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ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF ACCIDENTAL NUCLEAR CRITICALITY IN A FUEL REPROCESSING PLANT

A. INTRODUCTION

Section 50.34, "Contents of Applications: Technical Information," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," 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 and including determination of the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

In a fuel reprocessing plant, a criticality accident is one of the postulated accidents used to evaluate the adequacy of an applicant's proposed activities with respect to the public health and safety. The methods described in this guide result from review and action on a number of specific cases and, as such, reflect the latest general NRC-approved approaches to the problem. If an applicant desires to employ new information that may be developed in the future or to use an alternative method, NRC will review the proposal and approve its use, if found acceptable.

B. DISCUSSION

In the process of reviewing applications for permits and licenses authorizing the construction or operation of fuel reprocessing plants, the NRC staff has developed appropriately conservative assumptions that are used by the staff to evaluate an estimate of the radiological consequences of various postulated accidents. These assumptions are based on previous accident experience, engineering judgment, and on the analysis of applicable experimental results from safety research programs. This guide lists assumptions used to evaluate the magnitude and radiological consequences of a criticality accident in a fuel reprocessing plant.

A criticality accident is an accident resulting in the uncontrolled release of energy from an assemblage of fissile material. The circumstances of a criticality accident are difficult to predict. However, the most serious criticality accident would be expected to occur when the reactivity (the extent of the deviation from criticality of a nuclear chain reacting medium) could increase most rapidly and without control in the fissile accumulation of largest credible mass. In a fuel reprocessing plant where conditions that might lead to criticality are carefully avoided because of the potential for adverse physical and radiological effects, such an accident is extremely uncommon. However, experience with these and related facilities has demonstrated that criticality accidents could occur.

In a fuel reprocessing plant, such an accident might be initiated by (1) inadvertent transfer or leakage of a solution of fissile material from a geometrically safe containing vessel into an area or vessel not so designed, (2) introduction of excess fissile material solution to a vessel, (3) introduction of excess fissile material to a solution, (4) overconcentration of a solution, (5) failure to maintain sufficient neutron absorbing materials in a vessel, (6) precipitation of fissile solids from a solution and their retention in a vessel, (7) introduction of neutron moderators or reflectors (e.g., by addition of water to a highly undermoderated system), (8) deformation of or failure to maintain safe storage arrays, or (9) similar actions that can lead to increases in the reactivity of fissile systems. Some acceptable means for minimizing the likelihood of such accidents are described in

Regulatory Guides 3.4, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors,"[1] and 3.1, "Use of Borosilicate Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material."[1]

1. Criticality Accident Experience in Relation to the Estimation of the Most Severe Accident

Stratton (Ref. 1) has reviewed in detail 34 occasions prior to 1966 when the power level of a fissile system increased without control as a result of unplanned or unexpected changes in its reactivity. Although only six of these incidents occurred in processing operations, and the remainder occurred mostly in facilities for obtaining criticality data or in experimental reactors, the information obtained and its correlation with the characteristics of each system have been of considerable value for use in estimating the consequences of accidental criticality in process systems. The incidents occurred in aqueous solutions of uranium or plutonium (10), in metallic uranium or plutonium in air (9), in inhomogeneous water-moderated systems (9), and in miscellaneous solid uranium systems (6).

The estimated total number of fissions per incident ranged from 1E+15[2] to 1E+20 with a median of about 2E+17. More recently another incident in a plutonium processing facility in Windscale (U.K.) was described in which a total yield of about 1E+15 fissions apparently occurred (Ref. 2). In ten cases, the supercriticality was halted by an automatic control device. In the remainder, the shutdown was effected as a consequence of the fission energy release that resulted in thermal expansion, density reduction from the formation of very small bubbles, mixing of light and dense layers, loss of water moderator by boiling, or expulsion of part of the mass.

Generally, the criticality incidents were characterized by an initial burst or spike in the curve of fission rate versus time followed by a rapid but incomplete decay as the shutoff mechanism was initiated. As more than one shutdown mechanism may affect the reactivity of the system and the effect of a particular mechanism may be counteracted, the initial burst was frequently succeeded by a plateau period of varying length. This plateau was characterized by a lesser and declining fission rate and finally by a further dropoff as shutdown was completed. The magnitude of the initial burst was directly related to the rate of increase of reactivity and its magnitude above the just-critical value but was inversely related to the background neutron flux, which is much greater for plutonium than for uranium systems.

Those systems consisting only of solid fissile, reflector, or moderator materials exhibited little or no plateau period, whereas solution systems had well developed plateaus. For solution systems, the energy release during the plateau period, because of its duration, provided the major portion of the total energy released. For purposes of the planning necessary to deal adequately with criticality incidents in experimental and production-type nuclear facilities, Woodcock (Ref. 3) made use of these data to estimate possible fission yields from excursions in various types of systems. For example, spike yields of 1E+17 and 1E+18 and total yields of 3E+18 and 3E+19 fissions were suggested for criticality accidents occurring in solution systems of 100 gallons or less and more than 100 gallons, respectively. Little or no mechanical damage was predicted at these levels.

2. Methods Developed for Predicting the Magnitude of Criticality Accidents

The nuclear excursion behavior of solutions of enriched uranium has been studied extensively both theoretically and experimentally. A summary by Dunenfeld and Stitt (Ref. 4) of the kinetic experiments on water boilers, using uranyl sulfate solutions, describes the development of a kinetic model that was confirmed by experiment. This model defines the effects of thermal expansion and radiolytic gas formation as power-limiting and shutdown mechanisms.

The results of a series of criticality excursion experiments resulting from the introduction of uranyl nitrate solutions to vertical cylindrical tanks at varying rates are summarized by Lécorché and Seale (Ref. 5). This report confirms the applicability of the kinetics model for solutions, provides correlations of peak power with reactivity addition rate, notes the importance of a strong neutron source in limiting peak power, and indicates the nature of the plateau following the peak.

Many operations with fissile materials in a fuel reprocessing plant are conducted with aqueous (or organic solvent) solutions of fissile materials. Consequently, well-founded methods for the prediction of total fissions and maximum fission rate for accidents that might occur in solutions (in process or other vessels) by the addition of fissile materials should be of considerable value in evaluating the effects of possible reprocessing plant criticality accidents. From the results of the excursion studies and from accident data, Tuck (Ref. 6) has developed methods for estimating (1) the maximum number of fissions in a 5-second interval (the first spike), (2) the total number of fissions, and (3) the maximum specific fission rate

[1] Copies may be obtained from the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Document Control.

[2] 1E+15 = 1 x 10^{15}. This notational form will be used throughout this guide.

in vertical cylindrical vessels, 28 to 152 cm in diameter and separated by > 30 cm from a bottom reflecting surface, resulting from the addition of up to 500 g/l solutions of Pu-239 or U-235 to the vessel at rates of 0.1 to 7.5 gal/min. Tuck also gives a method for estimating the power level from which the steam-generated pressure may be calculated and indicates that use of the formulas for tanks >152 cm in diameter is possible with a loss in accuracy.

Methods for estimating the number of fissions in the initial burst and the total number of fissions, derived from the work reported by Lécorché and Seale (Ref. 5), have also been developed by Olsen and others (Ref. 7). These were evaluated by application to ten actual accidents which have occurred in solutions and were shown to give conservative estimates in all cases except one.

Fission yields for criticality accidents occurring in solution and some heterogeneous systems, e.g., liquid/fixed geometry, can be reasonably estimated using existing methods. However, methods for estimating the possible fission yield from other types of heterogeneous systems, e.g., liquid/powder, are less reliable because of the uncertainties of predicting system reactivity rate. The uncertainties of geometry and moderation result in a broad range of possible yields.

