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An Updated Nuclear Criticality Slide Rule

Technical Basis

Prepared by
B. L. Broadhead, C. M. Hopper, R. L. Childs, J. S. Tang

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

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1 INTRODUCTION

Emergency planning and preparedness are essential elements in any nuclear facility safety program. U.S. Nuclear Regulatory Commission (NRC)-licensed nonreactor nuclear facilities that process fissionable materials have the potential for a broad variety of nuclear criticality accidents requiring emergency response evaluations. Volume 1 of this report describes the development of a hand-held "slide-rule" emergency response tool for rapid, mobile, in-field radiation dose, dose rate, and shielding evaluations having relevance to five different types of nonreactor nuclear facility criticality accidents. In Volume 2, via a set of nomographs, the slide rule permits the estimation of unknown parameters based upon known parameters presented on the slide rule. This capability permits continued updating of information during the evolution of emergency response, including exposure information about "accident victims," estimates of potential exposures to emergency response re-entry personnel, estimates of future radiation field magnitudes, and fission yield estimates. Fission neutron and fission product gamma shielding factors are also provided for common materials of construction. The work also provides recommended input to the NRC Response Technical Manual (RTM) on a decision-tree process to predict fission yields of potential accidents involving homogeneous solutions.

In January 1974, a limited distribution, informal report, Y-DD-145,[1] "A Slide Rule for Estimating Nuclear Criticality Information," was authored by C. M. Hopper for the Oak Ridge Y-12 Plant as a tool for emergency response to nuclear criticality accidents. This original slide rule permitted the estimation of accident yields, anticipated radiation exposures to rescue personnel, radiation shielding, and anticipated radiation fields to be encountered during an emergency response. Several shortcomings of the slide rule were evident:

1. a limited foundation based upon data (from experimentally simulated accidents) which was extended by the use of first-order, one-dimensional (1-D) mathematical relationships and shield/decay data derived from handbooks;

2. only two types of nuclear criticality accidents were considered, a bare high-enriched uranium metal system and a high-enriched uranyl nitrate solution system; and

3. the use of the inverse squared distance rule for reduction of radiation fields by distance, neglecting air attenuation, secondary radiation production ("skyshine") and the air-ground radiation interface.

Subsequent work by Wilkinson et al.[2] addressed the second shortcoming and the 1-D effects of the third shortcoming (e.g., air attenuation and secondary radiation production).

Mixed units (i.e., English and metric) are used throughout this report and in the graphic presentations of the slide rule to accommodate historic and typical use in the U.S. industry. Historically, nonreactor nuclear facilities were built to English unit specifications (e.g., 50,000-gallon tank, 16-in.-diam. pipes/tubes, 2 gallons/minute pump capacity, etc.) whereas operating process specifications have evolved to metric units (e.g., grams of U or grams ^{235}U per liter of solution, kg U, grams of U per cubic centimeter, etc.). The unit of typical use is presented in the text which is then followed by an alternative unit in parentheses. The intent of providing mixed units is to ease data conversion and manipulation during a potentially stressful period of emergency response when data exchange is provided in mixed units.

2 APPROACH

Since the goal of this work was to update previous studies, careful consideration was given to ensure the new results were a logical extension of the previous work. Five systems were selected, two of which were nearly identical to the two original slide-rule systems. For three of the five systems, experimental data of some fashion exist and were used to verify that the slide-rule results were comparable to the existing measurement database. A flowchart that characterizes the basic steps in the generation of the original and this updated slide-rule tool is shown in Fig. 1. The interplay of the various analysis phases, prompt dose vs distance, fission-product gamma dose vs distance and time, total dose vs distance and time, and 1-m integral dose vs distance and time are noted. In this work, the initial analyses consisted of a scoping study using 1-D discrete-ordinates methods for static studies and point-depletion/decay methods combined with 1-D discrete-ordinates methods for time-dependent studies. Comparison of these scoping slide rules to the original work allows for a determination of the validity of assumptions in the published slide-rule information. Shortcomings in 1-D methods are evident by comparison to the available measured data.

Extension of the 1-D scoping studies using two-dimensional (2-D) methods allows for estimation of 2-D effects. The influence of the air/ground interface should be evident, as well as the possible contributions due to radiation skyshine. The inclusion of skyshine-only information is also included to allow determination of accident characteristics where substantial shielding is present between the postulated accident and the desired location of radiation hazard information. The inclusion of this revised information into a hand-held format very similar to the original slide rule with much wider distribution should enhance the tool's viability and function greatly.

In a similar format to the nuclear criticality slide rule, this work has developed a simplified method of obtaining prompt and total fission yield information. The prompt fission yield estimation procedure is based on simplified relationships developed by Hansen.[3] The relationships correlate the reactivity insertion rate, system volume, neutron lifetime, intrinsic source, and incremental reactivity worths for fission yields for a fairly large range of potential solution criticality scenarios. Also, a total fission yield estimation equation by Barbry[4] is provided for water-moderated criticality scenarios.

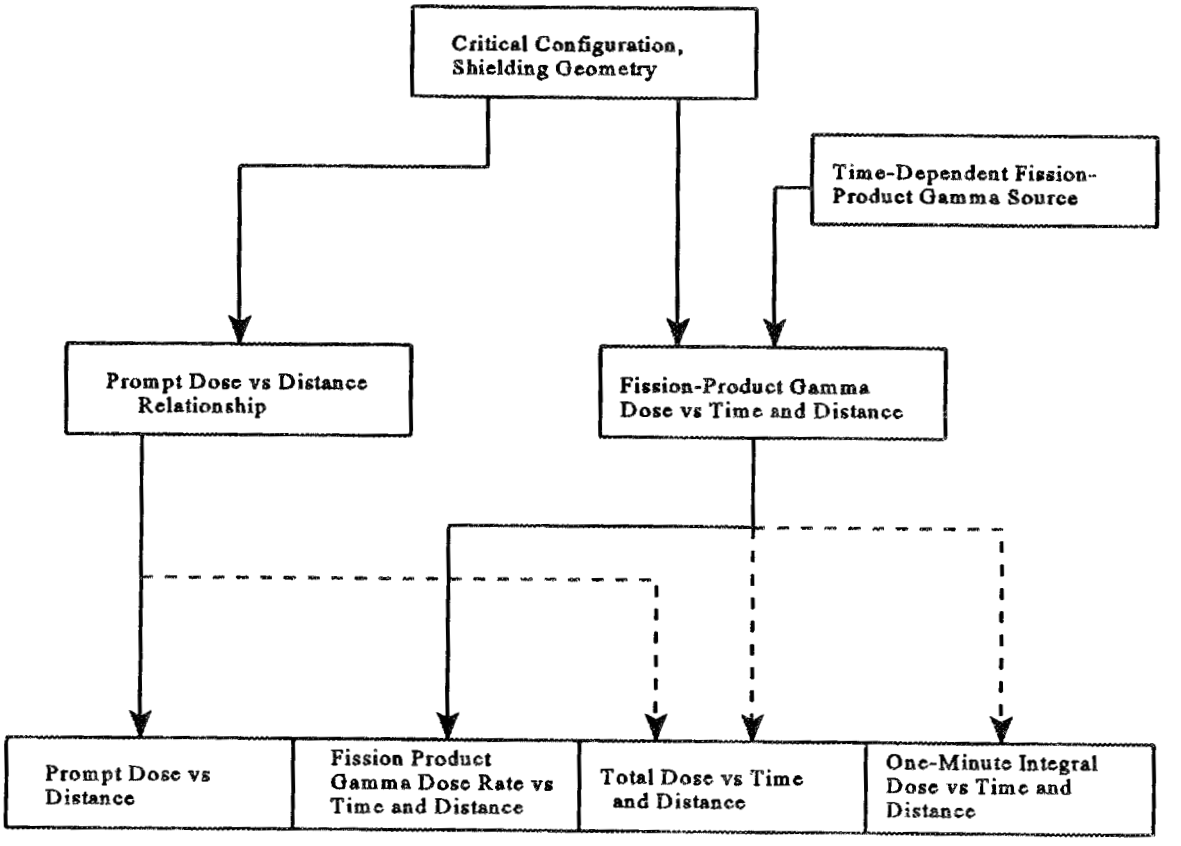


Figure 1 Slide-rule generation flowchart

3 SELECTION OF CRITICAL SYSTEMS

The previous slide rule was applicable only to a single, bare uranium metal unit or a solution of highly enriched uranyl nitrate; thus utilization was limited to highly enriched uranium (HEU). To broaden the range of application, other system types were needed. The selection of additional systems considered the fissile material form, the ^{235}U enrichment, and the plausible system condition in which criticality might occur (e.g., addition of moderator material). The systems selected were as follows:

1. unreflected sphere of 4.95 wt % enriched aqueous uranyl fluoride, $U(4.95)O_2F_2 \cdot H_2O$, solution having a hydrogen-to-^{235}U ratio of 410 (solution density = 2.16 g/cm^3),

2. unreflected sphere of damp 5 wt % enriched uranium dioxide, $U(5)O_2$, having a hydrogen-to-^{235}U ratio of 200,

3. unreflected sphere of 93.2 wt % enriched uranyl nitrate, $U(93.2)O_2(NO_3)_2 \cdot 6H2O$, solution having a hydrogen-to-^{235}U atom ratio of 500 (solution density = 1.075 g/cm^3),

4. unreflected sphere of 93.2 wt % enriched uranium metal sphere (metal density = 18.85 g/cm^3), and

5. unreflected sphere of damp 93.2 wt % enriched uranium oxide, U_3O_8 plus water, having a hydrogen-to-^{235}U atom ratio of 10 (uranium oxide density = 4.15 g/cm^3).

These critical systems should approximately model a wide range of fissile material operations. The generic situations include high- and low-enriched process solution operations (#3 and #1), high- and low-enriched powder operations (#5 and #2), and high-enriched metal operations (#4).

4 DEVELOPMENT OF PROTOTYPE NOMOGRAPHS

Initial computational activities for this work included generation of tabular results to facilitate the production of a slide-rule nomograph using 1-D discrete-ordinates methods. These tabular data were produced as a means of assessing the approximations in the original slide rule and development of a prototype tool for the final slide-rule product in this study. Comparison of the 1-D slide-rule data with that of the original slide rule and available measurement data highlighted shortcomings in the previous slide-rule tools. Additional 2-D investigations in this work were also compared to the original slide rule and measurement data. The generation of a prototype slide rule using 1-D tabulations allowed for production and testing of an automated procedure used in assembly of the final slide rule.

The development of a 1-D prototype slide rule is an extension of a previous study by Wilkinson et al.[2] That work utilized the same tools and procedures as this work; however, the end result was a renormalization of the calculated results to the measured data. This work separately analyzes the 1- and 2-D results and measured data for effects/differences, and then produces a final slide rule based on 2-D automated procedures.

4.1 SAS1X DESCRIPTION

The 1-D tool used in this work is the SAS1X module as implemented in the SCALE system.[5,6] The SAS1X procedure executes BONAMI-S, NITAWL-II, and XSDRNPM-S to process cross sections for subsequent shielding calculations by XSDRNPM-S. BONAMI-S and NITAWL-II perform cross-section resonance processing using the Bondarenko and Nordheim methods,[6] respectively. XSDRNPM-S is a general-purpose, 1-D, discrete-ordinates code which solves the radiation transport equation. The first XSDRNPM calculation initiated by SAS1X was originally intended to produce only cell-weighted cross sections for use via the XSDRNPM-S shielding calculation. The current version of SAS1X has been modified to store the leakage spectrum from the first XSDRNPM-S step in an angular flux file. The SAS1X sequence then reads the angular flux file written by the first XSDRNPM-S criticality calculation and prepares a boundary source for input to subsequent XSDRNPM-S shielding calculations. This boundary source is input to the boundary of the first zone of the shielding problem. This first zone must be designated as a void region, and its boundary dimension must be identical to the outer boundary of the XSDRNPM-S criticality calculation.

4.2 ONE-DIMENSIONAL MODEL DESCRIPTION

Spherical models for critical systems containing the five materials described above were generated by performing an XSDRNPM critical radius search on each system. These systems are not meant to model exactly any specific system, but to be representative of systems with similar characteristics. The resulting critical system parameters for each of the five spherical configurations are given in Table 1. The details of these configurations are given primarily for completeness. The primary use for these data is the development of typical leakage characteristics from these representative critical systems, which should be somewhat insensitive to the detailed system descriptions.

