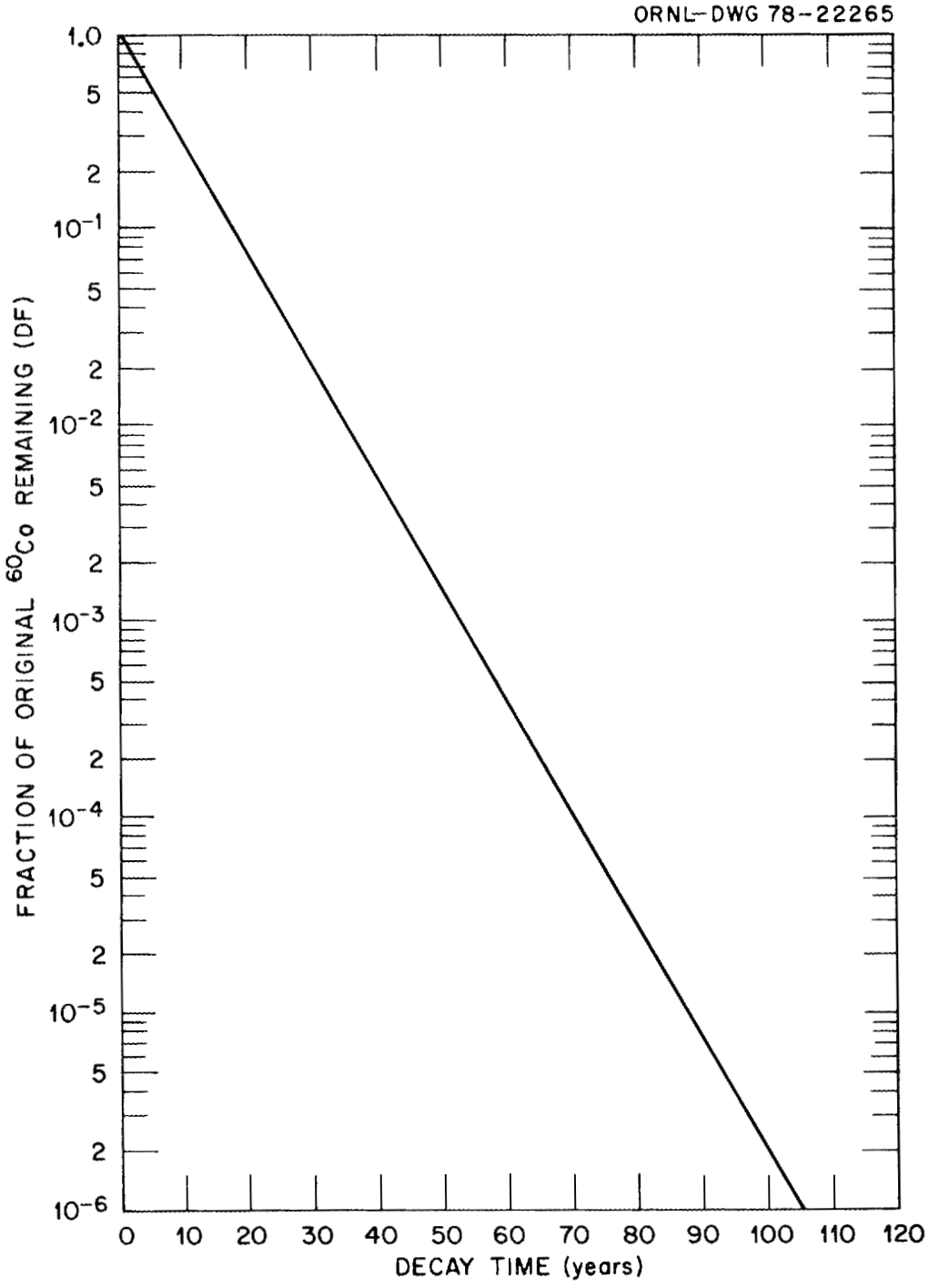


Fig. 10. Decay of $^{60}$Co spike after fabrication.