Woodcock (Ref. 3) estimated that in solid plutonium systems, solid uranium systems, and heterogeneous liquid/powder systems (fissile material not specified) total fission yields (substantially occurring within the spike) of 1E+18, 3E+19, and 3E+20, respectively, could be predicted. Mechanical damage varied from slight to extensive. Heterogeneous systems consisting of metals or solids in water were estimated to achieve a possible magnitude of 1E+19 following an initial burst of 3E+18 fissions. Operations in a fuel reprocessing plant involve only a small number of complete assemblies of fuel rods, except in the fuel storage pool. In the latter area, a rigid array of assemblies is maintained and normally only a single assembly may be in motion in the vicinity of the array. Consequently, the rate of reactivity addition in such a system would be quite low, and the predicted magnitude of a criticality incident would be correspondingly low. These estimates could aid in the analysis of situations in plant systems. However, they should not be taken as absolute values for criticality assumptions for the purpose of this guide.

For systems other than solution systems, the estimation of the peak fission rate and the total number of fissions accompanying an accidental nuclear criticality may be accomplished with the aid of information derived from accident experience, from experiments on reactors utilizing bare uranium metal (Ref. 8), and from the SPERT-1 reactor transient tests with light- and heavy-water moderated uranium-aluminum and UO_2-stainless steel fuels (Ref. 9). Oxide core tests in the latter group provide some information on energy release mechanisms that may be effective, for example, in spent fuel storage or fuel leaching systems in a reprocessing plant. Review of unusual reprocessing structures, systems, and components for the possibility of accidental criticality should also consider recognized anomalous situations in which the possibility of accidental nuclear criticality may be conceived (Ref. 10).

The application of the double-contingency principle[3] to fissile material processing operations has been successful in reducing the probability of accidental criticality to a low value. As a consequence, the scenarios required to arrive at accidental criticality involve the assumption of multiple breakdowns in the nuclear criticality safety controls. It has therefore been a practice to simply and conservatively assume an accidental criticality of a magnitude equal to, or some multiple of, the historical maximum for all criticality accidents outside reactors without using any scenario clearly defined by the specific operations being evaluated. In the absence of sufficient guidance, there has been wide variation in the credibility of the postulated magnitude of the occurrence (particularly the size of the initial burst), the amount of energy and radioactivity assumed to be released, and the magnitude of the calculated consequences.

It is the staff's judgment that the evaluation of the criticality accident should assume the simultaneous breakdown of at least two independent controls throughout all elements of the operation. Each control should be such that its circumvention is of very low probability. Experience has shown that the simultaneous failure of two independent controls is very unlikely if the controls are derived, applied, and maintained with a high level of quality assurance. However, if controls highly dependent on human actions are involved, this approach will call for some variation in the assumed number of control failures. The criticality accidents so conceived should then be analyzed to determine the most severe within the framework of assumed control failures, using realistic values of such variables as the fissile inventory, vessel sizes, and pump transfer rates.

3. Radiological Consequences of Accidental Criticality

Past practice has been to evaluate the radiological consequences to individuals of postulated accidental criticality in fuel reprocessing plants in terms of a frac-

[3] The double-contingency principle is defined in ANSI N16-1-1969, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," which is endorsed by Regulatory Guide 3.4.

tion of the guideline values in 10 CFR Part 100, "Reactor Site Criteria."

The consequences of a criticality accident may be limited by containment, shielding, isolation distance, or evacuation of adjacent occupied areas subsequent to detection of the accident. If the impact of a criticality accident is to be limited through evacuation of adjacent occupied areas, there should be prior, formal arrangements with individual occupants and/or local authorities sufficient to ensure that such movements can be effected in the time allowed.

C. REGULATORY POSITION

1. Following are the plant assessment and assumptions related to energy release from a criticality accident and the minimum criticality accident to be considered:

a. When defining the characteristics of an assumed criticality accident in order to assess the adequacy of structures, systems, and components provided for the mitigation of the consequences of accidents, the applicant should evaluate credible criticality accidents in all those elements of the plant provided for the storage, handling, or processing of fissile materials or into which fissile materials in significant amounts could be introduced. To determine the circumstances of the criticality accidents, controls judged equivalent to at least two highly reliable, independent criticality controls should be assumed to be circumvented. The magnitude of the possible accidents should then be assessed, on an individual case basis, to estimate the extent and nature of possible effects and to provide source terms for dose calculations. The most severe accident should then be selected for the assessment of the adequacy of the plant.

Calculation of the radioactivity of fission products and transuranic elements initially present and later produced in the incident should be accomplished by computer codes ORIGEN (Ref. 11) and RIBD (Ref. 12), respectively. An equivalent calculation may be substituted, if justified on an individual case basis.

b. If the results of the preceding evaluation indicate that no possible criticality accident exceeds in severity the criticality accident postulated in this section, then the conditions of the following example may be assumed for the purpose of assessing the adequacy of the facility. A less conservative set of conditions may be used if they are shown to be applicable by the specific analyses conducted in accordance with paragraph C.1.a above.

An excursion is assumed to occur in a vented vessel of unfavorable geometry containing a solution of 400 g/l of uranium enriched to less than 5% U-235. The solution is also assumed to contain all of the transuranic elements and fission products, except the noble gases, expected to be present in the spent fuel at the maximum burnup and the minimum postirradiation decay time for which the plant is designed. These data included in this guide (see Table 1) list the radioactivity of available significant nuclides assuming 100% dissolution, the burnup to be 33,000 MWd/MTU, and a postirradiation decay time of 150 days.

The vessel is assumed to be located within a ventilated cell which provides shielding equivalent to 5 feet of concrete with a density of 142 lb/ft^3. The excursion produces an initial burst of 1E+18 fissions in 0.5 second followed successively at 10-minute intervals by 47 bursts of 1.9E+17 fissions for a total of 1E+19 fissions in 8 hours. The excursion is assumed to be terminated by evaporation of 100 liters of a solution containing 400 g/l of uranium (<5% enriched) and concentrations of associated fission products and transuranic elements corresponding to the sum of those produced in the incident plus those present in irradiated fuel (assuming 100% dissolution) for the plant design condition. However, the noble gas fission products initially present in the fuel are assumed to have been removed prior to the incident. Table 2 lists the radioactivity of significant nuclides released from the criticality accident.

2. Assumptions related to the release of radioactive material are as follows:[4]

a. It should be assumed that all of the noble gas fission products (except those removed prior to the excursion), 25% of the iodine radionuclides, and 0.1% of the ruthenium radionuclides resulting from the excursion or initially present in the spent fuel are released directly to the cell atmosphere. It should also be assumed that an aerosol, which is generated from the evaporation of solution during the excursion, is released directly to the cell atmosphere. The aerosol should be assumed to comprise 0.05% of the salt content of the solution that is evaporated. The cell volume and ventilation rate should be considered on an individual case basis.

b. The effects of radiological decay during transit in cell and in the plant exhaust system should be taken into account on an individual case basis.

c. The reduction in the amount of radioactive material available for release to the environment through the plant stack(s) as a result of the normal operation of sorption or filtration systems in the plant exhaust systems may be taken into account, but the amount of reduction in the concentration of radioactive materials should be evaluated on an individual case basis.

[4] Certain assumptions for release of radioactive material, dose conversions, and atmospheric diffusion reflect the staff's position indicated in Regulatory Guide 1.3 (Ref. 22).

3. Acceptable assumptions for dose and dose conversion are as follows:

a. The applicant should show that the consequences of the prompt gamma and neutron dose are sufficiently mitigated to allow occupancy of areas necessary to maintain the plant in a safe condition following the accident. The following semi-empirical equations should be used for these calculations. These equations are acceptable to the NRC staff and were developed from experimental data. Different methods may be substituted, if justified on an individual case basis. Potential total dose attenuation due to shielding and dose exposures should be evaluated on an individual case basis.

(1) Prompt[5] Gamma Dose

$$D\gamma = 2.1E-20N\, d^{-2}\, e^{-3.4d}$$

where

$D\gamma$ = gamma dose (rem)

N = number of fissions

d = distance from source (km)

Data presented in *The Effects of Nuclear Weapons* (Ref. 13, p. 384) should be used to develop dose reduction factors. For concrete, the dose should be reduced by a factor of 2.5 for the first 8 inches, a factor of 5.0 for the first foot, and a factor of 5.5 for each additional foot.

