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REGULATORY GUIDE

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ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS

A. INTRODUCTION

Section 50.34 of 10 CFR Part 50, "Contents of Applications: Technical Information," requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. A fuel handling accident in the fuel handling and storage facility resulting in damage to fuel cladding and subsequent release of radioactive material is one of the postulated accidents used to evaluate the adequacy of these structures, systems, and components with respect to the public health and safety. This safety guide gives acceptable assumptions that may be used in evaluating the radiological consequences of this accident for boiling and pressurized water reactors.

B. DISCUSSION

A fuel handling accident during refueling operations could release a fraction of the fission product inventory in a nuclear power plant to the environment. An illustrative accident sequence consists of the dropping of a fuel assembly resulting in breaching of the fuel rod cladding, release of a portion of

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the volatile fission gases from the damaged fuel rods, absorption of water soluble gases in and transport of soluble and insoluble gases through the water, air filtration (if provided) prior to release into the environment, and dispersion of the released fission products into the atmosphere.

The number and exposure histories of fuel assemblies assumed to be damaged determine the total amount of radioactive material available for immediate release into the water during a fuel handling accident. Although the design of the fuel, the fuel transfer equipment, the fuel pool, and the methods used to handle discharged fuel should all be considered in arriving at the number of fuel assemblies or rods assumed to be damaged, this guide rather than being addressed to this determination is addressed to the determination of the radiological consequences of a handling accident once an assumption as to the number of assemblies or rods damaged has been made.

A conservative approach to determining the quantity of radioactive material available for release from a fuel assembly is to assume that the assembly with the peak inventory is the one damaged. The inventory for the peak assembly represents an upper limit value and is not expected to be exceeded. The inventory should be calculated assuming maximum full power operation at the end of core life immediately preceding shutdown and such calculation should include an appropriate radial peaking factor.

Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation would be available for immediate release into the water in the event of clad damage. (Migration of fission products is a function of several variables including operating temperature, burnup, and isotopic half life taken in to consideration in establishing the release fractions listed in this guide.) As compared to the quantity immediately released, the quantity of radioactive material released subsequent to the immediate release is considered for the purposes of this guide to be negligible.

The assumptions set forth in this guide are based on engineering judgment and results from safety research programs conducted by the AEC and the nuclear industry and are believed to be appropriately conservative. In some cases unusual site characteristics, plant design features, or other factors may require different assumptions which will be considered on an individual case basis. Major changes in fuel composition or management may also require alterations of these assumptions.

C. REGULATORY POSITION

1. The assumptions[1] related to the release of radioactive material from the fuel and fuel storage facility as a result of a fuel handling accident are:

a. The accident occurs at a time after shutdown identified in the technical specifications as the earliest time fuel handling operations may begin. Radioactive decay of the fission product inventory during the interval between shutdown and commencement of fuel handling operations is taken into consideration.

[1] The assumptions given are valid only for oxide fuels of the types currently in use and in cases where the following conditions are not exceeded:

a. Peak linear power density of 20.5 kW/ft for the highest power assembly discharged.

b. Maximum center-line operating fuel temperature less than 4500°F for this assembly.

c. Average burnup for the peak assembly of 25,000 MWD/ton or less (this corresponds to a peak local burnup of about 45,000 MWD/ton).

b. The maximum fuel rod pressurization[2] is 1200 psig.

c. The minimum water depth[2] between the top of the damaged fuel rods and the fuel pool surface is 23 feet.

d. All of the gap activity in the damaged rods is released and consists of 10% of the total noble gases other than Kr-85,30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident. For the purpose of sizing filters for the fuel handling accident addressed in this guide, 30% of the I-127 and I-129 inventory is assumed to be released from the damaged rods.

e. The values assumed for individual fission product inventories are calculated assuming full power operation at the end of core life immediately preceding shutdown and such calculation should include an appropriate radial peaking factor. The minimum acceptable radial peaking factors are 1.5 for BWRs and 1.65 for PWRs.

f. The iodine gap inventory is composed of inorganic species (99.75%) and organic species (.25%).

g. The pool decontamination factors for the inorganic and organic species are 133 and 1, respectively, giving an overall effective decontamination factor of 100 (i.e., 99% of the total iodine released from the damaged rods is retained by the pool water). This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75% inorganic and 25% organic species.

h. The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).

i. The radioactive material that escapes from the pool to the building is released from the building[3] over a two hour time period.

j. If it can be shown that the building atmosphere is exhausted through adsorbers designed to remove iodine, the removal efficiency is 90% for inorganic species and 70% for organic species.[4]

k. The effluent from the filter system passes directly to the emergency exhaust system without mixing[5] in the surrounding building atmosphere and is then released (as an elevated plume for those facilities with stacks[6]).