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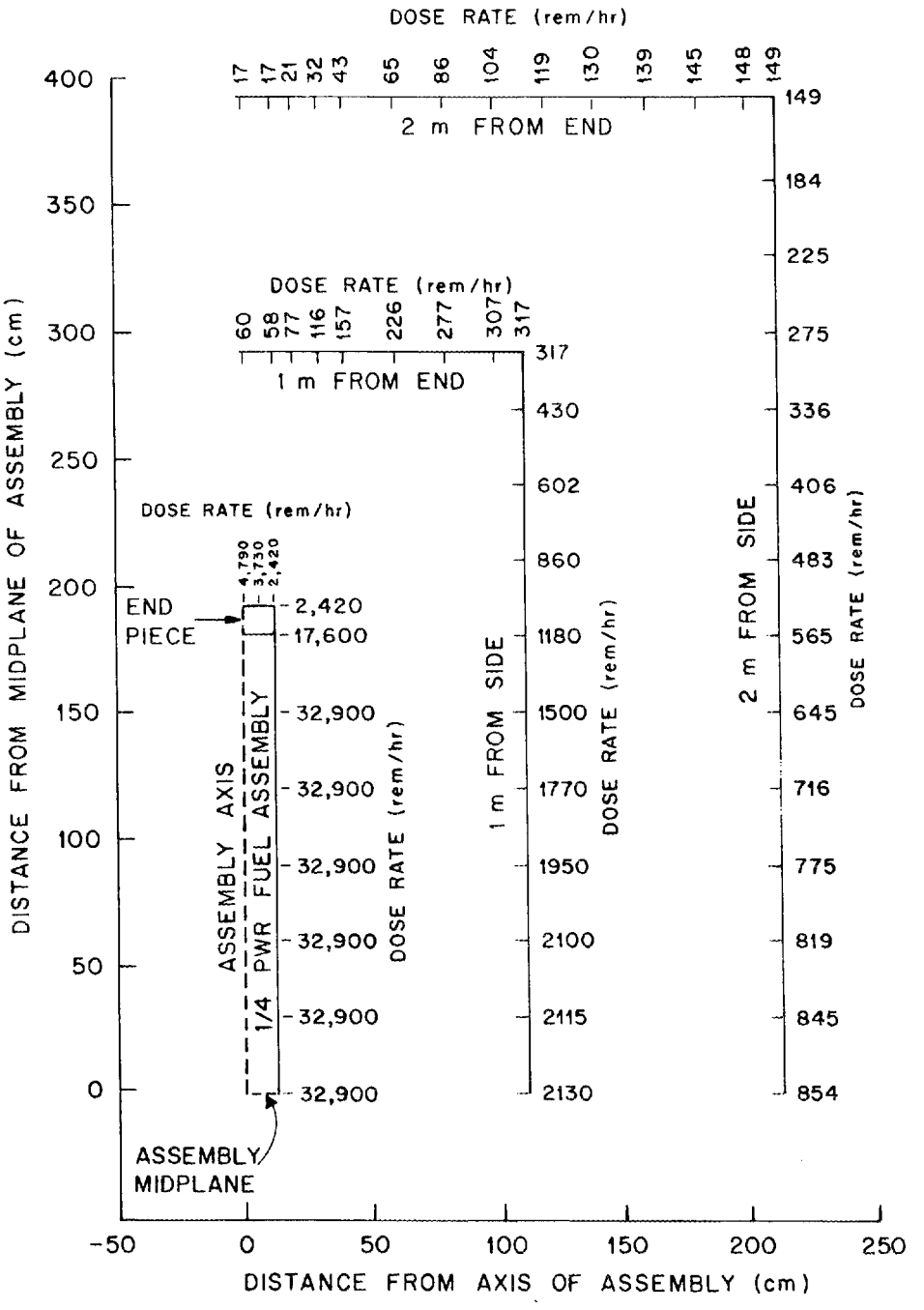


Fig. 11. Dose rate from 30 ppm $^{60}Co$ in a PWR fuel assembly.

The dose rates can be adjusted so as to reflect other times or concentrations simply by multiplying the value taken from Fig. 11 by (C x DF/30).

Finally, it should be noted that, even though Eq. (2) is only applicable to $^{60}Co$, the SG parameter that is given in Fig. 9 is applicable to any uniformly distributed source. All that is required for another spike is a single absolute value for the midplane surface dose and the construction of a simple DF-vs-time curve similar to Fig. 10.

## 2.4 Discussion

The purpose of this discussion is to briefly contrast and compare the dose rate calculations for the fission product and the spike. An absolute comparison of the values is not relevant since the intensity of the fission product source will vary with burnup and time and the intensity of the spike source will vary with time and can be changed arbitrarily by adjusting the concentration of the spike. However, the differences in the shapes of the dose rates around the fuel assembly and the reasons for them are of considerable interest. These comparisons will be drawn using Figs. 8 and 11 since they are easier to visualize when absolute dose rates are arranged in their real geometric pattern.

The most startling characteristic of the dose rate pattern from the fission products (Fig. 8) is the rapid decline in dose rate as one moves from the assembly midplane to the end of the assembly. This decline results from the approximately cosine shape of the fuel burnup, which drops to nearly zero at the end of the assembly. Thus, the $UO_2$ at the ends of the fuel elements in effect acts as a gamma shielding from the more intense gamma rays being emitted from the higher-burnup fuel near the midplane of the assembly. A second aspect of Fig. 8 that is not immediately evident by inspection is that, as one moves away from the assembly midplane in the perpendicular direction, the dose rate decreases more rapidly than would be the case for an inverse relationship with distance. The inverse variation with distance would result if the fuel assembly were infinitely long. The more rapid decline in dose rate is simply a reflection of the finite length of the fuel assembly.

Comparison of the dose rates from the uniformly distributed spike (Fig. 11) with those for the cosine-shaped fission product distribution (Fig. 8) indicates that the presence of substantial amounts of the spike in the ends of the fuel elements results in a much larger dose rate near the ends of the assembly as compared with that at the midplane. In fact, the ratio of the dose rate on the end piece surface to that at the midplane surface is about 50 times larger for the uniformly distributed source than for the fission product source.

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