(2) Prompt Neutron Dose

$$Dn = 7E-20N\, d^{-2}\, e^{-5.2d}$$

where

Dn = neutron dose (rem)

N = number of fissions

d = distance from source (km)

For concrete, the dose should be reduced by a factor of 2.3 for the first 8 inches, 4.6 for the first foot, and a factor of 20 for each additional foot.

b. No correction should be made for depletion of the effluent plume of radioactive iodine due to deposition on the ground or for the radiological decay of iodine in transit.

c. For the first 8 hours, the breathing rate of a person offsite should be assumed to be 3.47E-4 m^3/sec. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.75E-4 m^3/sec. These values were developed from the average daily breathing rate (2E + 7 cm^3/day) assumed in the report of ICRP Committee II-1959 (Ref. 14).

d. External whole body doses should be calculated using "infinite cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. "Such a cloud would be considered an infinite cloud for a receptor at the center because any additional [gamma and] beta emitting material beyond the cloud dimensions would not alter the flux of [gamma rays and] beta particles to the receptor." [See *Meteorology and Atomic Energy-1968* (Ref. 15), Section 7.4.1.1; editorial additions made so that gamma and beta emitting material could be considered.] Under these conditions the rate of energy absorption per unit volume is equal to the rate of energy released per unit volume. 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

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

The surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount (i.e., $\beta D\tilde{\infty} = 0.23\ \bar{E}_\beta \chi$). For gamma emitting material, the dose rate in air at the cloud center is

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

From a semi-infinite cloud, the gamma dose rate in air is

$$\gamma D\tilde{\infty} = 0.25\, \bar{E}_\gamma \chi$$

where

$\beta D\tilde{\infty}$ = beta dose rate from an infinite cloud (rad/sec)

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

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

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

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

e. The following specific assumptions are acceptable with respect to the radioactive cloud dose calculations:

(1) The dose at any distance from the plant should be calculated based on the maximum concentration time integral (in the course of the accident) in the plume at that distance, taking into account specific meteorological, topographical, and other

[5] Most of the neutron and part of the gamma radiation are emitted in the actual fission process. Some gamma radiation is produced in various secondary nuclear processes, including decay of fission products. For the purposes of this guide, "prompt" gamma doses should be evaluated including the effects of decay of significant fission products during the first minute of the excursion.

characteristics that may affect the maximum plume concentration. These site-related characteristics should be evaluated on an individual case basis. In the case of beta radiation, the receptor is assumed to be exposed to an infinite cloud at the maximum ground level concentration at that distance from the plant. In the case of gamma radiation, the receptor is assumed to be exposed to only one-half the cloud owing to the presence of the ground. The maximum cloud concentration should always be assumed to be at ground level.

(2) The appropriate average beta and gamma energies emitted per disintegration used should be as given in the *Table of Isotopes* (Ref. 16).

(3) The whole body dose should be considered as the dose from gamma radiation at a depth of 5 cm and the genetic dose at a depth of 1 cm. The skin dose should be the sum of the surface gamma dose and the beta dose at a depth of 7 gm/cm^2. The beta skin dose may be estimated by applying an energy dependent attenuation factor (D_d/D_B) to the surface dose according to a method developed by Loevinger, Japha, and Brownell (Ref. 17). (See Figure 1.)

f. The "critical organ" dose from the inhaled radioactive materials should be estimated. The "critical organ" is that organ which receives the highest radiation dose after the isotope is absorbed into the body. For the purpose of this guide, the following assumptions should be made:

(1) The radionuclide dose conversion factors are as recommended by the report of Committee II, ICRP (Ref. 14).

(2) The effective half-life for the nuclide is as recommended in ICRP Publication 6 (Ref. 18).

(3.) The plutonium and other actinide nuclide clearance half time, or fraction of nuclide clearing the organ, is as recommended by the ICRP task group on lung dynamics (Ref. 19). A computer code, DACRIN, (Ref. 20) is available for this model. Task group lung model (TGLM) clearance parameters are presented in Table 3; the model is shown schematically in Figure 2.

g. The potential dose for all significant nuclides should be estimated for the population distribution on a site-related basis.

4. Acceptable assumptions for atmospheric diffusion are as follows:

a. Elevated releases should be considered to be at a height equal to no more than the actual stack height.[6] Certain site-dependent conditions may exist, such as surrounding elevated topography or nearby structures, that will have the effect of reducing the actual stack height. The degree of stack height reduction should be evaluated on an individual case basis.

Also, special meteorological and geographical conditions may exist which can contribute to greater ground level concentrations in the immediate neighborhood of a stack. For example, fumigation should always be assumed to occur; however, the length of time that a fumigation condition exists is strongly dependent on geographical and seasonal factors and should be evaluated on an individual case basis.[7] (See Fig. 3 for elevated releases under fumigation conditions.)

b. For plants with stacks, the atmospheric diffusion model should be as follows:

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

$$\chi/Q = \frac{\exp(-h_e^2/2\sigma_z^2)}{\pi\mu\sigma_y\sigma_z}$$

where

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

Q = amount of material release (Ci/sec)

μ = windspeed (m/sec)

σ_y = the horizontal standard deviation of the plume (meters). (See Ref. 21, Figure V-1, p.48.)

σ_z = the vertical standard deviation of the plume (meters). (See Ref. 21, Figure V-2, p.48.)

h_e = effective height of release (m)[8]

[6] Credit for an elevated release should be given only if the point of release is (1) more than two and one-half times the height of any structures close enough to affect the dispersion of the plume or (2) located far enough from any structure that could have an effect on the dispersion of the plume. For these plants without stacks, the atmospheric diffusion factors assuming ground level releases, as shown in Regulatory Position 4.c, should be used.

[7] For sites located more than 2 miles from large bodies of water, such as oceans or one of the Great Lakes, a fumigation condition should be 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 should be assumed to exist at the time of the accident and continue for 4 hours.

[8] $h_e = h_s - h_t$, where h_s is the height of the release above plant grade and h_t is the maximum terrain height, above plant grade, between the point of release and the point at which the calculation is made. h_t should not be allowed to exceed h_s.

(2) For time periods of greater than 8 hours, the plume from an elevated release should be assumed to meander and spread uniformly over a 22.5° sector.[9] The resultant equation is

$$\chi/Q = \frac{2.032 \exp(-h^2/2\sigma_z^2)}{\sigma_z \mu x}$$

where

x = distance from the release point (meters); other variables are as given in b(1).

(3) The atmospheric diffusion model[10] for an elevated release as a function of the distance from the plant is based on the information in the table below

Time Following Accident	Atmospheric Conditions
0 to 8 hours	See Figure 4 for Envelope of Pasquill diffusion categories [based on Figure A7, *Meteorology and Atomic Energy—1968* (Ref. 15), assuming various stack heights] windspeed 1 m/sec; uniform direction.
8 to 24 hours	See Figure 5 for Envelope of Pasquill diffusion categories; windspeed 1 m/sec; variable direction within a 22.5° sector.

c. For facilities exhausted without stacks, the atmospheric diffusion model should be as follows:

(1) The 0-to-8 hour ground level release concentrations may be reduced by a factor ranging from one to a maximum of three (see Figure 6) for additional dispersion when calculating nearby potential exposures. The volumetric building wake correction factor, as defined in Section 3-3.5.2 of *Meteorology and Atomic Energy—1968* (Ref. 15), should be used in the 0-to-8 hour period only; it is used with a shape factor of one-half and the minimum cross-sectional area of a major building.

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

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

where

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

Q = amount of material release (Ci/sec)

μ = windspeed (m/sec)

σ_y = the horizontal standard deviation of the plume (m). (See Ref. 21, Figure V-1, p.48.)

σ_z = the vertical standard deviation of the plume (m). (See Ref.21, Figure V-2, p.48.)

(3) For time periods of greater than 8 hours, the plume should be assumed to meander and spread uniformly over a 22.5° sector.[9] The resultant equation is

$$\chi/Q = \frac{2.032}{\sigma_z \mu x}$$

where

x = distance from point of release to the receptor; other variables are as given in c(2).

(4) The atmospheric diffusion model for ground level releases is based on the information in the following table:

Time Following Accident	Atmospheric Conditions
0 to 8 hours	Pasquill Type F, windspeed 1 m/sec, uniform direction
8 to 24 hours	Pasquill Type F, windspeed 1 m/sec, variable direction within a 22.5° sector.