In Table 1, the assumed composition of air is also given. These values roughly correspond to assumed 76% and 24% weight contributions for nitrogen and oxygen at a density of 1.23 g/L (0.164 ounces/gal). These densities of air were used in the 1-D slide-rule calculations to generate the air attenuation of particles leaking from the critical systems. The 1-D shielding calculational models consisted of essentially the boundary source determined from the critical system leakage, followed by an air sphere with radius of up to 1920 m (6299 ft). The neglect of water vapor in the air is not expected to have an appreciable effect on the reported results. While effects due to water vapor are anticipated beyond 1500 m (4921 ft), the results are only reported to about 1300 m (4265 ft), the remaining air is included only

Table 1 Critical system parameters[a]

Parameter	Uranyl fluoride (4.95%)	Damp UO_2 (5%)	Uranyl nitrate solution (93.2%)	U metal[c] (93.2%)	Damp U_3O_8 (93.2%)
Number density[b]					
U-235	1.3173-4[d]	2.6060-4	1.3154-4	4.5012-2	6.4361-3
U-238	2.5342-3	4.9592-3	9.6010-6	2.6704-3	4.6956-4
N	–	–	2.8205-4	–	–
O	3.1989-2	3.6544-2	3.4012-2	–	5.0641-2
F	5.3345-3	–	–	–	–
H	5.3314-2	5.2203-2	6.5769-2	–	6.4460-2
Spherical radius (cm)	25.5476	23.2133	18.9435	8.6518	11.8841
H/X	410	200	500	0	10

[a]Air number densities are N (4.00-5), O (1.11-5).
[b]Units of atom/barn-cm.
[c]For the metal system, the following material number densities were also used: U-234 (4.8503-4) and U-236 (9.6182-5).
[d]Read as 1.3173E-4.

to correctly estimate the backscattering component at 1300 m (4265 ft). The calculated fluxes were converted to dose rates using the Henderson flux-to-dose-rate factors.[7] The cross-section library used throughout this work was the SCALE 27N-18COUPLE set. Comparison of the results using this library with other fine-group sets yielded good results.

4.3 METHODS OF SLIDE-RULE DATA GENERATION

It was recognized in the initial planning of this project that a phased approach to the updating of the original slide rule would be the most beneficial path to pursue. This phased approach would develop a prototype slide-rule tool using 1-D methods, which would then aid in the development of the final slide rule based on multidimensional methods. This section describes the generic steps that were needed in the development of both the prototype and final slide rules. The 1- and 2-D methods used to generate the prompt dose vs distance relationships in this work (see Fig. 1) are straightforward and should require no further explanation other than those given in the above description. The results from these prompt dose calculations are used directly in the first set of slide-rule plots as indicated in Fig. 1. The portions requiring further explanation include the details of the time-dependent fission-product gamma-ray source generation and the methods utilized to obtain relationships for 1-min integral doses vs distance and time.

The time-dependent sources were generated with the SCALE[6] module ORIGEN-S which uses point-depletion methods to solve for the radioisotopic compositions after arbitrary irradiation/decay periods. This work assumes an instantaneous event and then tabulates the expected dose rates for periods of 1, 5, and 10 s and 1, 5, 10, 50, 100, 500, and 1000 min after the event for all five critical systems. Specifically, ORIGEN-S first estimates the production of

radioisotopes instantaneously during the event, decays the concentrations to the time points given above, and then computes the gamma-ray radiation source based on discrete gamma-ray line data. These sources are then utilized in 1-D discrete-ordinates calculations using identical methods, geometry, and codes as those of the prompt dose calculations. This time-dependent, fission-product, gamma-ray dose rate information is used directly in the second set of slide-rule plots as indicated in Fig. 1.

The remaining two slide-rule plots require processing the above-described prompt and time-dependent dose information. The total dose vs time and distance consists of the prompt dose as a function of distance plus the fission-product gamma-ray dose rate information integrated from time zero until the tabulated time of 1 s to 1000 min. The final slide-rule plot is the 1-min time-integrated dose due to fission-product gamma rays at each of the same time points from 1 s to 1000 min. For both of these plots, the integration is performed analytically with an equation of the form

$$y = ax^b ,$$

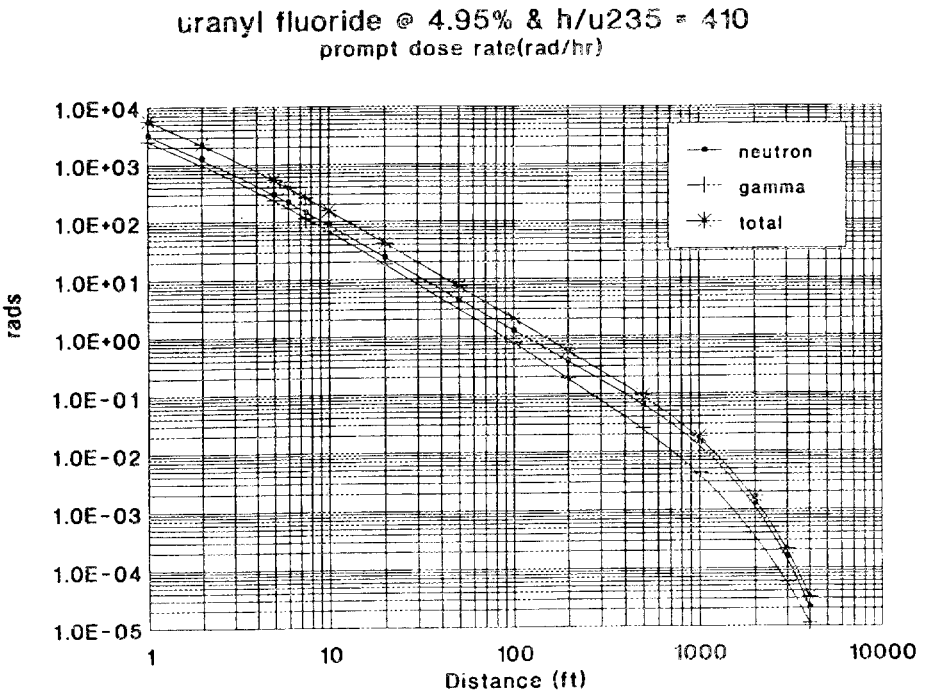
where a and b are obtained by fitting the time-dependent doses.

There are slight differences in this approach as compared with that of the previous study.[2] The previous work assumed a variable accident duration (0.0001 s for the metal system, 1 s for the damp U_3O_8, 0.5 s for the uranyl nitrate solution, and 20 min for the uranyl fluoride systems) as compared to a fixed instantaneous duration in this study. It is not anticipated that variations in the accident duration of 1 s or less would have an appreciable effect on the slide-rule results. The results for a 20-min accident duration would be more appropriate for the dose rates after a multiple burst accident. The results of this study should be more appropriate for the time after a short initial burst. The time-integration procedures used in the two studies were also different, but both should yield reasonably consistent results.

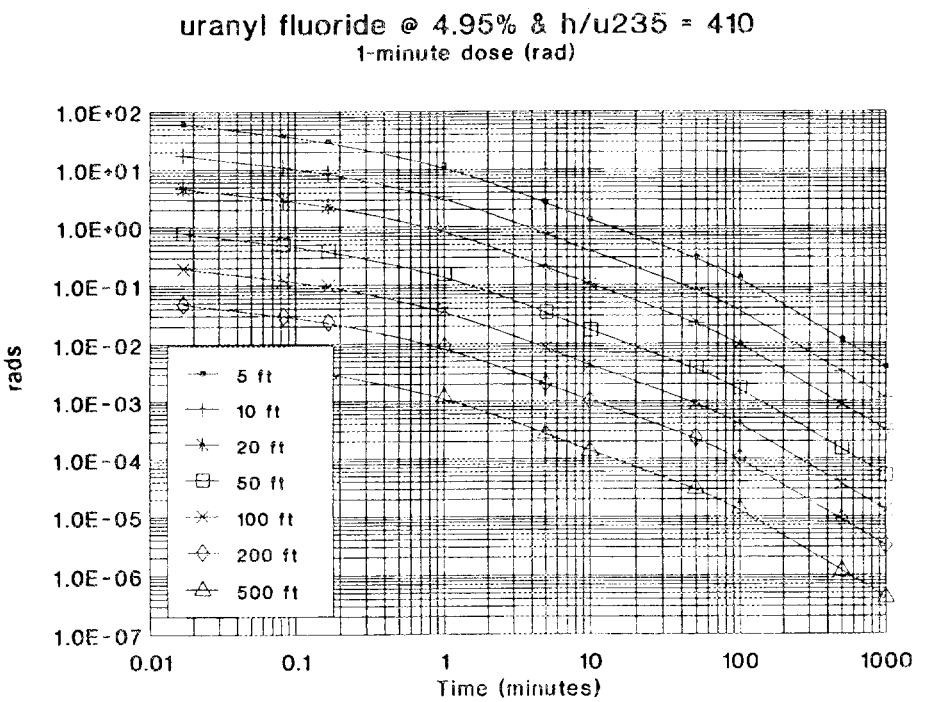
4.4 ONE-DIMENSIONAL SLIDE-RULE RESULTS

The tabular data generated as described above are shown graphically in Figs. 2 through 6 for 4.95 wt % enriched uranyl fluoride, $U(4.95)O_2F_2$; damp 5 wt % enriched uranium dioxide, $U(5)O_2$; 93.2 wt % enriched uranyl nitrate, $U(93.2)O_2(NO_3)_2$; 93.2 wt % enriched uranium metal; and damp 93.2 wt % enriched uranium oxide, U_3O_8. Each figure contains four parts: (a) prompt dose vs distance from criticality event, (b) delayed dose rate vs distance and time, (c) total dose vs distance and time, and (d) 1-min integrated dose vs distance and time. These graphs correspond to a normalized yield of 10^{17} fissions. First-pulse yields are highly variable, depending upon the system under consideration and have been observed in accidents and experiments to range from about 10^{15} fissions to not much more than 10^{18} fissions.

Comparison of the part (a) plots for all five systems shows the enhanced neutron leakage for low moderation systems and enhanced gamma-ray leakage for highly moderated systems. For the metal system, the neutrons dominate the leakage, while somewhere between a hydrogen-to-^{235}U ratio (H/X) of 400 to 500, the gamma-ray dose contribution becomes larger than the neutron dose contribution for distances less than 100 ft (30.48 m). In essence, for systems of increasing H/X, the neutron leakage decreases due to water moderation and capture, while the gamma-ray leakage increases due to the diminished attenuation in water relative to metal and the enhanced secondary gamma production from neutron capture. The overall dose-vs-distance shapes are very similar to each other as well as the previous results given in Ref. 2 and are characterized by an inverse-square portion followed by an air attenuation portion. Also seen for all cases is the slight difference in slope between the neutron and gamma results. The neutron attenuation

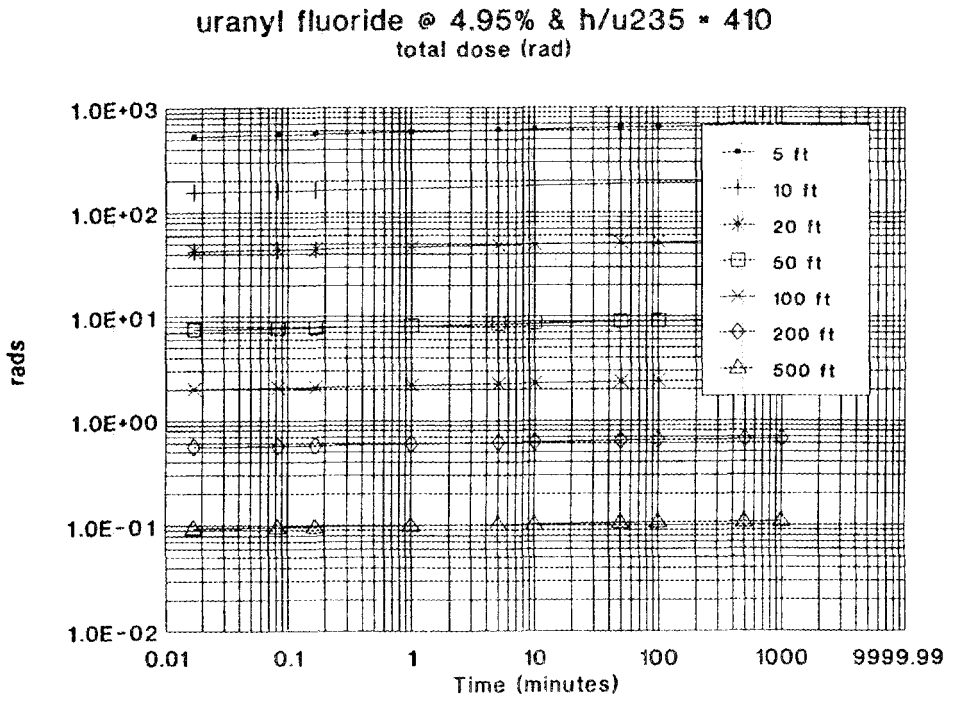


(a) estimated prompt doses (rads) based on total fission yield and distance from event