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2 For release pressures greater than 1200 psig and water depths less than 23 feet, the iodine decontamination factors will be less than those assumed in this guide and must be calculated on an individual case basis using assumptions comparable in conservatism to those of this guide.

3 The effectiveness of features provided to reduce the amount of radioactive material available for release to the environment will be evaluated on an individual case basis.

4 These efficiencies are based upon a 2-inch charcoal bed depth with ¼ second residence time. Efficiencies may be different for other systems and must be calculated on an individual case basis.

5 Credit for mixing will be allowed in some cases; the amount of credit will be evaluated on an individual case basis.

6 Credit for an elevated release will be given only if the point of release is (a) more than two and one-half times the height of any structure close enough to affect the dispersion of the plume or (b) located far enough from any structure which could affect the dispersion of the plume. For those plants without stacks the atmospheric diffusion factors assuming ground level release given in regulatory position 2.b. should be used.

2. The assumptions for atmospheric diffusion are:

a. **Ground Level Releases**

(1) The basic equation for atmospheric diffusion from a ground level point source is:

$$\chi / Q = \frac{1}{\pi u \sigma_y \sigma_z}$$

Where:

χ = the short term average centerline value of the ground level concentration (curies/m^3)

Q = amount of material released (curies/sec)

u = windspeed (meters/sec)

σ_y = the horizontal standard deviation of the plume (meters) [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion, " F. A. Gifford, Jr.]

σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]

(2) For ground level releases, atmospheric diffusion factors[7] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions:

(a) windspeed of 1 meter/sec;
(b) uniform wind direction;
(c) Pasquill diffusion category F.

(3) Figure 1 is a plot of atmospheric diffusion factors (χ/Q) versus distance derived by use of the equation for a ground level release given in regulatory position 2.a.(1) above under the meteorological conditions given in regulatory position 2.a.(2) above.

(4) Atmospheric diffusion factors for ground level releases may be reduced by a factor ranging from one to a maximum of three (see Figure 2) for additional dispersion produced by the turbulent wake of the reactor building. The volumetric building wake correction as defined in Subdivision 3-3.5.2 of **Meteorology and Atomic Energy–1968,** is used with a shape factor of ½ and the minimum cross-sectional area of the reactor building only.

[7] These diffusion factors should be used until adequate site meteorological data are obtained. In some cases, available information on such site conditions as meteorology, topography and geographical location may dictate the use of more restrictive parameters to insure a conservative estimate of potential offsite exposures.

b. **Elevated Releases**

(1) The basic equation for atmospheric diffusion from an elevated release is:

$$\chi / Q = \frac{e^{-h^2} / 2\sigma_z{}^2}{\pi u \sigma_y \sigma_z}$$

Where:

χ = the short term average centerline value of the ground level concentration (curies/m^3)

Q = amount of material released (curies/sec)

u = windspeed (meters/sec)

σ_y = the horizontal standard deviation of the plume (meters [See Figure V-1, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4 , "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," F. A. Gifford, Jr.]

σ_z = the vertical standard deviation of the plume (meters) [See Figure V-2, Page 48, Nuclear Safety, June 1961, Volume 2, Number 4, "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion , " F. A. Gifford, Jr.]

h = effective height of release (meters)

(2) For elevated releases, atmospheric diffusion factors[7] used in evaluating the radiological consequences of the accident addressed in this guide are based on the following assumptions:

(a) windspeed of 1 meter/sec;
(b) uniform wind direction;
(c) envelope of Pasquill diffusion categories for various release heights;
(d) a fumigation condition exists at the time of the accident.[8]

(3) Figure 3 is a plot of atmospheric diffusion factors versus distance for an elevated release assuming no fumigation, and Figure 4 is for an elevated release with fumigation.