(5) Figures 7A and 7B give the ground level release atmospheric diffusion factors based on the parameters given in c(4).

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the staff's plans for using this regulatory guide.

Except in those cases in which the applicant proposes an alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used in the evaluation of submittals for operating license or construction permit applications docketed after December 1, 1977.

If an applicant wishes to use this regulatory guide in developing submittals for applications docketed on or before December 1, 1977, the pertinent portions of the application will be evaluated on the basis of this guide.

[9] The sector may be assumed to shift after 8 hours if local meteorological data are available to justify a wind direction change. This should be considered on an individual case basis.

[10] In some cases, site-dependent parameters such as meteorology, topography, and local geography may dictate the use of a more restrictive model to ensure a conservative estimate of potential off-site exposures. Site-related meteorology should be developed on an individual case basis. If adequate local meteorological data are not available, this model should be used.

REFERENCES

1. W. R. Stratton, "Review of Criticality Incidents," LA-3611, Los Alamos Scientific Laboratory (Jan. 1967).

2. T. G. Hughes, "Criticality Incident at Windscale," *Nuclear Engineering International*, Vol. 17, No. 191, pp.95-7 (Feb. 1972).

3. E. R. Woodcock, "Potential Magnitude of Criticality Accidents," AHSB(RP)R-14, United Kingdom Atomic Energy Authority.

4. M.S. Dunenfeld, R. K. Stitt, "Summary Review of the Kinetics Experiments on Water Boilers," NAA-SR-7087, Atomic International (Feb. 1973).

5. P. Lécorché, R. L. Seale, "A Review of the Experiments Performed to Determine the Radiological Consequences of a Criticality Accident," Y-CDC-12, Union Carbide Corp. (Nov. 1973).

6. G. Tuck, "Simplified Methods of Estimating the Results of Accidental Solution Excursions," *Nucl. Technol.*, Vol. 23, p.177 (1974).

7. A. R. Olsen, R. L. Hooper, V. O. Uotinen, C. L. Brown, "Empirical Model to Estimate Energy Releases from Accidental Criticality," *ANS Trans.*, Vol. 19, pp. 189-91 (1974).

8. T. F. Wimmette et al., "Godiva 2—An Unmoderated Pulse Irradiation Reactor," *Nucl. Sci. Eng.*, Vol. 8, p.691 (1960).

9. W. E. Nyer, G. O. Bright, R. J. McWhorter, "Reactor Excursion Behavior," International Conference on the Peaceful Uses of Atomic Energy, paper 283, Geneva (1966).

10. E. D. Clayton, "Anomalies of Criticality," *Nucl. Technol.*, Vol. 23, No. 14 (1974).

11. M. J. Bell, "ORIGEN—The ORNL Isotope Generation and Depletion Code," ORNL-4628, Oak Ridge National Laboratory (May 1973).

12. R. O. Gumprecht, "Mathematical Basis of Computer Code RIBD," DUN-4136, Douglas United Nuclear, Inc. (June 1968).

13. *The Effects of Nuclear Weapons*, Revised Ed., Samuel Glasstone, Editor, U.S. Depart. of Defense (Feb. 1964).

14. "Permissible Dose for International Radiation," Publication 2, Report of Committee II, International Commission on Radiological Protection (ICRP), Pergamon Press (1959).

15. *Meteorology and Atomic Energy—1968*, D. H. Slade, Editor, U.S. Atomic Energy Commission (July 1968).

16. C. M. Lederer, J. M. Hollander, I. Perlman, *Table of Isotopes*, 6th Ed., Lawrence Radiation Laboratory, Univ. of California, Berkeley, CA.

17. *Radiation Dosimetry*, G. J. Hine and G. L. Brownell, Editors, Academic Press, New York (1956).

18. *Recommendations of ICRP*, Publication 6, Pergamon Press (1962).

19. "The Metabolism of Compounds of Plutonium and Other Actinides," a report prepared by a Task Group of Committee II, ICRP, Pergamon Press (May 1972).

20. J. R. Houston, D. L. Strenge, and E. C. Watson, "DACRIN—A Computer Program for Calculating Organ Dose from Acute or Chronic Radionuclide Inhalation," BNWL-B-389(UC-4), Battelle Memorial Institute, Pacific Northwest Laboratories, Richland, WA.(Dec.1974).

21. F. A. Gifford, Jr., "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," *Nuclear Safety*, Vol. 2, No. 4, p.48 (June 1961).

22. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," U.S. Nuclear Regulatory Commission, Washington, D.C.

TABLE 1

ASSUMED FISSION PRODUCT AND TRANSURANIC NUCLIDE RADIOACTIVITY IN SPENT FUEL SOLUTION PRIOR TO CRITICALITY INCIDENT

3.3% Enriched Fuel Irradiated to 33000 MWd/MTU, cooled 150 days and calculated by ORIGEN code.

NUCLIDE	CURIES/LITER
Tritium	2.9E - 1
Strontium—89	4.0E+1
Strontium—90	3.2E+1
Yttrium—90	3.2E+1
Yttrium—91	5.7E+1
Zirconium—95	1.2E+2
Niobium—95	2.2E+2
Ruthenium—103	3.7E+1
Rhodium—103M	3.7E+1
Ruthenium—106	1.7E+2
Rhodium—106	1.7E+2
Iodine—129	1.6E - 5
Iodine—131	9.1E - 4
Xenon—131m	1.4E - 3
Cesium—139	9.0E+1
Cesium—137	4.5E+1
Barium—137M	4.2E+1
Cerium—141	2.4E+1
Cerium—144	3.2E+2
Praseodymium—144	3.2E+2
Promethium—147	4.2E+1
Europium—154	2.3E 0
Plutonium—238	1.2E 0
Plutonium—239	1.4E - 1
Plutonium—240	2.0E - 1
Plutonium—241	4.8E+1
Americium—241	8.4E - 2
Curium—242	6.3E 0
Curium—244	1.0E 0

TABLE 2

RADIOACTIVITY OF IMPORTANT NUCLIDES RELEASED FROM THE CRITICALITY ACCIDENT IN THIS GUIDE (CI)

NUCLIDE	0 to 0.5 hr	0.5 to 8 hr	TOTAL
Kr-83m	3.7E 0	3.3E+1	3.7E+1
Kr-85m	1.6E+1	1.5E+2	1.7E+2
Kr-85	1.5E-4	1.4E-3	1.6E-3
Kr-87	1.0E+2	9.0E+2	1.0E+3
Kr-88	6.5E+1	5.9E+2	6.6E+2
Kr-89	4.1E+3	3.7E+4	4.1E+4
Xe-131m	3.8E-4	3.5E-3	3.9E-3
Xe-133m	5.5E-2	4.9E-1	5.5E-1
Xe-133	1.3E 0	1.2E+1	1.3E+1
Xe-135m	1.1E+1	9.9E+1	1.1E+2
Xe-135	1.6E+1	1.5E+2	1.7E+2
Xe-137	3.8E+3	3.5E+4	3.9E+4
Xe-138	1.2E+3	1.0E+4	1.1E+4
I-129	4.2E-11	3.9E-10	4.3E-10
I-131	1.8E-1	1.6E 0	1.8E 0
I-132	6.7E-1	6.1E 0	6.7E 0
I-133	3.5E 0	3.1E+1	3.5E+1
I-134	4.8E+1	[illegible]+2	4.8E+2
I-135	1.2E+1	1.0E+2	1.2E+2

TABLE 3

VALUES OF THE CLEARANCE PARAMETERS FOR THE TASK GROUP LUNG MODEL[a]

COMPARTMENT		CLASS D[b, c]		CLASS W[c]		CLASS Y[c]	
NP		T_k[d]	f_k[d]	T_k[d]	f_k[d]	T_k[d]	f_k[d]
	a	0.01	0.5	0.01	0.1	0.01	0.01
	b	0.01	0.5	0.4	0.9	0.4	0.99
TB	c	0.01	0.95	0.01	0.5	0.01	0.01
	d	0.2	0.05	0.2	0.5	0.2	0.99
P	e	0.5	0.8	50	0.15	500	0.05
	f	n.a.[e]	n.a.[e]	1.0	0.4	1.0	0.4
	g	n.a.[e]	n.a.[e]	50	0.4	500	0.4
	h	0.5	0.2	50	0.05	500	0.15
L	i	0.5	1.0	50	1.0	1000	0.9