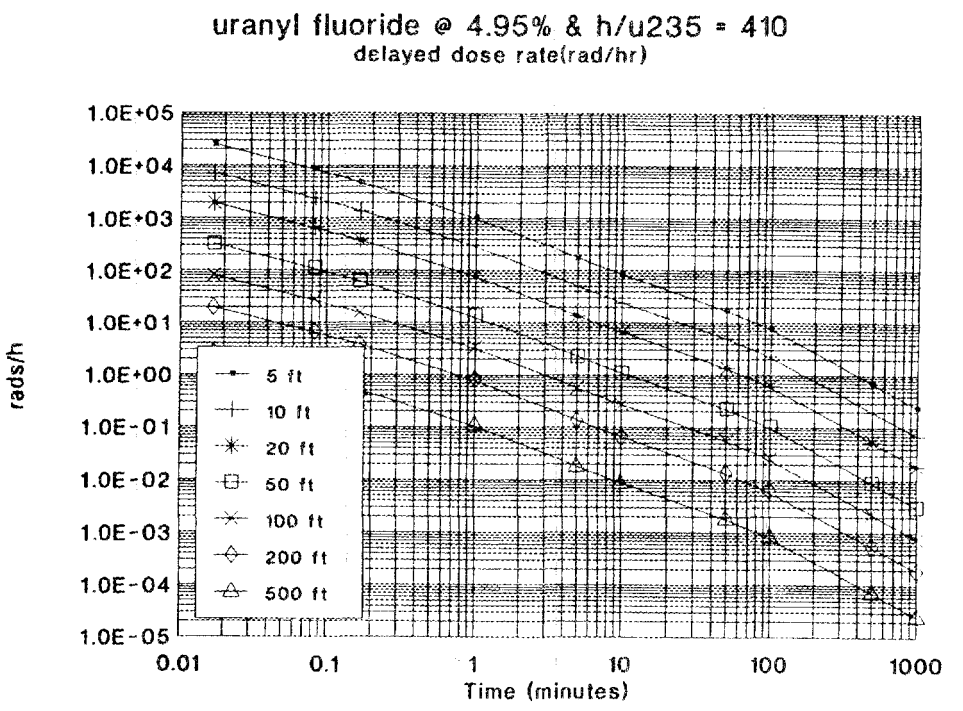


(b) accumulated one-minute dose (rads) based on estimated fission yield, distance from incident, and time of energy after accident

Figure 2 Uranyl fluoride (4.95%) 1-D slide rule (normalized to 10^{17} fissions)

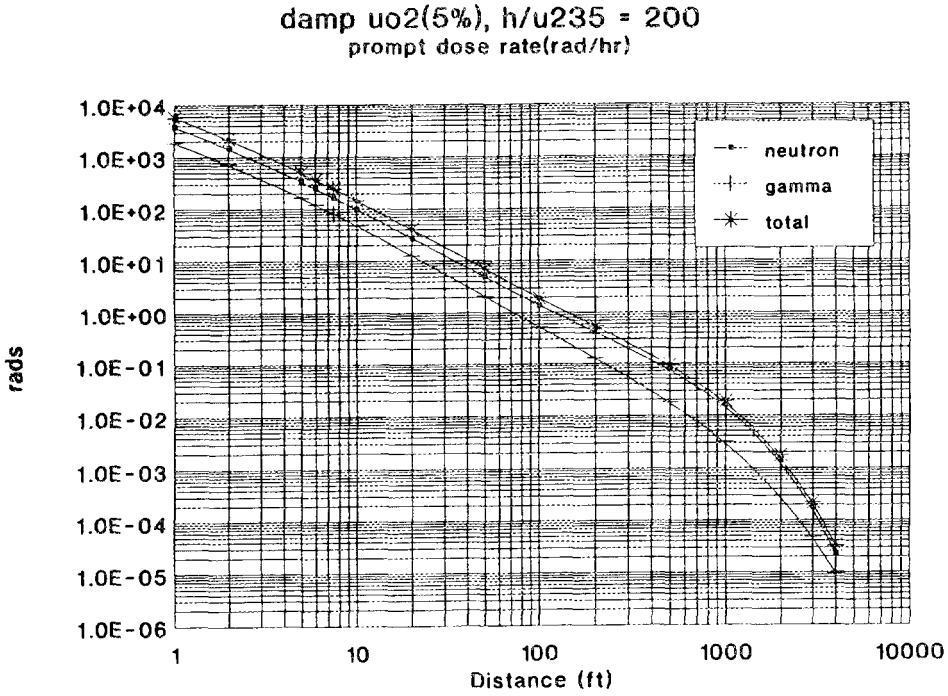


(c) integrated total dose (rads) based on estimated fission yield, distance from incident, and elapsed time after accident

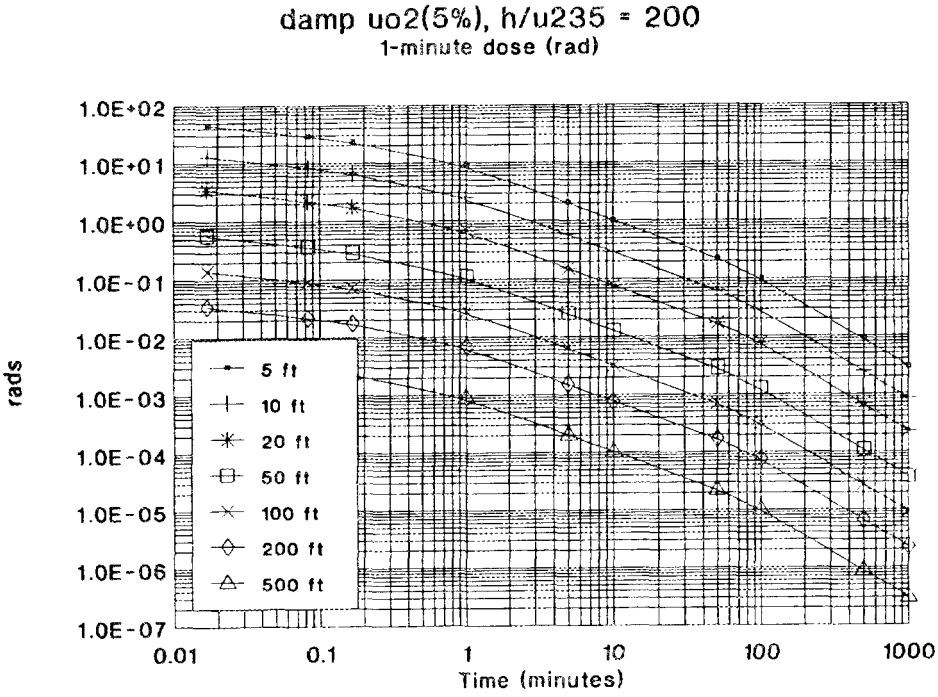


(d) estimated fission yield based on distant gamma dose rate (rads/h) and elapsed time

Figure 2 (continued)

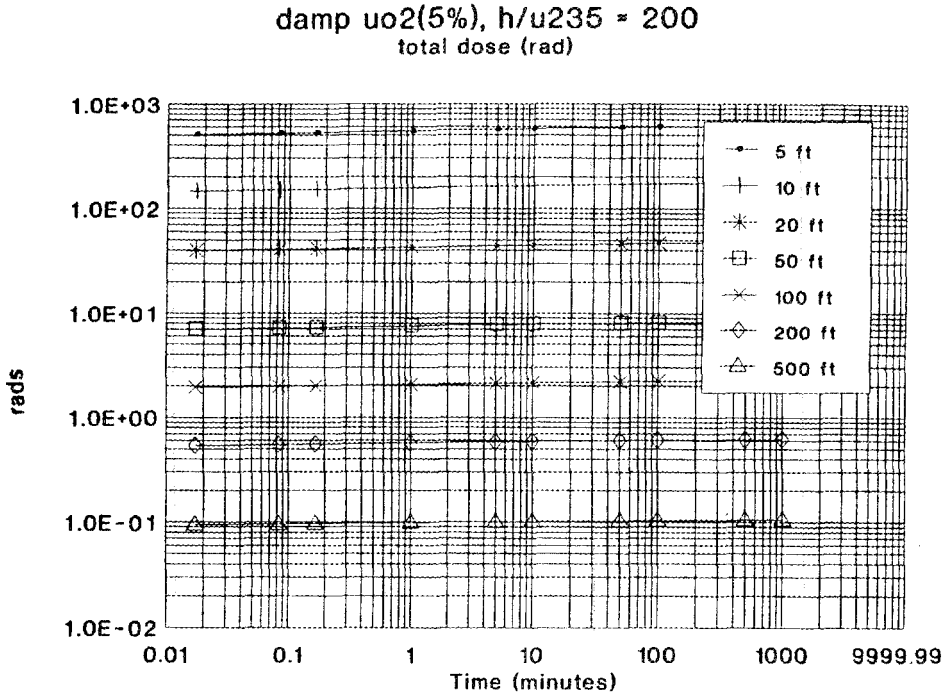


(a) estimated prompt doses (rads) based on total fission yield and distance from event

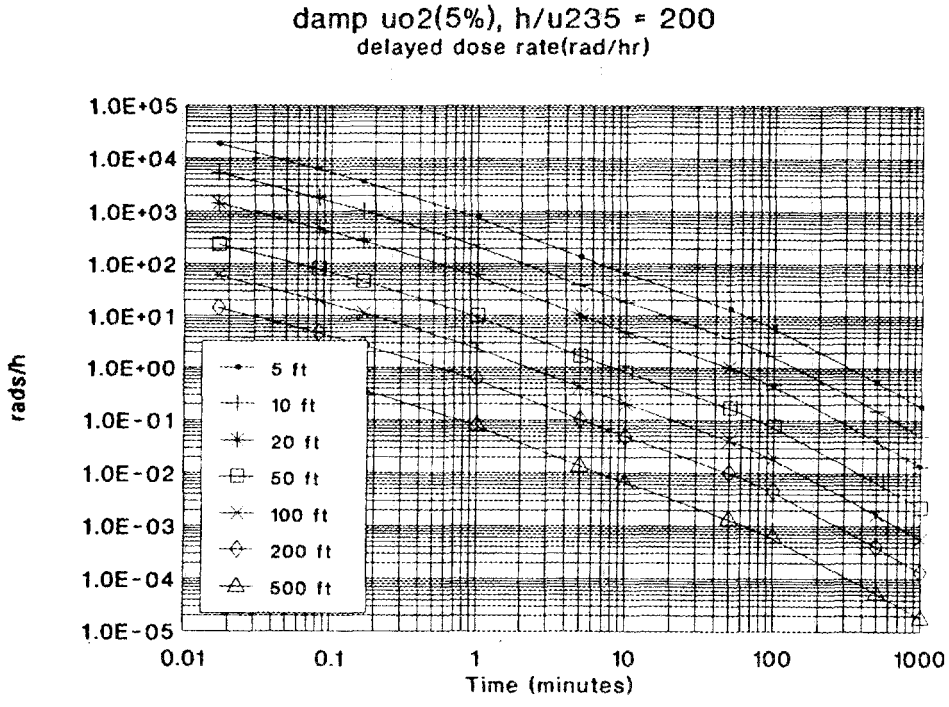


(b) accumulated one-minute dose (rads) based on estimated fission yield, distance from incident and time of entry after accident

Figure 3 Damp UO_2 (5%) 1-D slide rule (normalized to 10^{17} fissions)

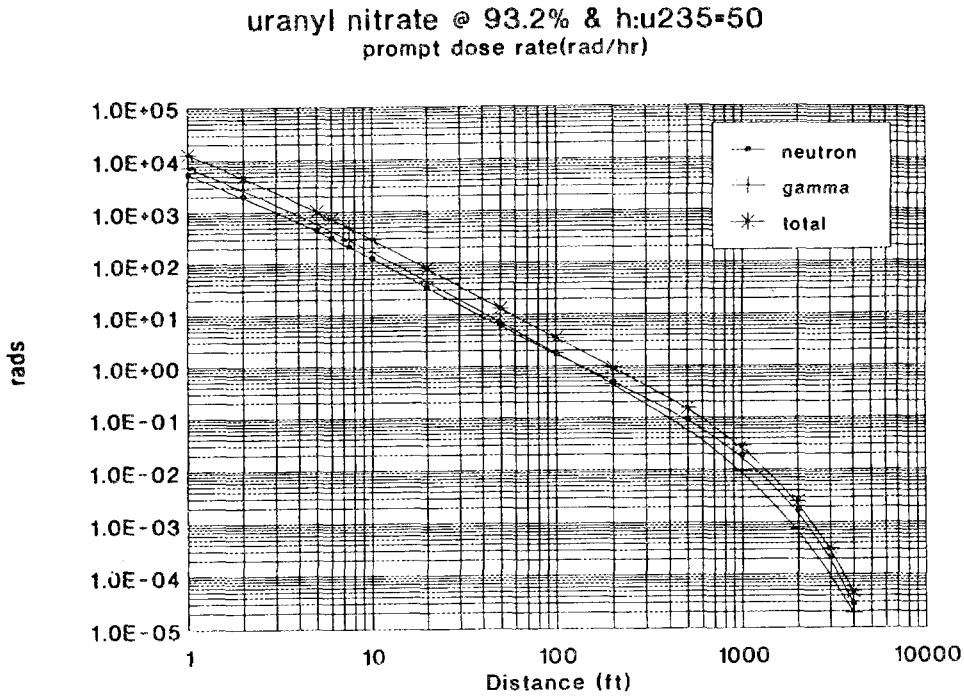


(c) integrated total dose (rads) based on estimated fission yield, distance from incident and elapsed time after accident