(4) Elevated releases are considered to be at a height equal to no more than the actual stack height. Certain site conditions may exist, such as surrounding elevated topography or nearby structures, which will have the effect of reducing the effective stack height. The degree of stack height reduction will be evaluated on an individual case basis.

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8 For sites located more than 2 miles from large bodies of water such as oceans or one of the Great Lakes, a fumigation condition is assumed to exist at the time of the accident and continue for one-half hour. For sites located less than 2 miles from large bodies of water a fumigation condition is assumed to exist at the time of the accident and continue for the duration of the release (2 hours).

3. The following assumptions and equations may be used to obtain conservative approximations of thyroid dose from the inhalation of radioiodine and external whole body dose from radioactive clouds:

a. The assumptions relative to inhalation thyroid dose approximations are:

(1) The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.

(2) No correction is made for depletion of the effluent plume of radioiodine due to deposition on the ground, or for the radiological decay of radioiodine in transit.

(3) Inhalation thyroid doses may be approximated by use of the following equation:

$$D = \frac{F_g IFPBR(\chi / Q)}{(DF_p)(DF_f)}$$

Where:

D	=	thyroid dose (rads)
F_g	=	fraction of fuel rod iodine inventory in fuel rod void space (0.1)
I	=	core iodine inventory at time of accident (curies)
F	=	fraction of core damaged so as to release void space iodine
P	=	fuel peaking factor
B	=	breathing rate = 3.47×10^{-4} cubic meters per second (i.e., 10 cubic meters per 8 hour work day as recommended by the ICRP)
DF_p	=	effective iodine decontamination factor for pool water
DF_f	=	effective iodine decontamination factor for filters (if present)
χ/Q	=	atmospheric diffusion factor at receptor location (sec/m^3)
R	=	adult thyroid dose conversion factor for the iodine isotope of interest (rads per curie). Dose conversion factors for Iodine 131-135 are listed in Table I.[9] These values were derived from "standard man" parameters recommended in ICRP Publication 2.[10]

[9] Dose conversion factors taken from "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, J. J. DiNunno, R. E. Baker, F. D. Anderson, and R. L. Waterfield (1962).

[10] Recommendations of the International Commission on Radiological Protection, "Report of Committee II on Permissible Dose for Internal Radiation (1959)," ICRP Publication 2, (New York: Pergamon Press, 1960).

Table 1. Adult Inhalation Thyroid Dose Conversion Factors

IODINE ISOTOPE	CONVERSION FACTOR (R) (RADS/CURIE INHALED)
131	1.48×10^6
132	5.35×10^4
133	4.0×10^5
134	2.5×10^4
135	1.24×10^5

b. The assumptions relative to external whole body dose approximations are:

(1) The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.

(2) External whole body doses are calculated using "Infinite Cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distances that the gamma rays and beta particles travel. The dose at any distance from the reactor is calculated based on the maximum ground level concentration at that distance. For an infinite uniform cloud containing χ curies of beta radioactivity per cubic meter, the beta dose rate in air at the cloud center is:[11]

$$\beta D'_\infty = 0.457 \bar{E}_\beta \chi$$

Where :

$\beta D'_\infty$ = beta dose rate from an infinite cloud (rad/sec)

$\bar{E}_\beta$ = average beta energy per disintegration (MeV/dis)

χ = concentration of beta or gamma emitting isotope in the cloud (curie/m^3)

Because of the limited range of beta particles in tissue, the surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount or:

$${}_\beta D'_\infty = 0.23 \bar{E}_\beta \chi$$

For gamma emitting material the dose rate in tissue at the cloud center is:

$${}_\gamma D'_\infty = 0.507 \bar{E}_\gamma \chi$$

[11] **Meteorology and Atomic Energy–1968,** Chapter 7.