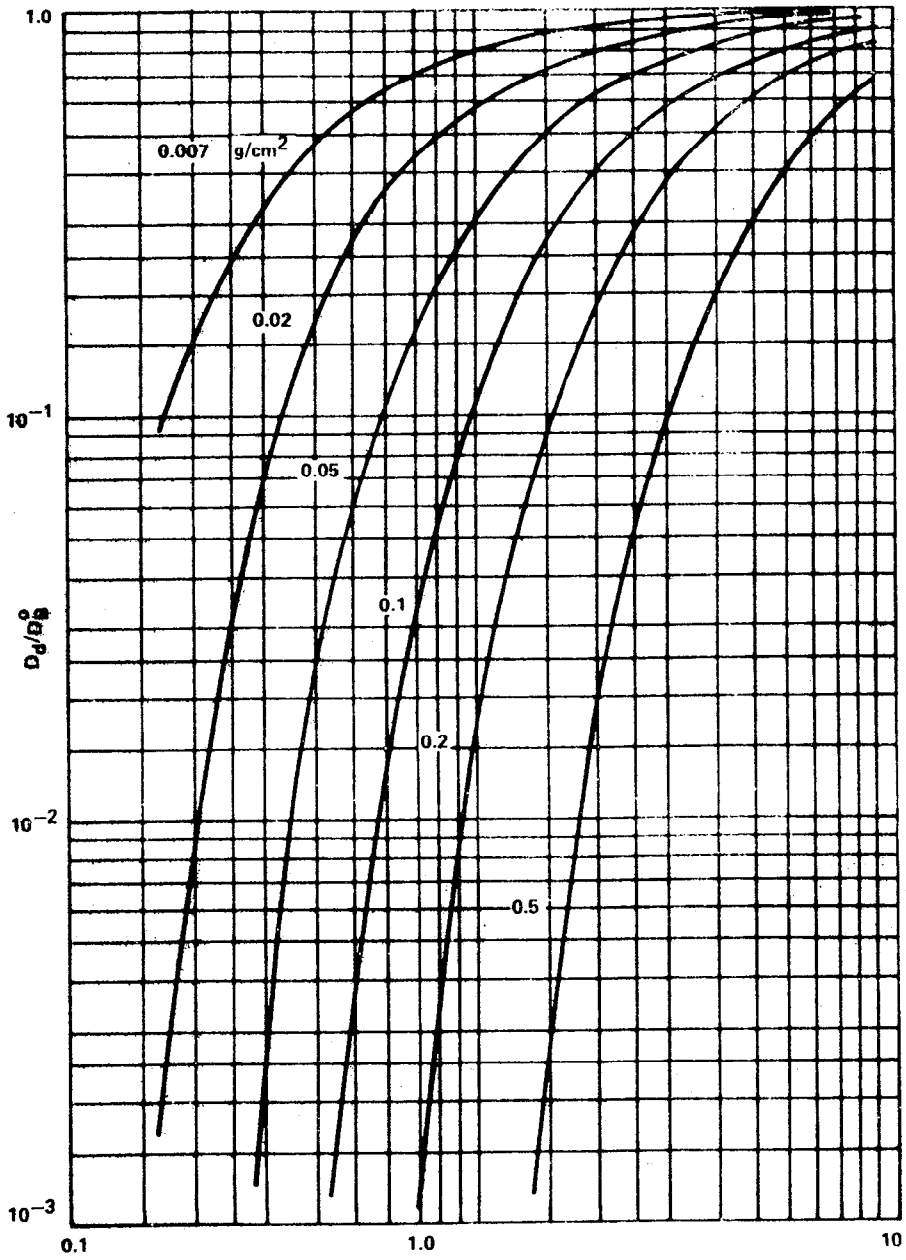
[a] See Figure 2 for the task group lung model (TGLM) schematic diagram.

[b] Data for soluble plutonium are included. To maintain dose conversion conservatism, this class should only be considered if justified on an individual case basis.

[c] Class D = readily soluble compounds where removal time is measured in days.
Class W = compounds with limited solubility where removal time is measured in weeks.
Class Y = insoluble compounds where removal time is measured in years.

[d] T_k is the biological removal half time in days; f_k is the fraction of original deposit leaving the organ via pathway indicated on the schematic model shown in Figure 2. Data are based on a mass median aerodynamic diameter of 1 micron and were developed by Battelle Memorial Institute, Pacific Northwest Laboratories, and presented in an interim report by E.C. Watson, J. R. Houston, and D. L. Strenge, April 1974

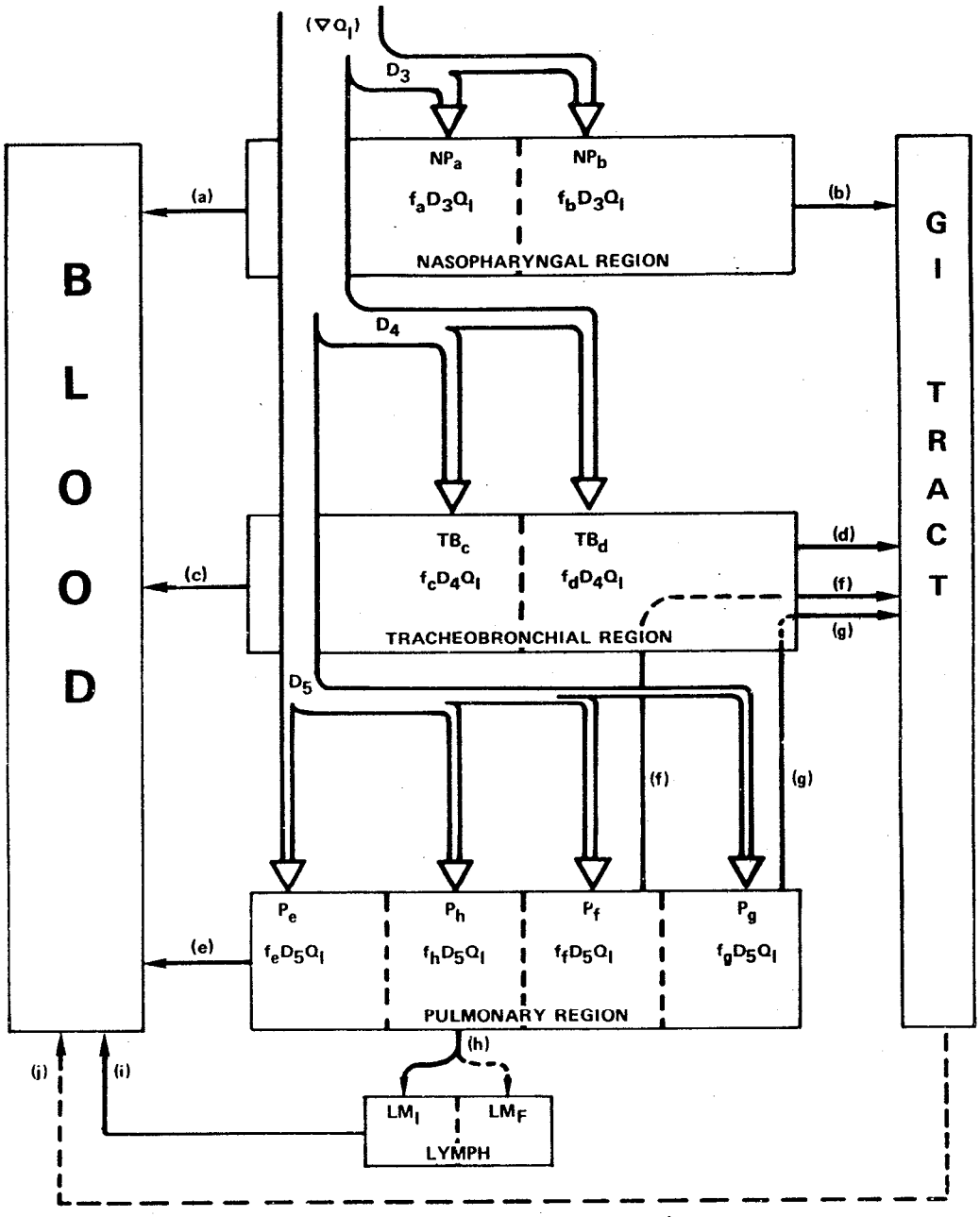
[e] n.a. means not applicable.