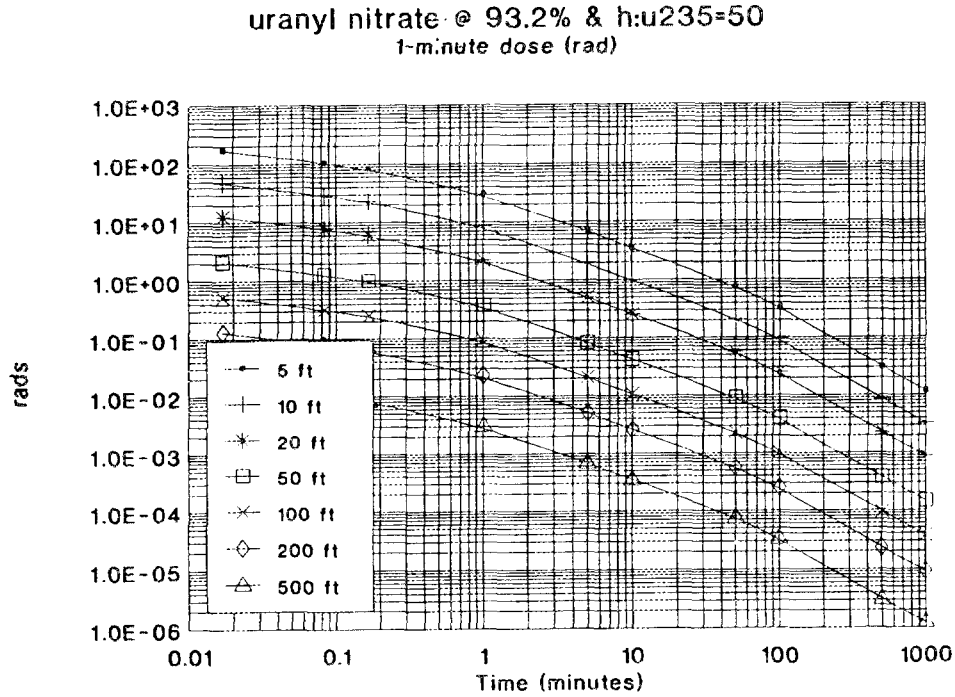


(d) estimated fission yield based on distant gamma dose rate (rads/h) and elapsed time

Figure 3 (continued)

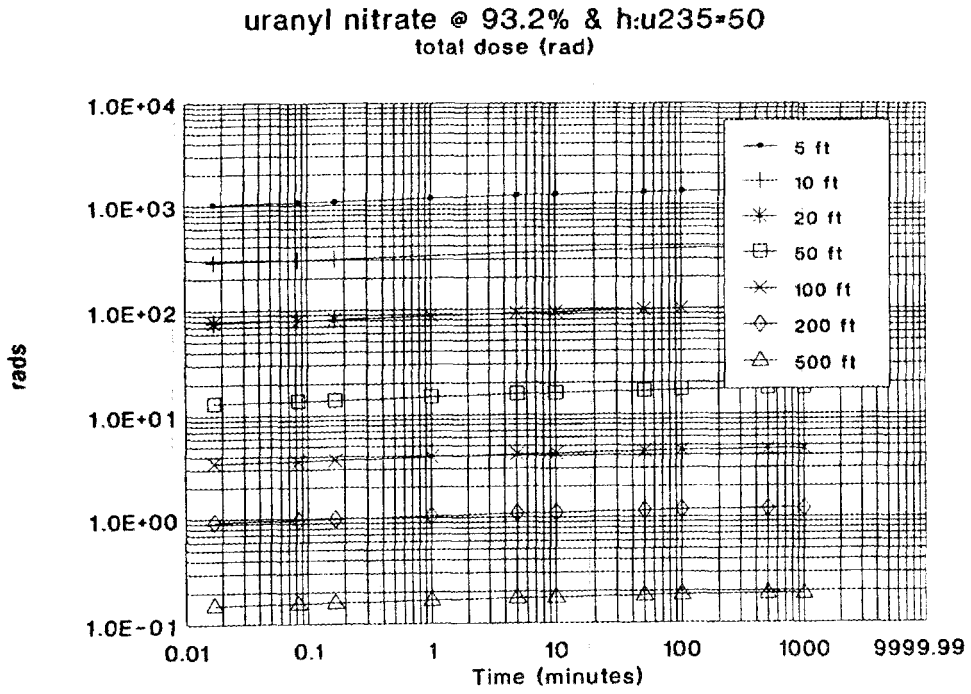


(a) estimated prompt doses (rads) based on total fission yield and distance from event

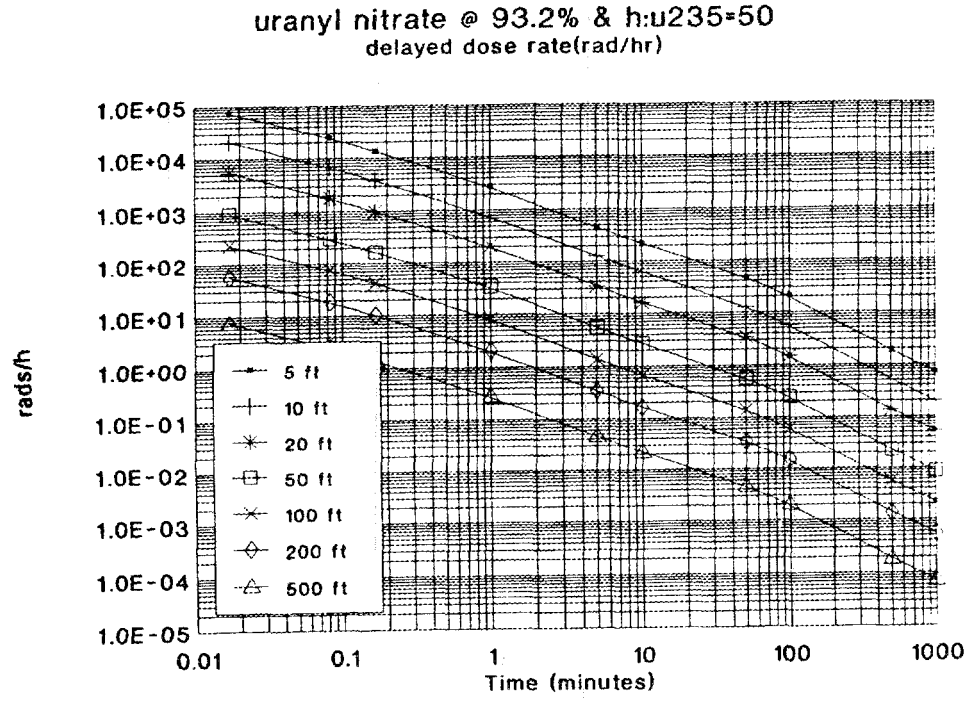


(b) accumulated one-minute dose (rads) based on estimated fission yield, distance from incident and time of entry after accident

Figure 4 Uranyl nitrate (93%) 1-D slide rule (normalized to 10^{17} fissions)

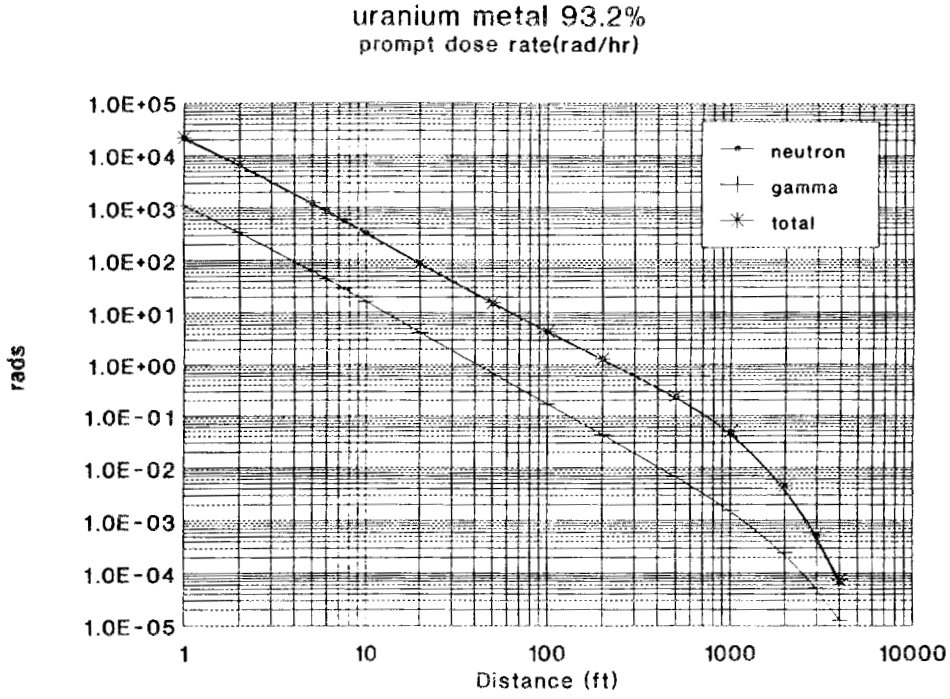


(c) integrated total dose (rads) based on estimated fission yield, distance from incident and elapsed time after accident

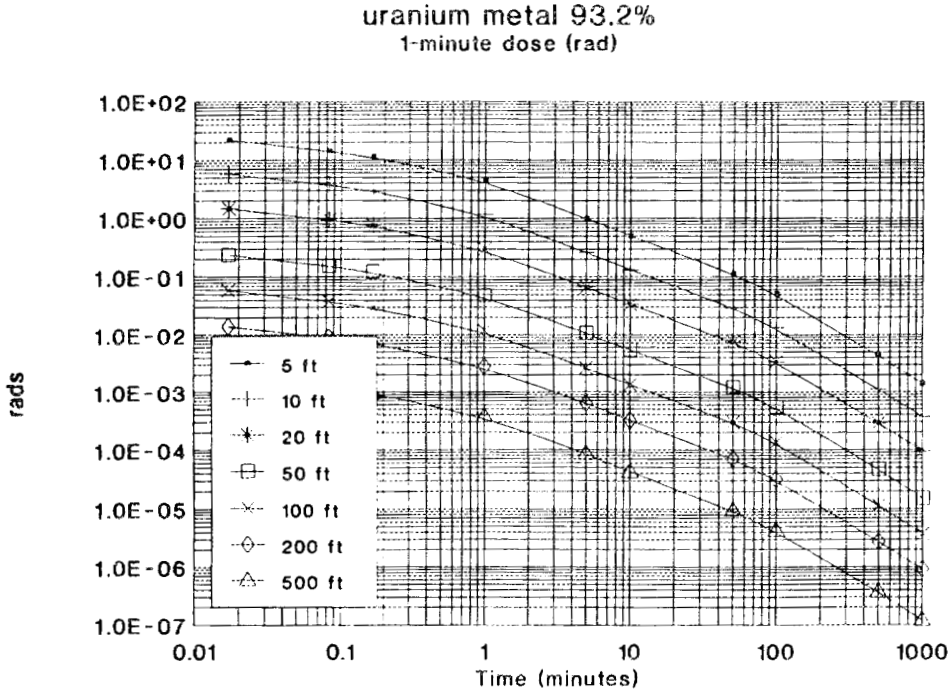


(d) estimated fission yield based on distant gamma dose rate (rads/h) and elapsed time

Figure 4 (continued)

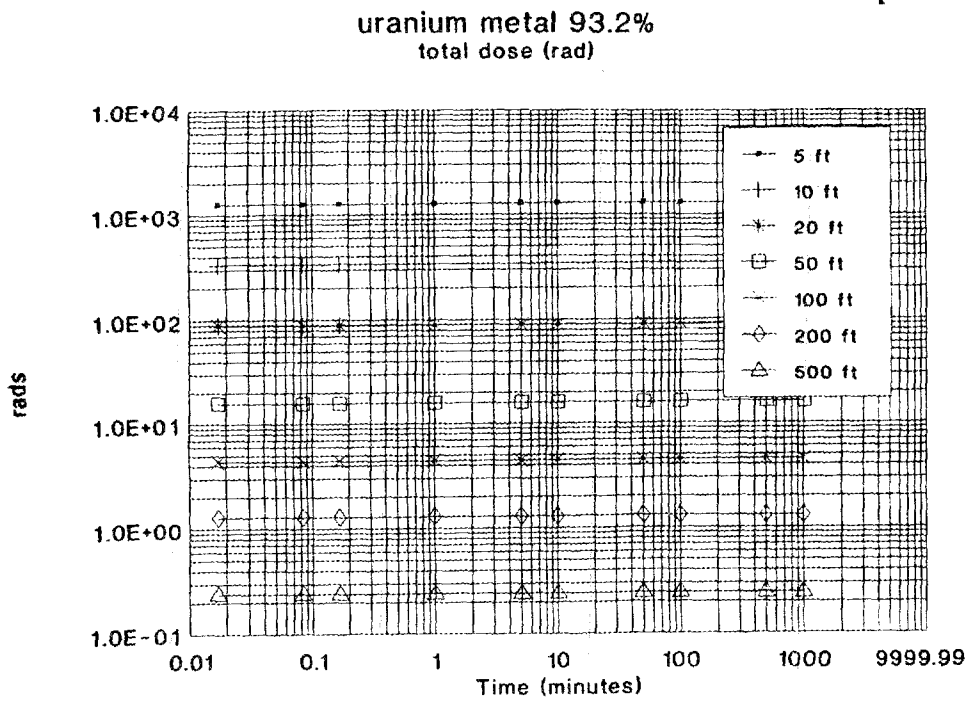


(a) estimated prompt doses (rads) based on total fission yield and distance from event