Where :

$_{\gamma}D'_{\infty}$ = gamma dose rate from an infinite cloud (rad/sec)

$\overline{E}_{\gamma}$ = average gamma energy per disintegration (MeV/dis)

However, because of the presence of the ground, the receptor is assumed to be exposed to only one-half of the cloud (semi-infinite) and the equation becomes:

$$_{\gamma}D' = 0.25\overline{E}_{\gamma}\chi$$

Thus, the total beta or gamma dose to an individual located at the center of the cloud path may be approximated as:

$$_{\beta}D_{\infty} = 0.23\overline{E}_{\beta}\psi \text{ or}$$

$$_{\gamma}D = 0.25E_{\gamma}\psi$$

Where ψ is the concentration time integral for the cloud (curie sec/m^3).

The beta and gamma energies emitted per disintegration, as given in **Table of Isotopes,**[12] are averaged and used according to the methods described in ICRP Publication 2.

12 C. M. Lederer, J. M. Hollander, and I. Perlman, **Table of Isotopes**, Sixth Edition (New York: John Wiley and Sons, Inc. 1967).

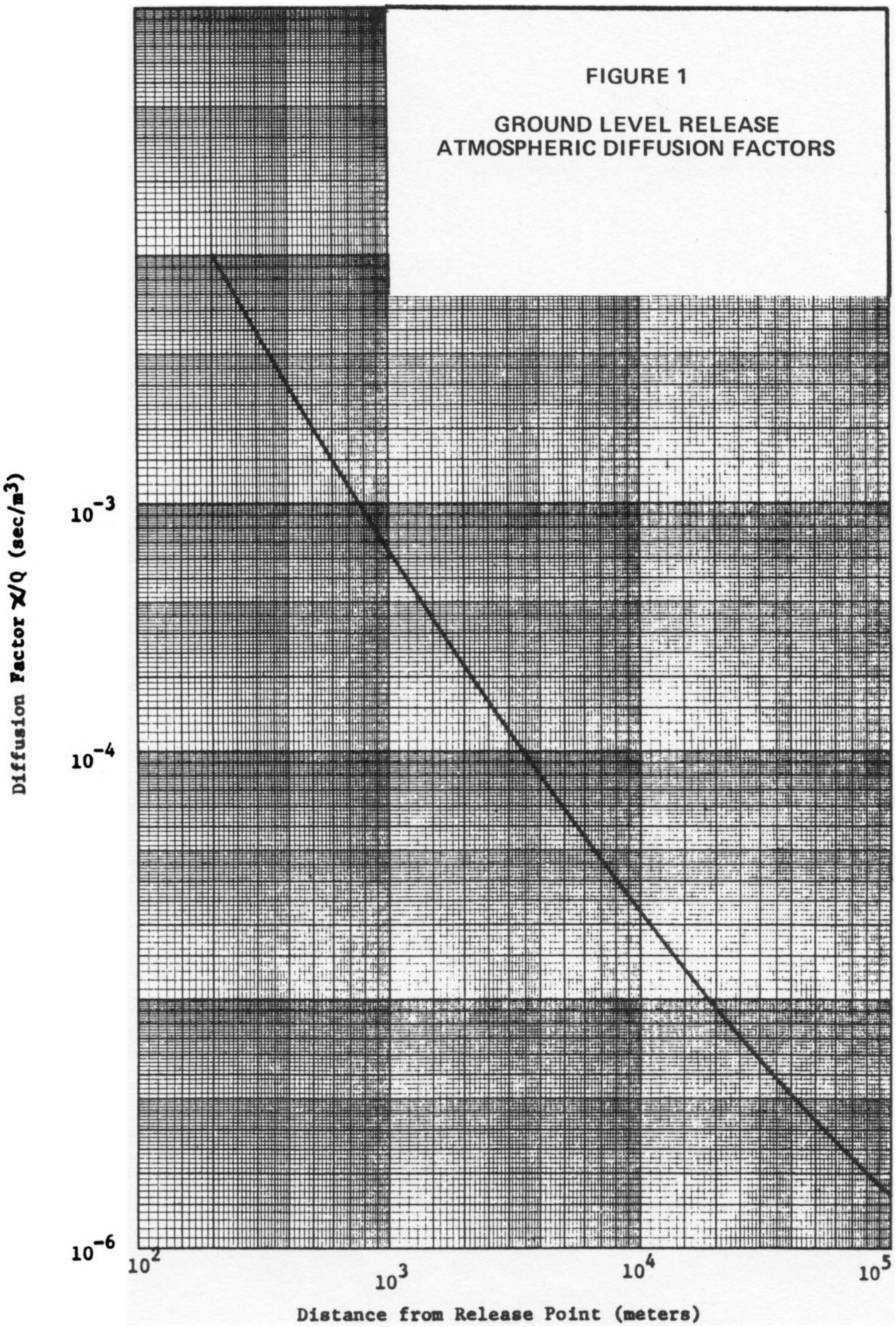

Figure 1. Ground Level Release Atmospheric Diffusion Factors

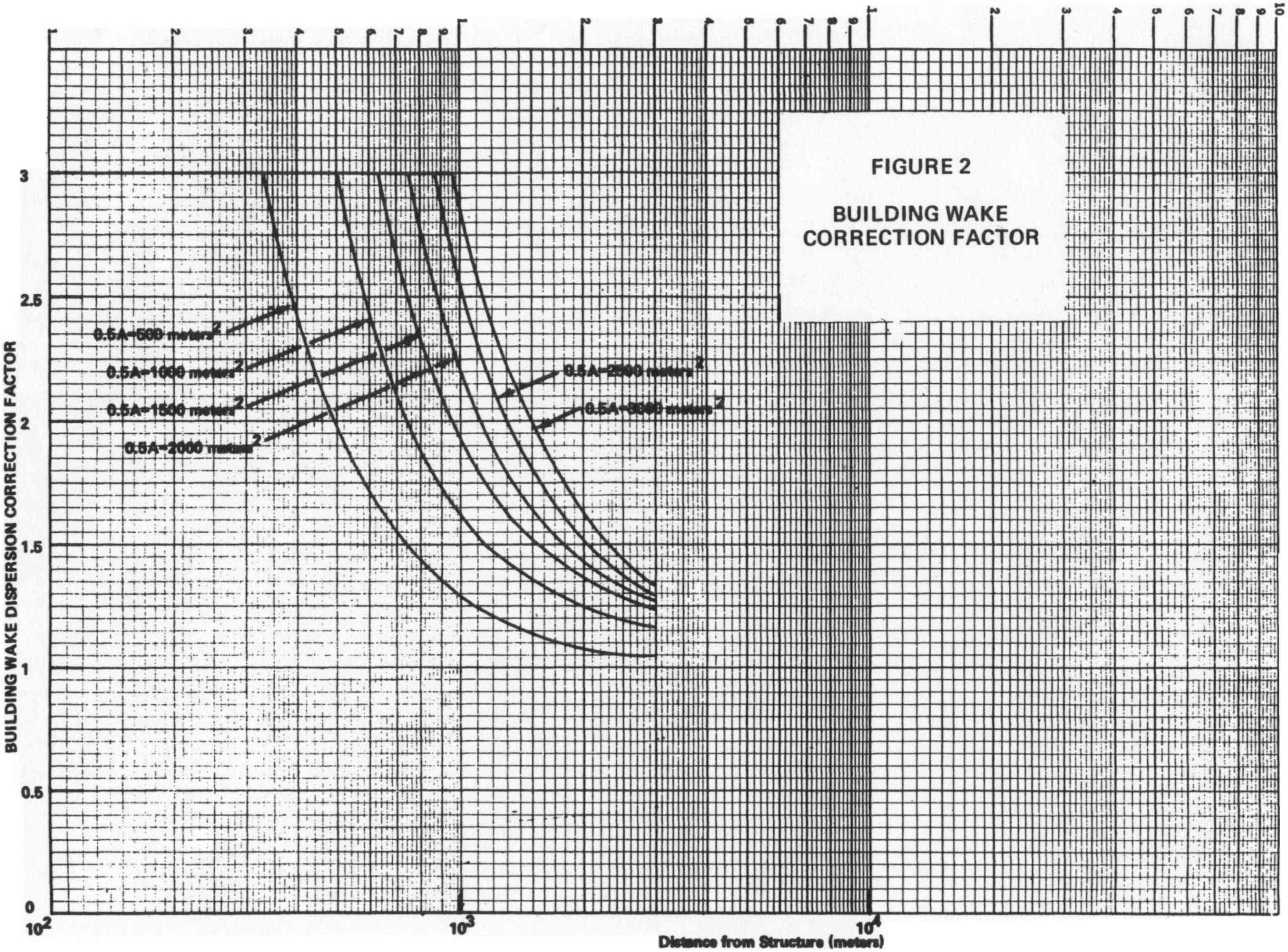