RATIO OF DEPTH DOSE TO SURFACE DOSE AS A FUNCTION BETA ENERGY SPECTRA[1] for Infinite Plane Source of Infinite Thickness and for Allowed Spectra

[1] Developed from Considerations Presented in Reference 17, Chapter 16

FIGURE 1



SCHEMATIC DIAGRAM DEVELOPED FROM ICRP TASK GROUP LUNG MODEL (Ref. 19)

FIGURE 2

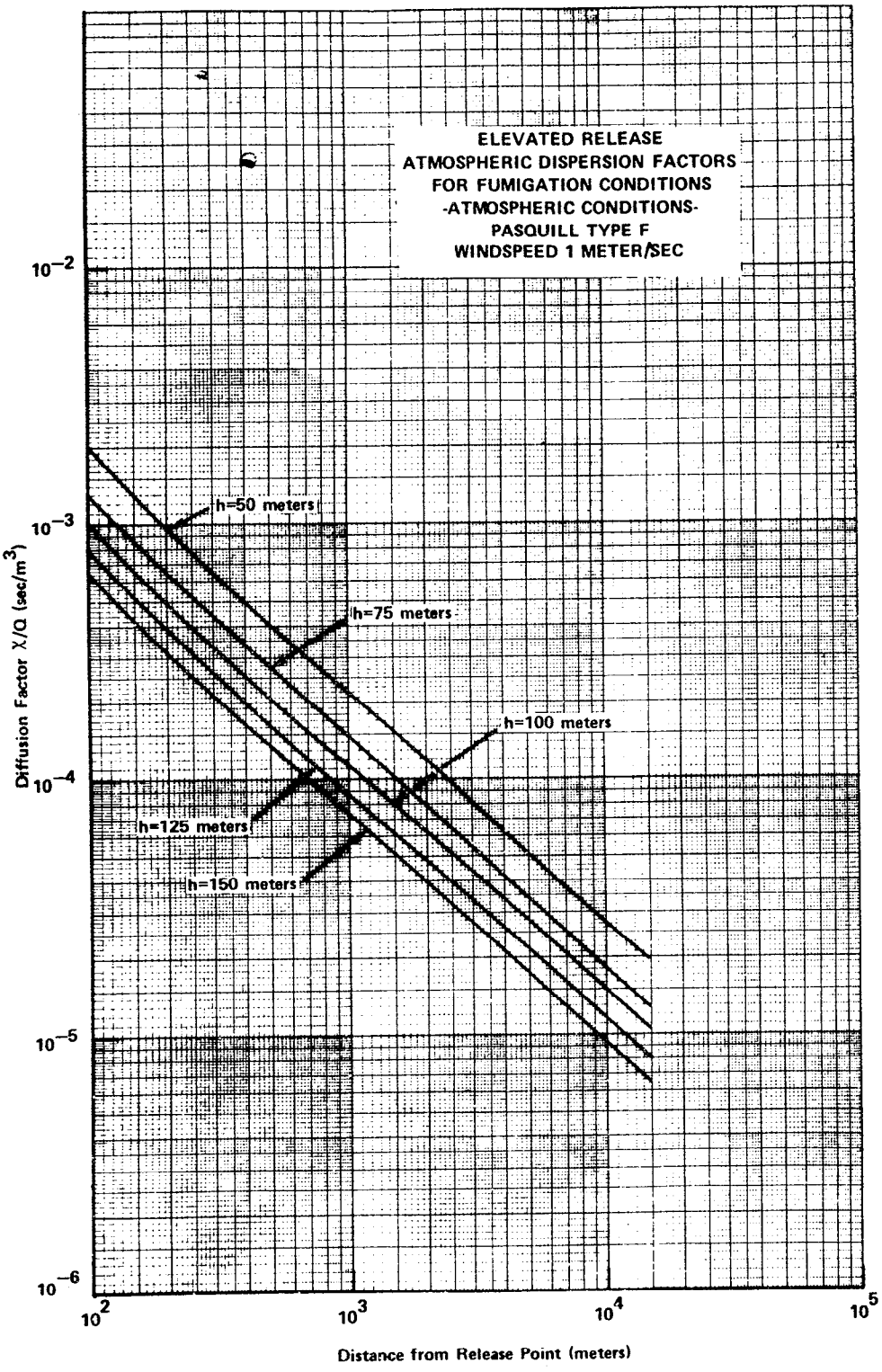
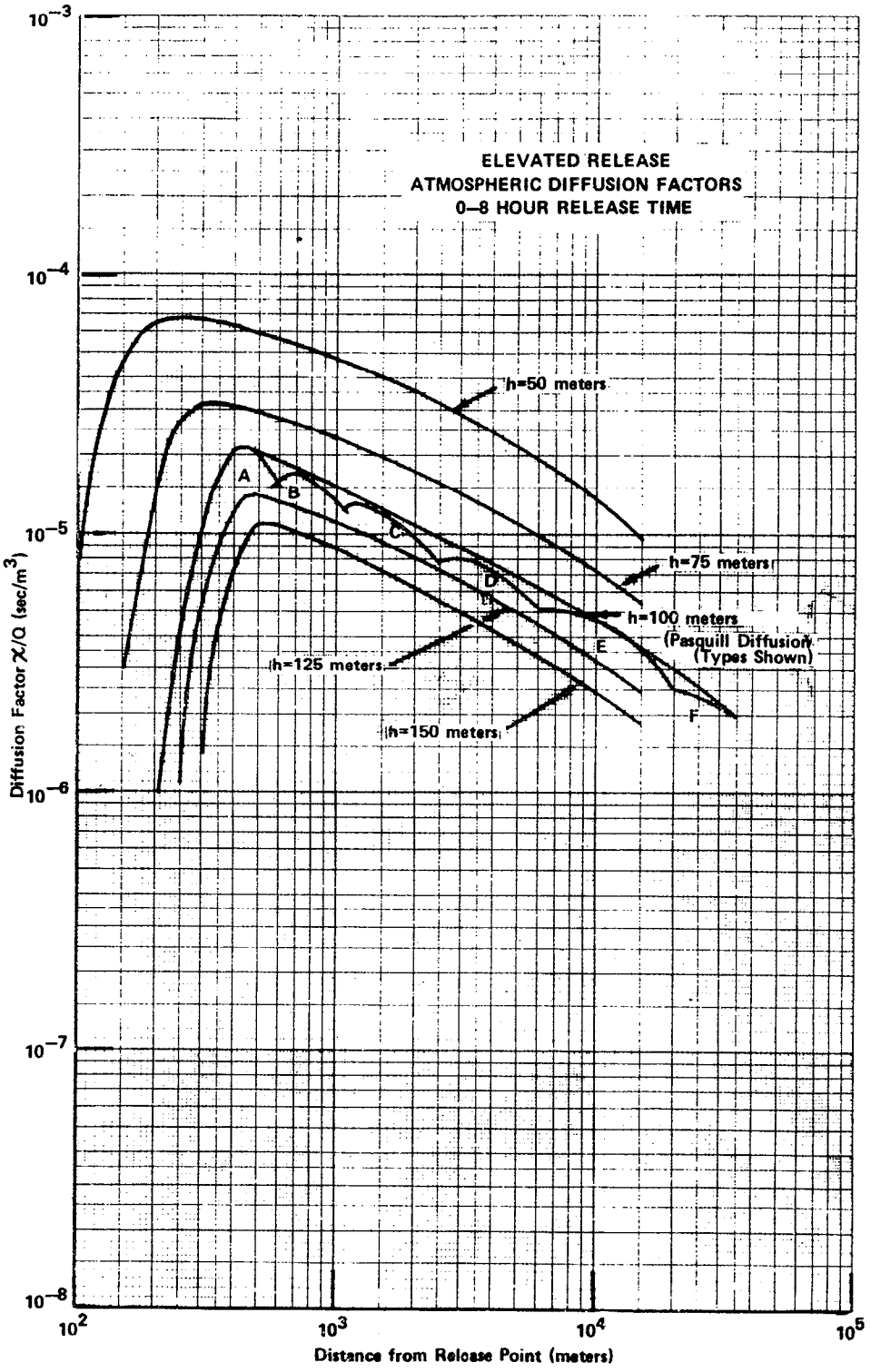