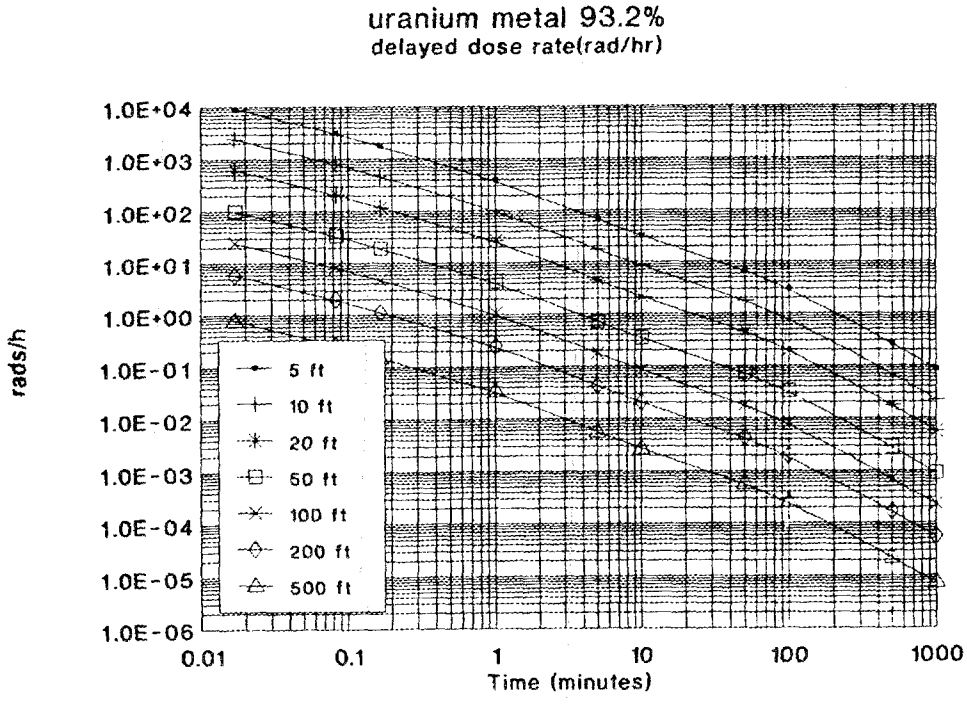


(b) accumulated one-minute (rads) based on estimated fission yield, distance from incident, and time of entry after accident

Figure 5 Uranium metal (93%) 1-D slide rule (normalized to 10^{17} fissions)

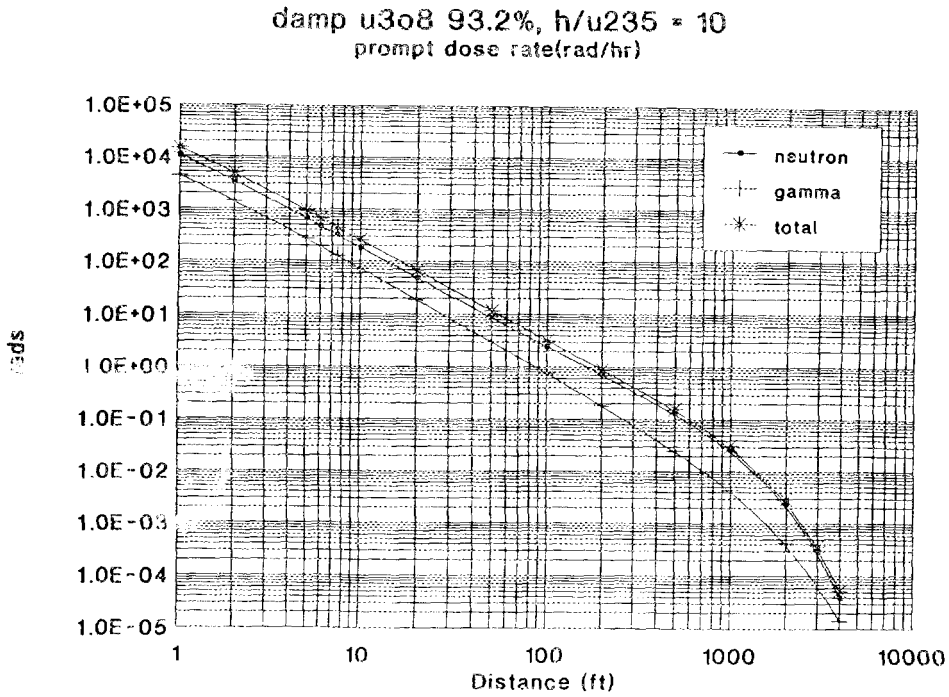


(c) integrated total dose (rads) based on estimated fission yield, distance from incident and elapsed time after accident

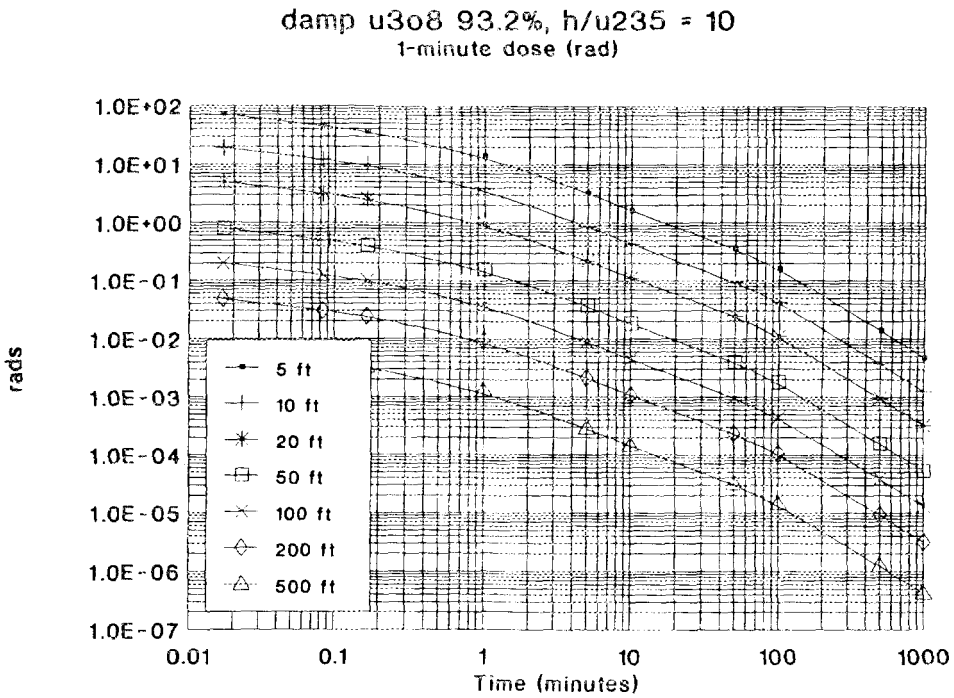


(d) estimated fission yield based on distant gamma dose rate (rads/h) and elapsed time

Figure 5 (continued)

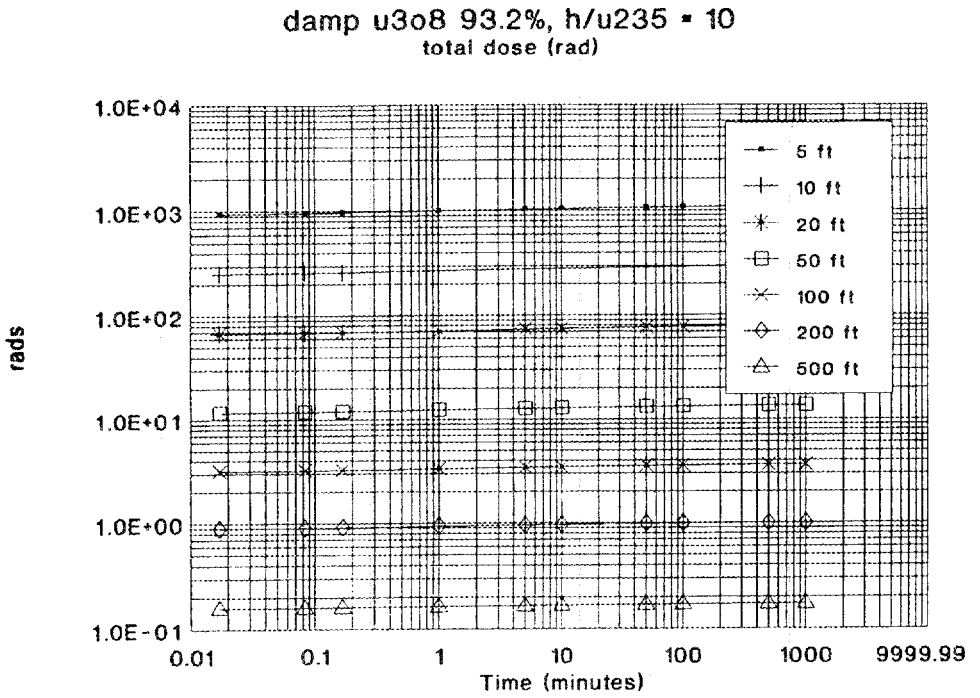


(a) estimated prompt doses (rads) based on total fission yield and distance from event

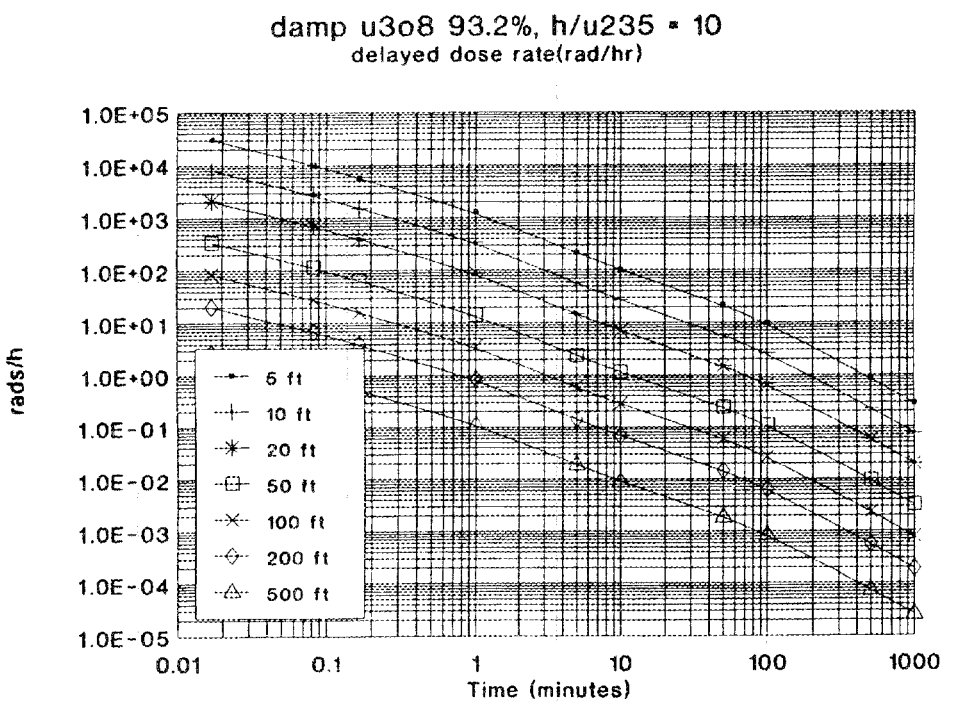


(b) accumulated one-minute (rads) based on estimated fission yield, distance from incident, and time of entry after accident

Figure 6 Damp U_3O_8 (93%) 1-D slide rule (normalized to 10^{17} fissions)



(c) integrated total dose (rads) based on estimated fission yield, distance from incident and elapsed time after accident



(d) estimated fission yield based on distant gamma dose rate (rads/h) and elapsed time

Figure 6 (continued)

with distance is slightly less than that of the gamma rays giving rise to a crossover effect for the uranyl nitrate slide-rule plot at about 100 ft (30.48 m). At distances less than about 100 ft (30.48 m), the gamma-ray contribution to the total dose exceeds the neutron contribution, but beyond 100 ft (30.48 m) the neutron contribution is larger than the gamma-ray contribution. This crossover effect is only seen for the uranyl nitrate plot since this is the only case where the neutron contribution is initially overshadowed by the gamma-ray contribution.

The crossover effect is due to the difference in slopes between the neutron and gamma-ray dose contributions. This difference in slope appears to be due to the presence of neutron backscattering or reflection from air beyond the particular dose location. This difference is exhibited by a comparison between the results in Ref. 2 and the 1-D slide-rule results given in Figs. 2 through 6. The last data points, near 4000 ft (1219.2 m), are up to a factor of 2 different. This difference is due to the exclusion of air beyond the 4000-ft (1219.2-m) radius in Ref. 2. This work includes air out to about 5000 ft (1524 m) with a resulting increase seen in the neutron doses only. Thus this increased reflection does not appear to affect the gamma-ray doses. This is confirmed by the lack of differences seen in the gamma-ray doses near the 4000-ft (1219.2-m) location as reported in Ref. 2 and this work.