Figure 2. Building Wake Correction Factor

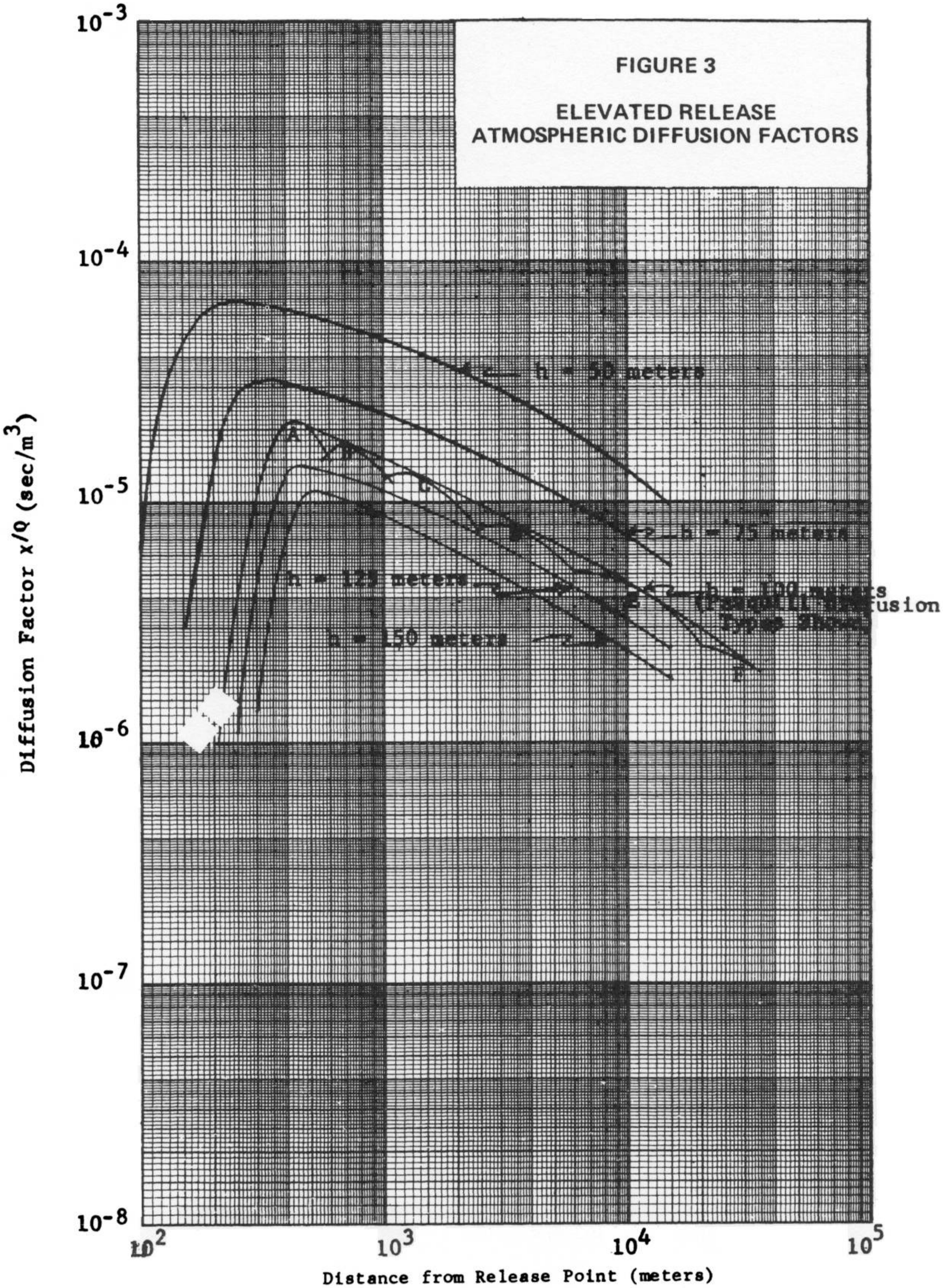

Figure 3. Elevated Release Atmospheric Diffusion Factors

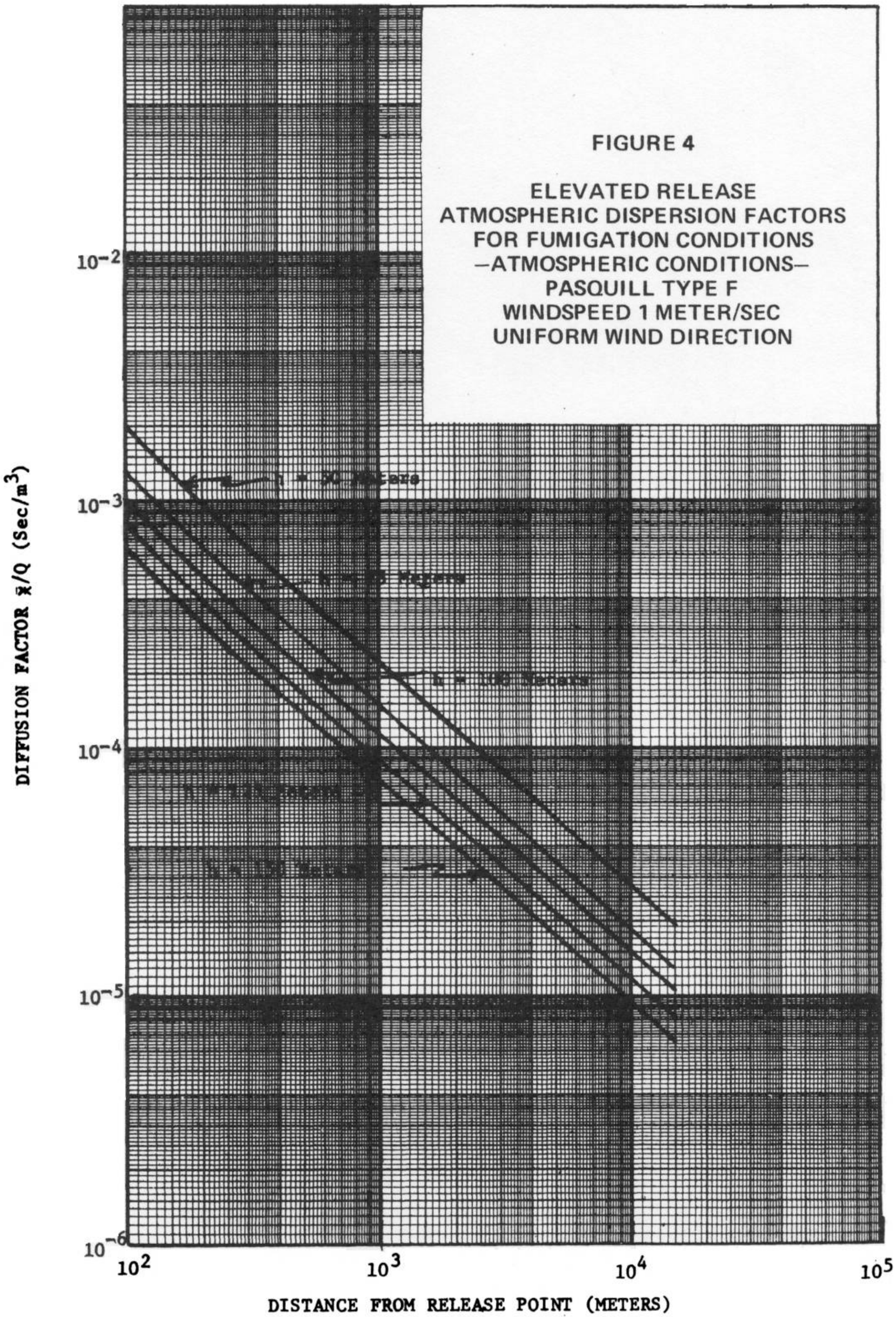

Figure 4. Elevated Release Atmospheric Dispersion Factors for Fumigation Conditions
—Atmospheric Conditions—
Pasquill Type F
Windspeed 1 Meter/Sec
Uniform Wind Direction