FIGURE 3(Ref. 22)



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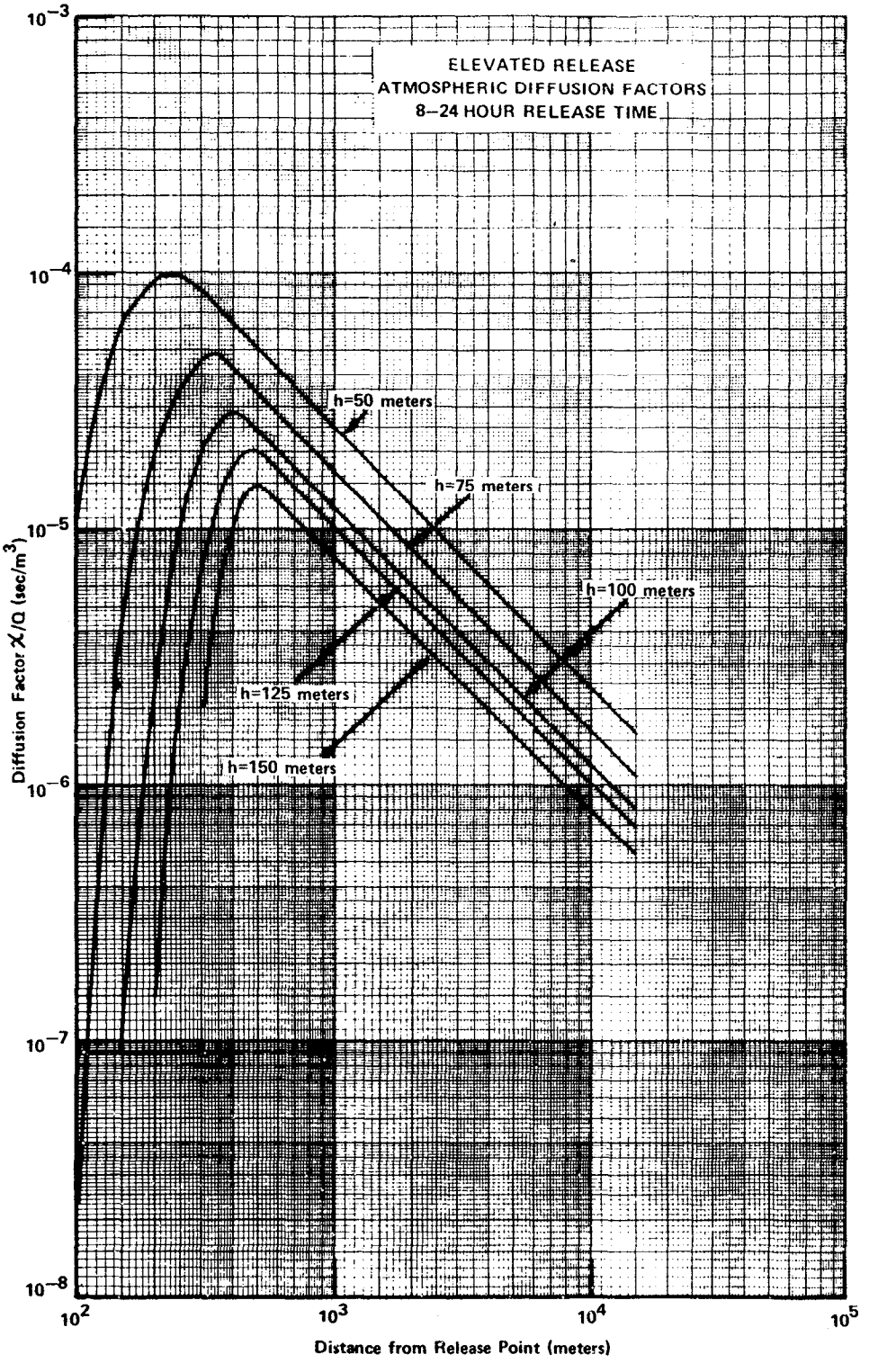

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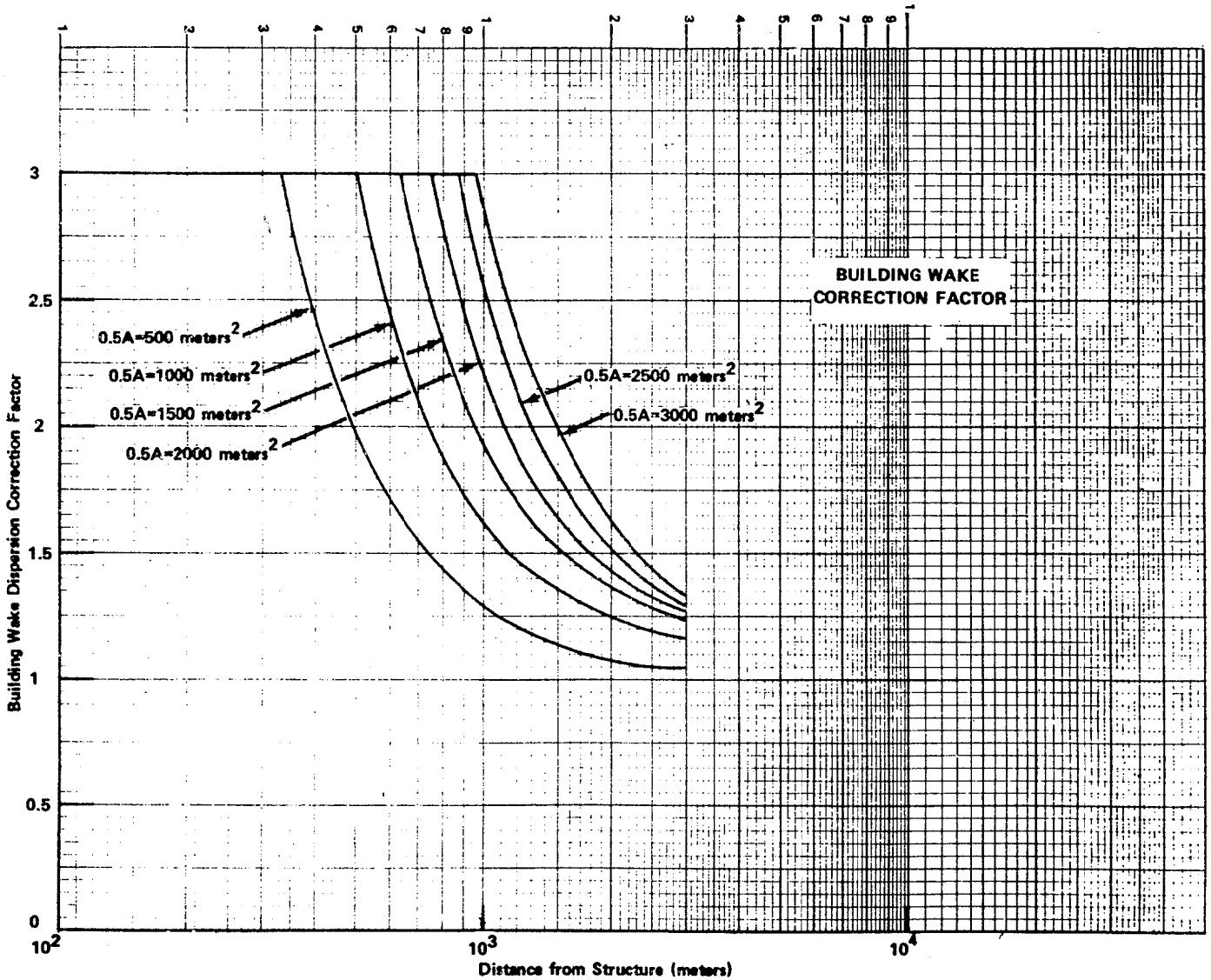