The delayed dose rate vs time and distance curves seen in the (b) section of Figs. 2 through 6 have trends similar to those of the prompt dose vs distance plots. Specifically, the cases with the highest fission-product dose rates are those with the highest H/X ratios. The highest delayed doses correspond to the uranyl nitrate solution, which has the highest moderation (H/X = 500) and the lowest material density, thereby providing the least self-shielding of radiation. The lowest fission-product dose rates correspond to the uranium metal system because of the high material density, thereby providing the greatest self-shielding of radiation. In all cases the curve shapes are very similar to each other, with some very interesting trends. The curves seem to be the union of two connected trends with an inflection point occurring at about 10 min. Upon further investigation, these trends appear to be caused by the initial contribution of fission products (Br, Kr, Rb) with half-lives on the order of a few seconds to minutes, followed by the contributions of their daughters (Sr, Y, Zr, Nb) with half-lives on the order of a few hours. There does not appear to be any variation in these trends with increasing distance, as would be the expected if the longer- lived isotopes had drastically differing gamma spectra from the short-lived isotopes.

The remaining two sets of slide-rule plots are processed from the first two slide-rule curves and, hence, do not contain any new results. The total dose vs time and distance does show the dominance of the prompt contribution to the total doses. The trends in the 1-min doses are essentially the same as those of the previously described delayed dose rate vs time and distance curves.

4.5 COMPARISON OF 1-D RESULTS WITH MEASUREMENTS AND OLD SLIDE RULE

The slide rule developed during this project presents the prompt dose and delayed dose-rate contributions separately. This was also the intent of the original slide rule; however, due to the normalization of the original slide rule to measured doses, the prompt gamma dose portion of the tool invariably included some delayed contribution. This is because of the difficulty of obtaining measurements that do not include at least a small portion of the delayed contribution. Most measurements were obtained either by a prompt spike with the necessary delay before the measurements can be retrieved or by a steady-state power operation. Either method includes the accompanying delayed gamma-ray contributions to the measured dose. In order to compare the results presented above to the measurements on which the original slide rule was based, it was necessary to include an estimate of the delayed contribution present in the measurements. The original slide rule for highly enriched uranium metal was based on measurements performed at the Health Physics Research Reactor (HPRR)[1] in a pulse mode. In this mode, the

dosimeters were typically removed and examined after about 30–40 min. In this time, the additional delayed gamma dose is estimated[8] to be 1.7 times the prompt gamma dose. The original slide rule for highly enriched uranium solution was based on a mockup of the Y-12 accident[9] which operated at steady-state power conditions. The stated operation time was approximately 40 min, resulting in an estimated additional delayed gamma dose 1.4 times the prompt gamma dose. Although not a part of the original slide rule, this same procedure was carried out for the uranyl fluoride slide rule, with the same correction factor of 1.4 assumed. The measurement data used for the uranyl fluoride slide rule are based on the SHEBA experiments.[8]

The results of the 1-D dose rate computations with corrections as described above for delayed contributions are compared with measurements in Figs. 7 through 9 for the U(93) metal, uranyl nitrate, and uranyl fluoride slide rules. These results are presented in a unique manner to show several effects. The actual measurements are generally at about the 6- (1.8-m) to 8-ft (2.4-m) range from the device; thus it was decided to plot the measured data at this range along with all other ranges assuming an inverse-square relationship with distance as in the original slide rule. This allows not only general comparison of agreement with experimental data, but also the applicability of deviation from the inverse-square relationship. In general, the 1-D results shown in Figs. 7 through 9 compare very favorably with the measured data. The maximum deviations are seen for the U-metal gamma results (calculations 50% less than the measurements) and the uranyl nitrate neutron doses (50% higher than the measurements). The general features of the dose dependence with distance are an underprediction as compared to the inverse-square relationships near the device, changing spectrum and hence attenuation characteristics as deeper layers of material are considered, or as material of the same thickness is encountered at differing locations. The approach taken in this work is to evaluate the variation in the half-thickness values for several scenarios and establish ranges of half-thickness corresponding to expected situations common to criticality accidents. This procedure allows for selection of half-thickness values corresponding to a particular situation. As an additional option, a set of single half-thickness values were chosen for approximate use in generic situations. This optional set of half-thickness values has the added benefit of being extremely easy to use.

The procedure followed to generate the half-thickness ranges was to calculate for two of the accident scenarios, the uranyl nitrate solution and the uranium metal system, the attenuation through a 30.48-cm (12.01-in.) shield at approximately 120 cm (47.28 in.) from the center of the accident, followed by an additional 30.48-cm (12.01-in.) shield at approximately 914 m (3000 ft) from the center of the accident, and a final case where only the second shield was present. This procedure allows for determination of the variation of the half-thickness through a 30.48-cm (12.0-in.) shield, as well as the influence of location (near to or far from the accident) and the influence of previous shielding (the presence or lack of the shield near the accident). The half-thickness results for both neutron and gamma attenuation through various structural materials are given in Table 2. These results can be utilized in a simple hand calculation to estimate the attenuation (i.e., one half-thickness equals attenuation of a factor of 2, two half-thicknesses equals an attenuation of a factor of 4, etc.). Additionally, these values can be used in the following equation to more closely approximate the attenuation:

$$DR_s = DR_u e^{-0.693x/X}$$

where

DR_s = shielded dose rate (rads/h),
DR_u = unshielded dose rate (rads/h),
x = thickness of shield (in.),
X = half-thickness value (in.).

Development of Prototype Nomographs

As a simplification of the procedures that utilize the above data, an additional set of single-value half-thicknesses were generated to correspond generically to a series of thin shields that would be typical of storage or process buildings and other nonreactor applications. The half-thicknesses for steel, concrete, and water shields are 2.7 (6.86 cm), 2.9 (7.37 cm), and 2.5 in. (6.35 cm) for neutron attenuation, and 1.8 (4.57 cm), 4.7 (11.94 cm), and 7.5 in. (19.05 cm) for gamma rays. These half-thicknesses are averages based on a series of 10 typical "thin" shields for steel [1 in. (2.54 cm)], concrete [3 in. (7.62 cm)], and water [3 in. (7.62 cm)] stacked one after the other with 24 ft (7.32 m) of air separating each shield. These single half-thickness values can be easily represented graphically for ease of use as shown in Fig. 10. From these plots the dose reduction factor due to each material between the accident and the detector location of interest can be obtained.

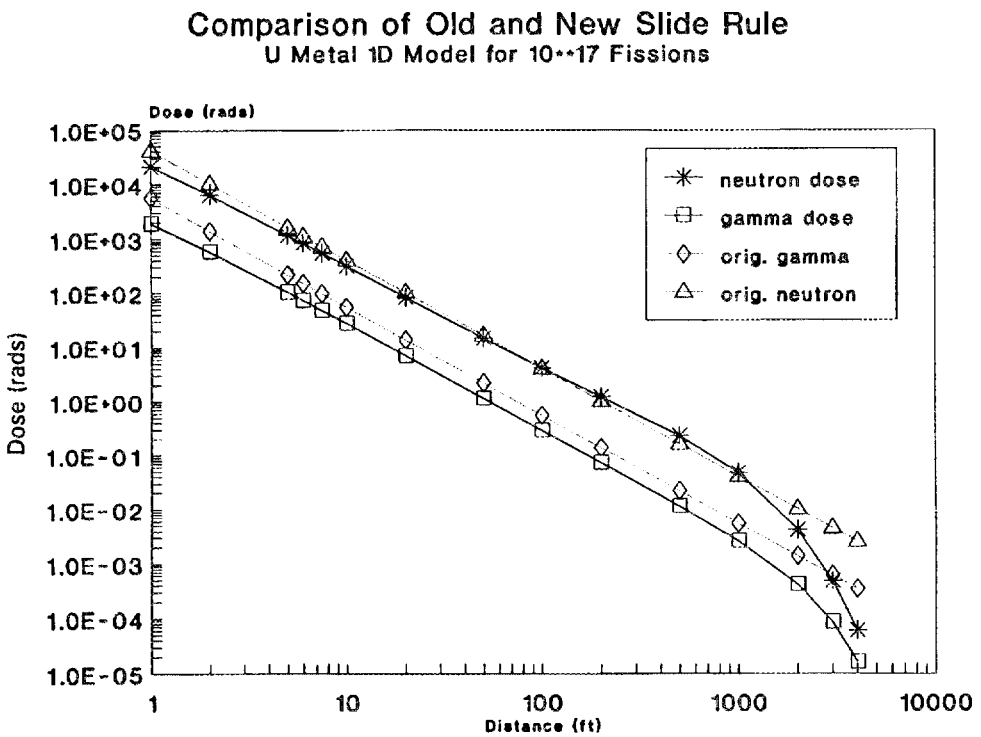


Figure 7 Old and new slide rules for U metal (10^{17} fissions)

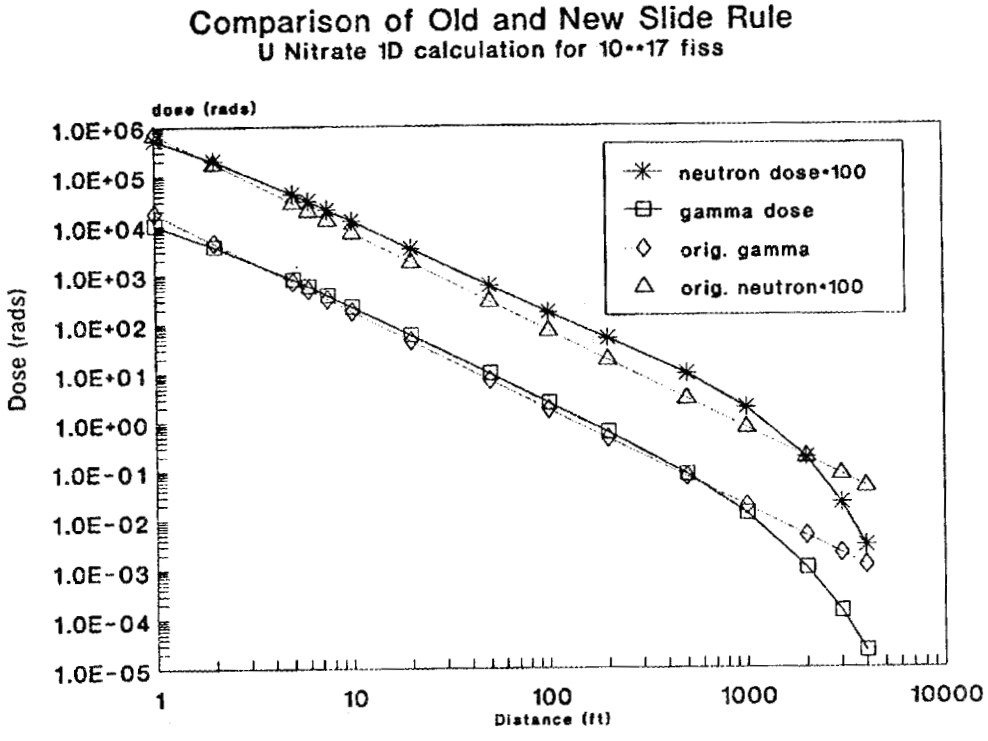


Figure 8 Old and new slide rules for U nitrate (10^{17} fissions)

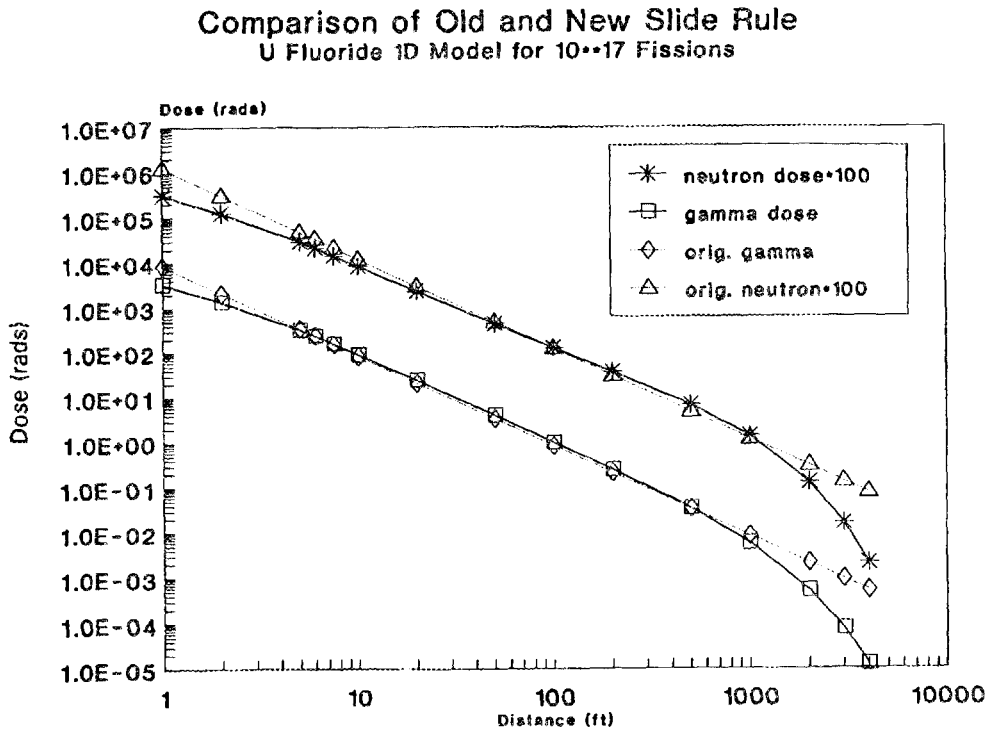


Figure 9 Old and new slide rules for U fluoride (note: old slide is simulated since only metal and nitrate solution slide rules were originally provided)

Table 2. Neutron and gamma-ray half-value thicknesses (inches)

Shield type	Neutron	Gamma ray
Steel, density = 7.82 g/cc	4.9-3.5[a]	1.5-4.0[a,b]
Concrete, density = 2.3 g/cc	2.2-2.5[c]	5.5-4.3[a]
Water, density = 1.0 g/cc	1.2-2.5[c]	5.0-7.5[d]

[a]Use first value for small shield thickness, second value for large thickness [the values are valid for thicknesses of 1 (2.54 cm) to 12 in. (30.48 cm)].

[b]Use second value for a dry system.

[c]Near the accident [<50 ft (15.24 m)] use the second value for thin shields and the first value for thick shields, reverse for larger distances from the accident.

[d]Near the accident [<50 ft (15.24 m)] use the first value, further away use the second.

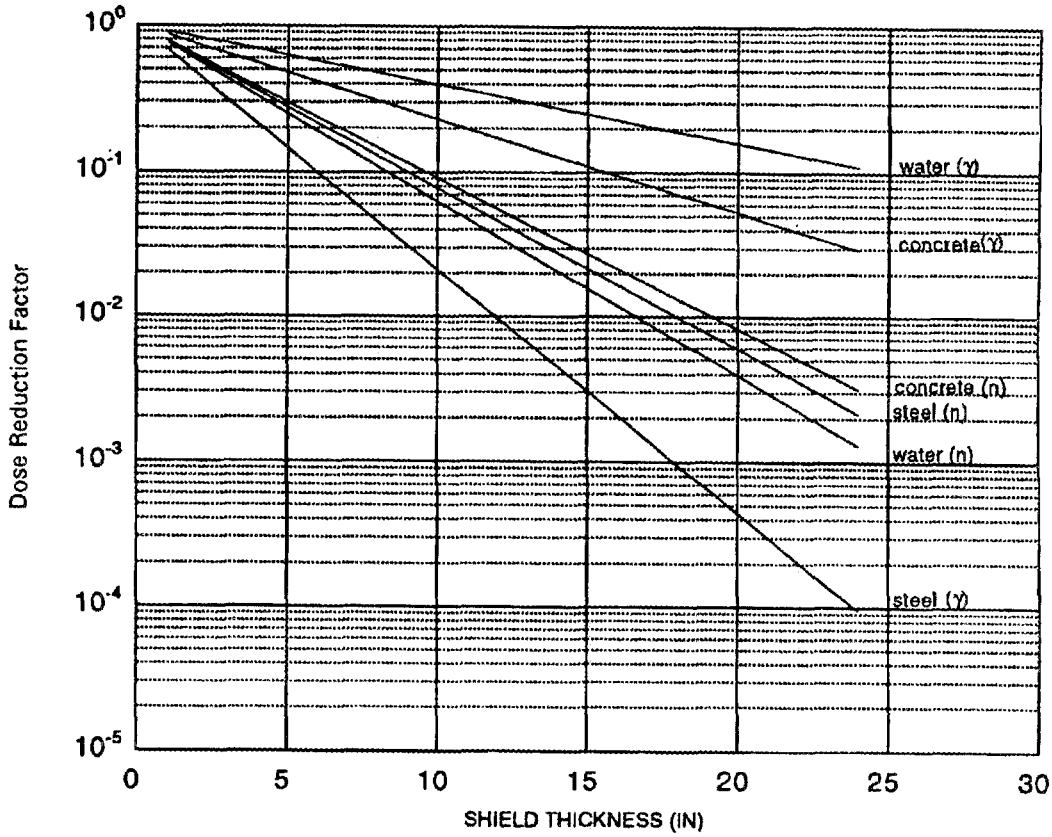


Figure 10 Dose reduction for various shields

5 EXTENSION OF SLIDE RULE TO 2-D BASIS

The generation of a prototype 1-D slide rule as described above allowed for the estimation of many of the differences between the original measurement-based slide rule and a calculational-based slide rule. Extension of the 1-D scoping studies using 2-D methods allows for estimation of 2-D effects. The influence of the air/ground interface should be evident as well as the possible contributions due to radiation skyshine. The inclusion of skyshine-only information is also included to allow determination of accident characteristics where substantial shielding is present between the postulated accident and the desired location of radiation hazard information. The inclusion of this revised information into a hand-held format very similar to the original slide rule, with much wider distribution, should greatly enhance the tool's viability and function.

The development of the 2-D slide-rule information utilizes the 2-D discrete-ordinates code, DORT.[10] This procedure allows for the inclusion of air/ground effects along with a separate estimate of the skyshine component of the dose vs distance.

5.1 DORT DESCRIPTION

DORT is a 2-D, discrete-ordinates, coupled neutron gamma-ray radiation transport code. Fixed sources and sources resulting from particle interaction within the medium are allowed. The principal application is to the deep-penetration transport of neutrons and photons. Since many physical systems associated with radiation can be approximated fairly accurately with a 2-D analysis, DORT provides a rigorous analytical solution method. DORT is particularly applicable to air-over-ground problems, since both the air and ground can be accurately modeled along with a finite-dimensional source of neutron/photons.

5.2 2-D MODEL DESCRIPTION

The geometry for the 2-D slide-rule models consisted of a simple air-over-ground configuration with a source located at the center of a right-circular cylinder and 1 m (3.3 ft) above the ground. The source was generated via the code GRTUNCL[11] which uses the leakage spectrum generated in the 1-D analysis as input. Using the assumed leakage spectrum, GRTUNCL calculates the uncollided flux and the first-collision source at all locations in the 2-D geometry. This first collision source is a volumetric source instead of a point source. The collided flux is then calculated via DORT and added back to the uncollided flux by an auxiliary routine. This auxiliary code also calculates the resulting doses at specific locations 1 m (0.30 ft) above the ground using the Henderson flux-to-dose factors.[12] The DORT model simulates the air-over-ground geometry with 170 radial mesh intervals from 0 to 153,000 cm (5023.5 ft) and 197 axial intervals from –30.48 (-12 in.) to 153,000 cm (5023.5 ft). The ground is included from –30.48 (-12 in.) to 0 cm in the axial direction with air above 0 cm. The axial and radial mesh widths were set equal to each other, where possible, with a 30.48-cm (12-in.) mesh from near the source to about 5.5 m (16.7 ft) away. Beyond 5.5 m (16.7 ft), the mesh size was increased using a rule that the radius from mesh to mesh not increase more than about 10%. This ensures that the flux drop per interval due to the inverse-square attenuation is less than about 20%. A 2-D infinite-air case utilizing this mesh compared well with a 1-D infinite-air case with a very fine mesh. The angular quadrature used was an S_{10} with 70 angles. Comparisons with a finer angular quadrature set with 240 angles showed only small differences in the calculated doses. The air composition and cross sections for the 2-D computations were identical to the air composition used in the 1-D analyses. The ground was modeled as a 1-ft (30.48-cm) layer of concrete (SCALE material REG-CONCRETE) with a density of 2.3 g/cm^3 (143 lb/ft^3). These models were used in both the prompt and delayed dose slide rules. The only differences in the prompt vs delayed analyses were the leakage spectra, in all cases the same as the corresponding prompt or delayed gamma spectra from the 1-D analyses.

5.3 SKYSHINE CONTRIBUTION

The use of 2-D computational methods for the slide-rule update allowed for the inclusion of an additional set of results corresponding to a heavily shielded criticality accident. In this case, the direct radiation reaching a postulated detector location would be very small. However, the skyshine contribution could be significant where significant air scatter paths are available. This situation could arise for a criticality accident within a pit or shielded by a large number of limited-height drums or other shields between the accident and the detector location(s). The procedure used to generate these additional results was very similar to the standard procedure described above, except that the GRTUNCL uncollided flux and collided source generation were limited to a leakage spectrum with angles directed upward in a 90° cone. The effect of this limitation is to force all neutrons or photons leaking from the critical assembly to be directed upward. The only way for the particles to reach the detector locations is to scatter in the air and then travel to the detector. Indeed the uncollided flux for these cases was 0.

5.4 2-D SLIDE-RULE RESULTS

The 2-D slide-rule results, generated as described above and processed using the same methods as the 1-D slide-rule results, are given in Figs. 11 through 15 for the 4.95 wt % enriched uranyl fluoride, $U(4.95)O_2F_2$; damp 5 wt % enriched uranium dioxide, $U(5)O_2$; 93.2 wt % enriched uranyl nitrate, $U(93.2)O_2(NO_3)_2$; 93.2 wt % enriched uranium metal; and damp 93.2 wt % enriched uranium oxide, U_3O_8. The general appearance of these 2-D slide rules is nearly identical with the 1-D slide rules. For this reason, the differences between the 1- and 2-D results were investigated further. In Figs. 16 through 20, ratios of the 2- to 1-D results are shown for the prompt neutron and gamma-ray doses vs distance. These plots allow for a visual representation of the 1- vs 2-D effects. As stated earlier, the 1- and 2-D results were compared for an infinite-air model with very good agreement. Thus the differences seen in Figs. 16 through 20 are primarily the ground effects for the various systems. The general trends seen are 20 to 40% higher doses in two dimensions at about 10 ft (304.8 cm) and 50-80% lower 2-D doses at 4000 ft (1219.2 m). The higher doses near 10 ft (304.8 cm) arise from the effect of reflection from the ground. The lower doses for distances around 1000 ft (304.8 m)arise from the increased attenuation due to the ground interface. The trends for the uranium metal plots in Fig. 19 are quite different from the others. The neutron dose peak at 10 ft (304.8 cm) is very similar to the remaining plots; however, the photon dose peak is substantially larger than in the other curves. Also, the photon dose ratios show an additional peak at 500 ft (152.4 m) where the 2-D doses are more than a factor of 2 higher than the 1-D doses. This peak is due to the secondary gamma rays produced in the ground, which is obviously not present in the 1-D calculation. The location of this peak appears to coincide with the air attenuation and the accompanying thermalization of neutrons, thus enhancing the probability of thermal neutron capture in the ground.

These 2-D results could be plotted against the original slide-rule results shown in Figs. 7 through 9; however, based on the magnitudes of the 2- to 1-D ratios seen in Figs. 15 through 19, the general trends would remain the same.

5.5 2-D SUMMARY

The 2-D slide rules presented above represent a significant improvement over the original slide rules in that air/ground effects are included from the accident out to 4000 ft (1219.2 m). The original slide rules assumed an inverse-square relationship that was normalized to measurements at some 6 (1.8 m) to 8 ft (2.4 m) from the accident, which only partially accounts for ground effects. These 2-D slide rules also tabulate the skyshine effect, which should be useful for situations where the accident is heavily shielded, but significant pathways through the air exist for bypassing this shielding.