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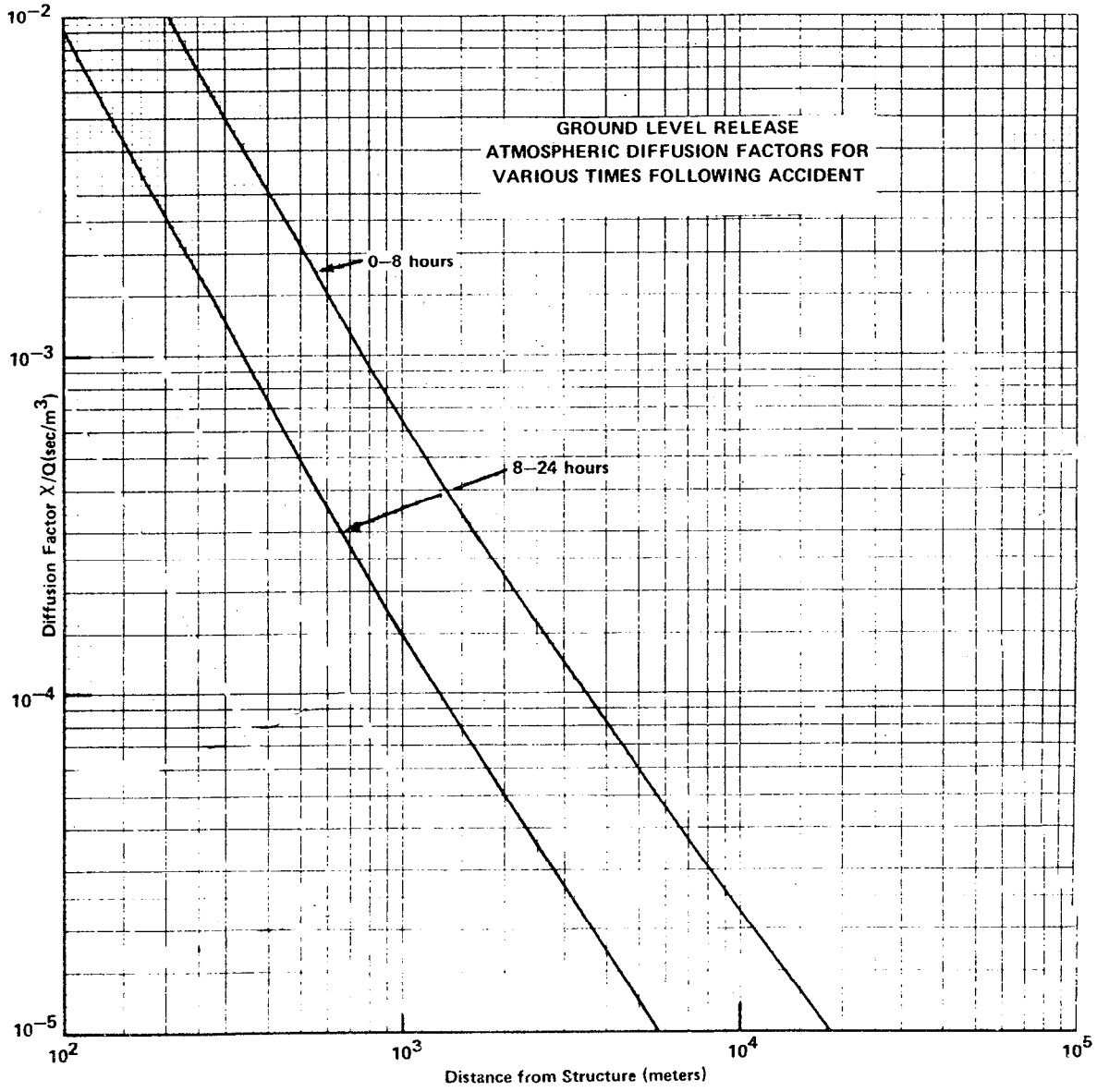

FIGURE 7A(Ref. 22)

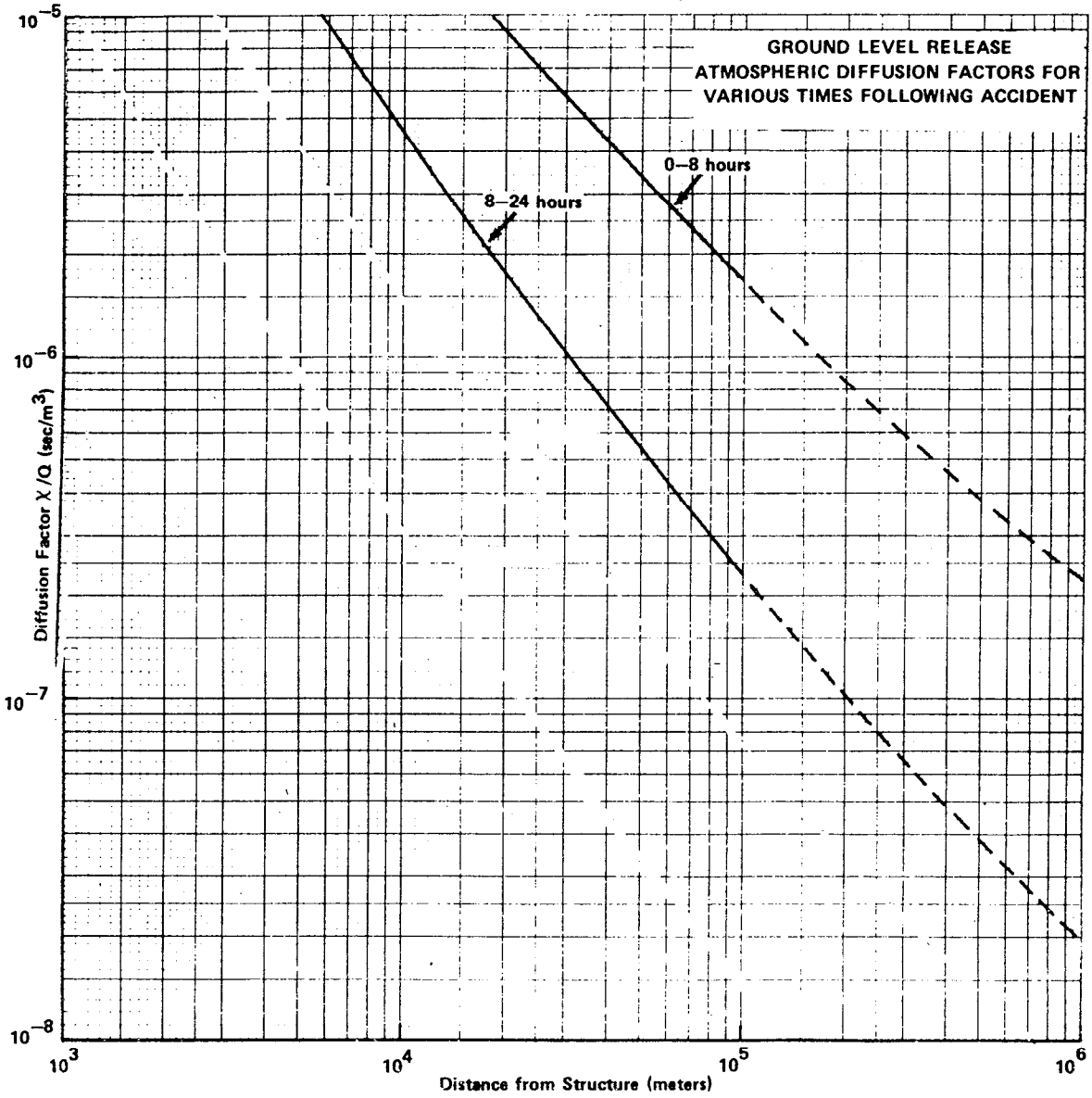

FIGURE 7B(Ref. 22)