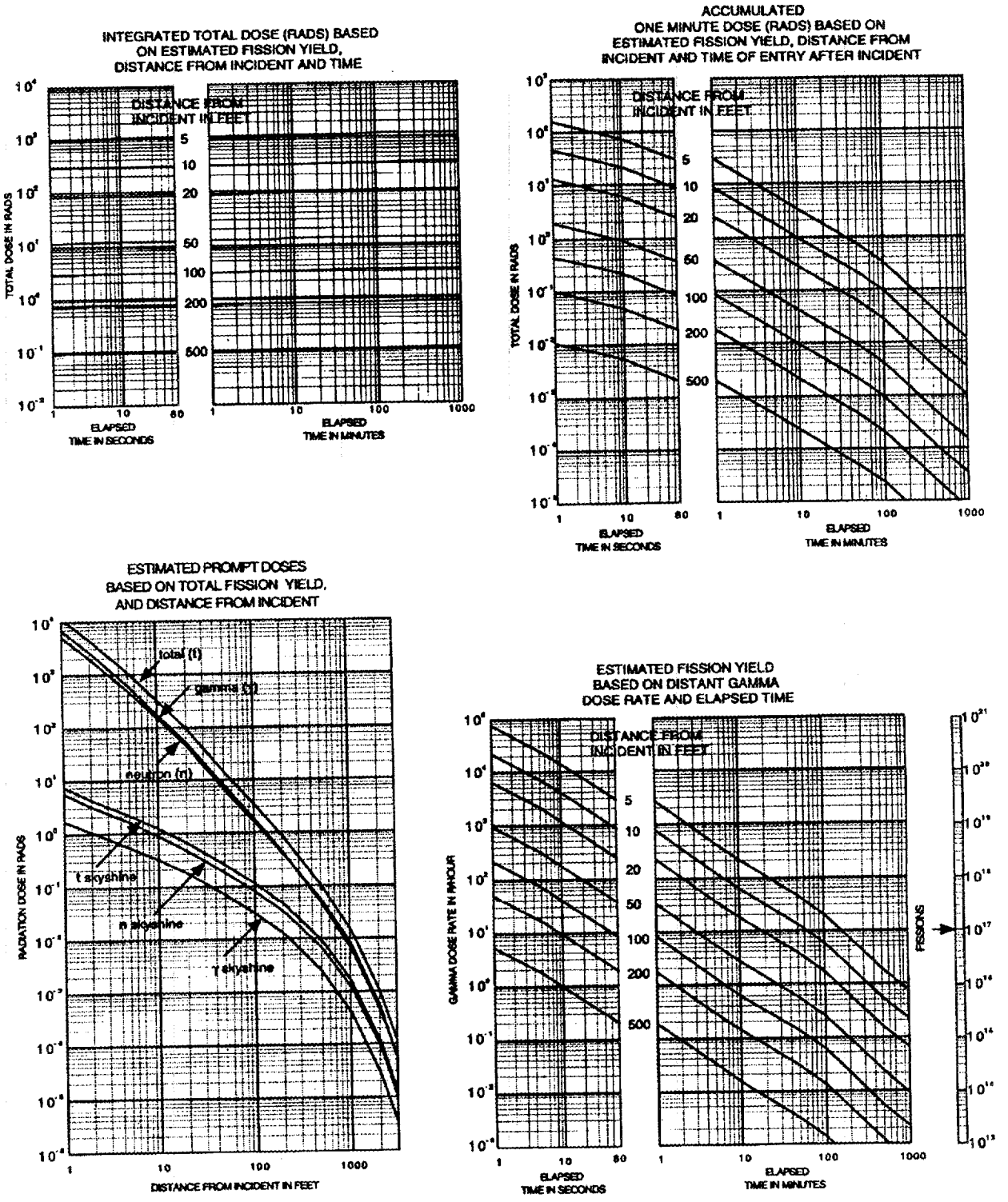


Figure 11 Solution of $U(93.2)O_2(NO_3)_2$ @ $H/^{235}U = 500$

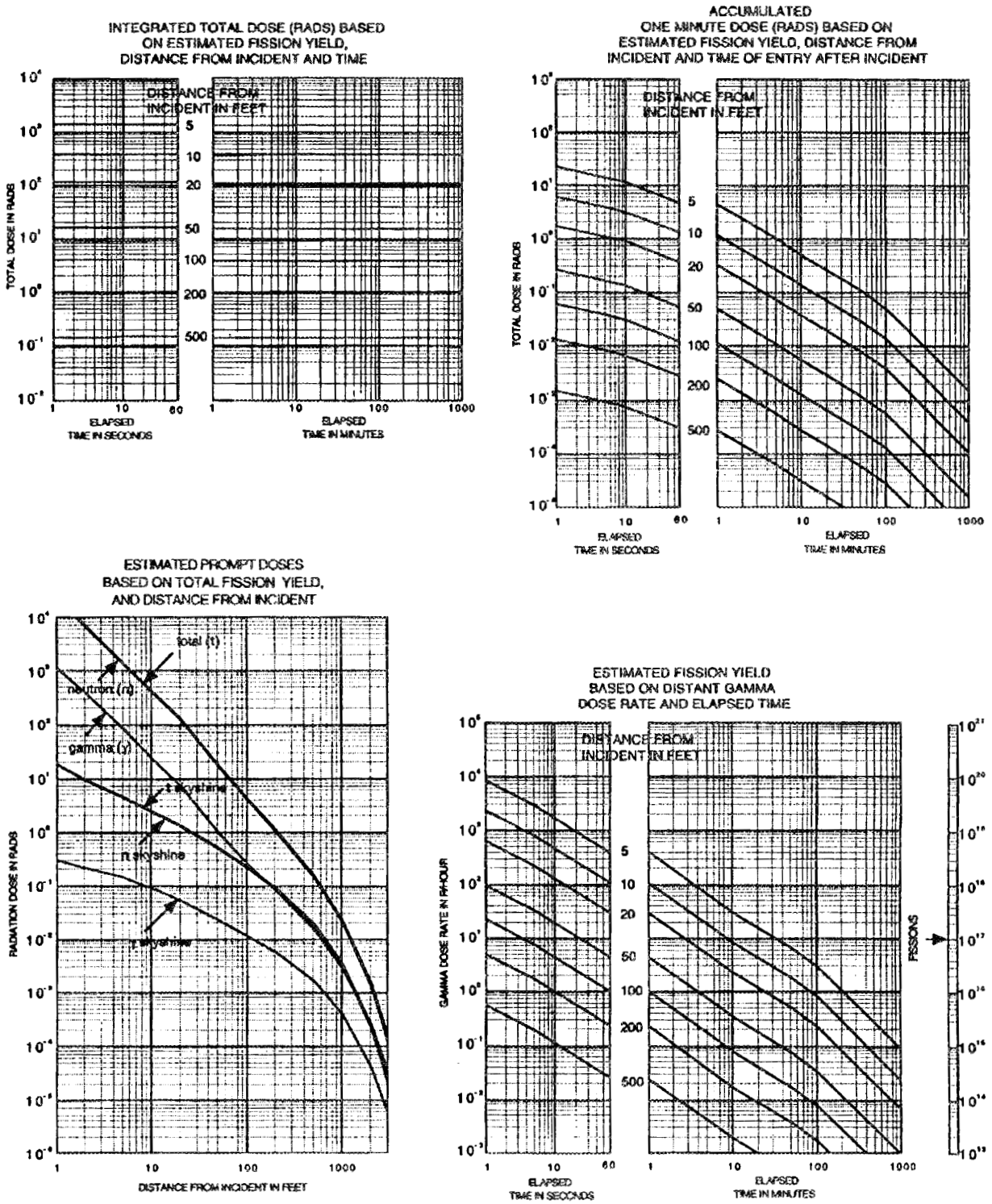


Figure 12 U(93.2) metal

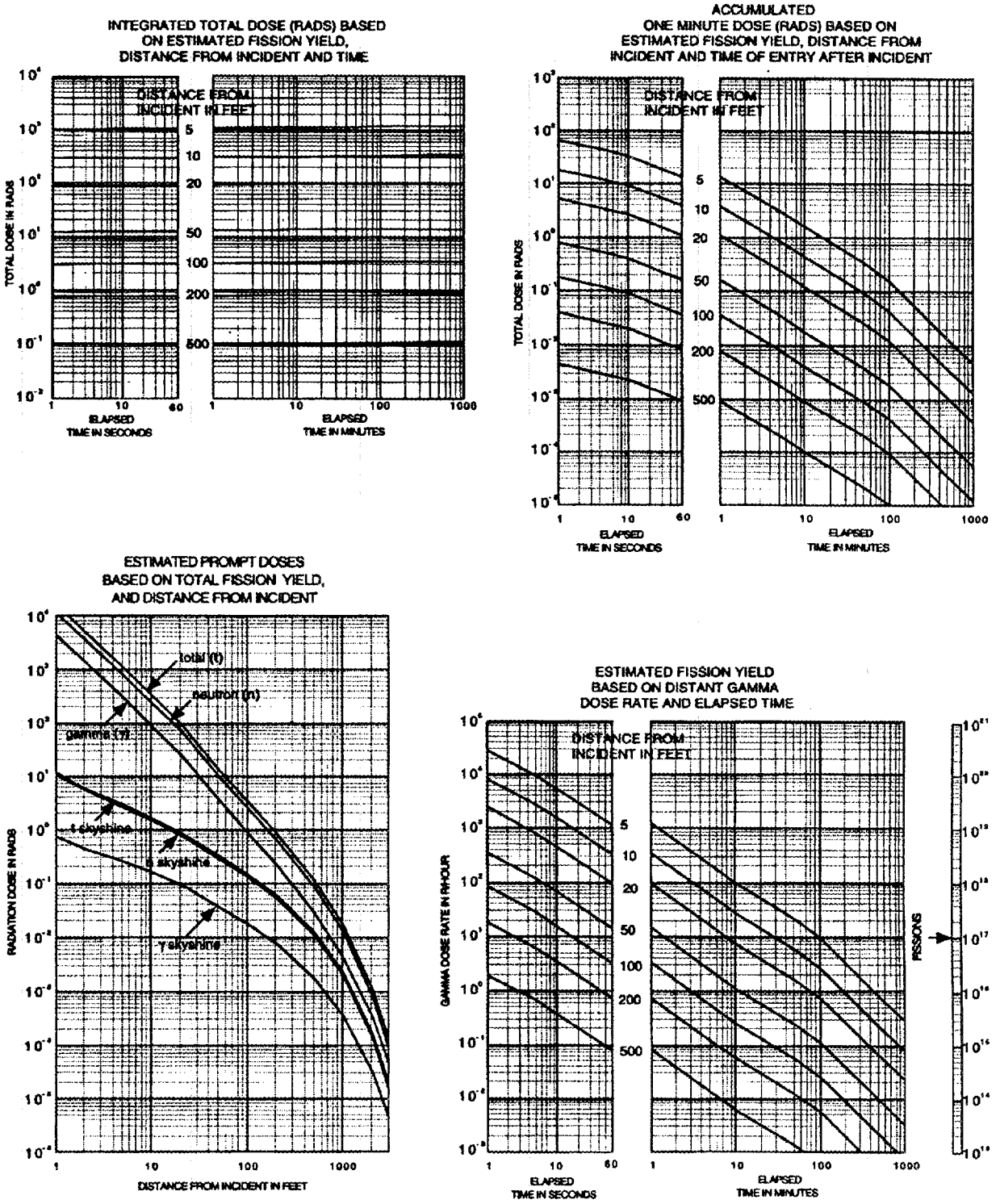


Figure 13 Damp $U(93.2)_3O_8$ @ $H/^{235}U = 10$

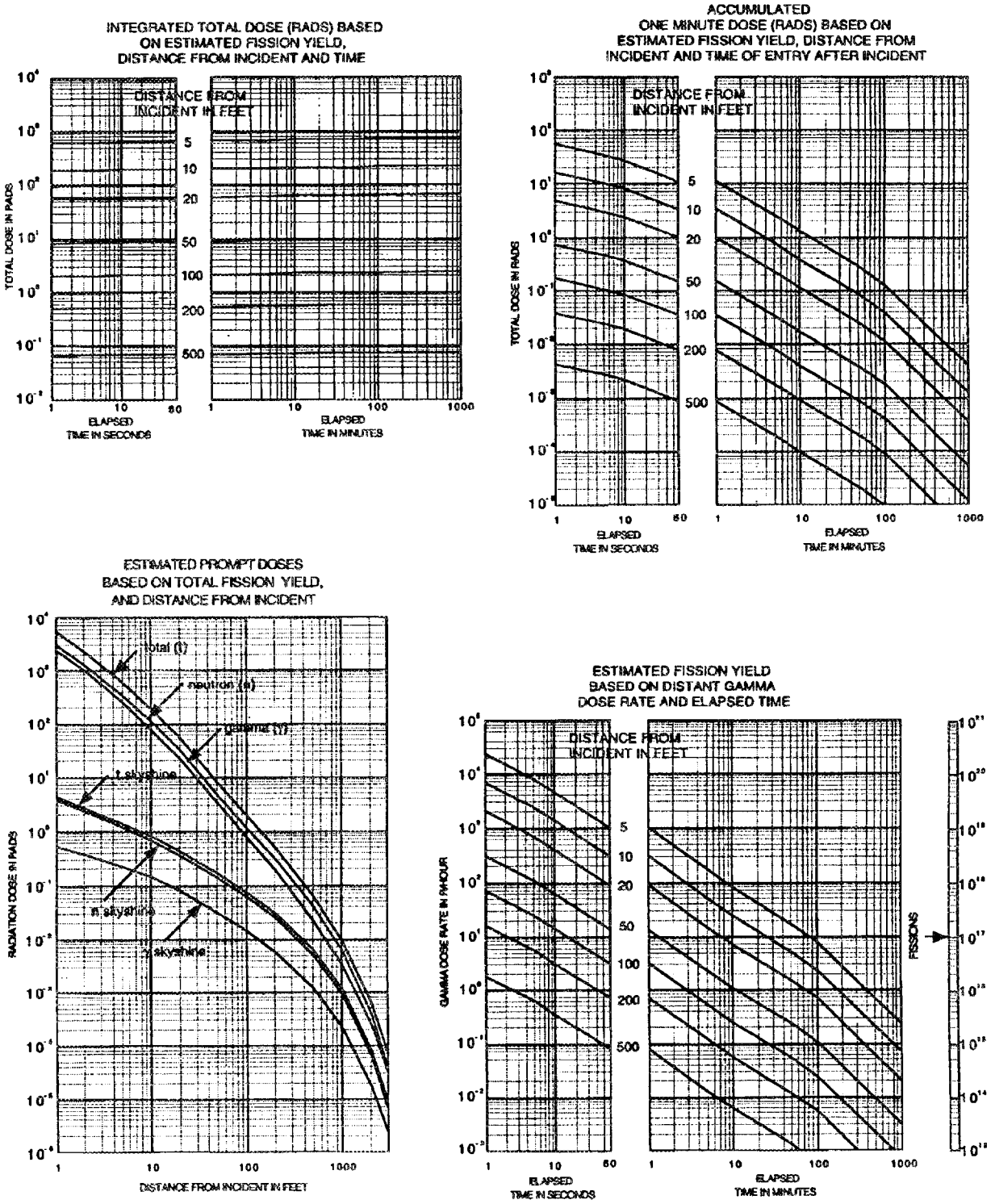


Figure 14 Damp $U(4.95)O_2F_2$ @ $H/^{235}U = 410$

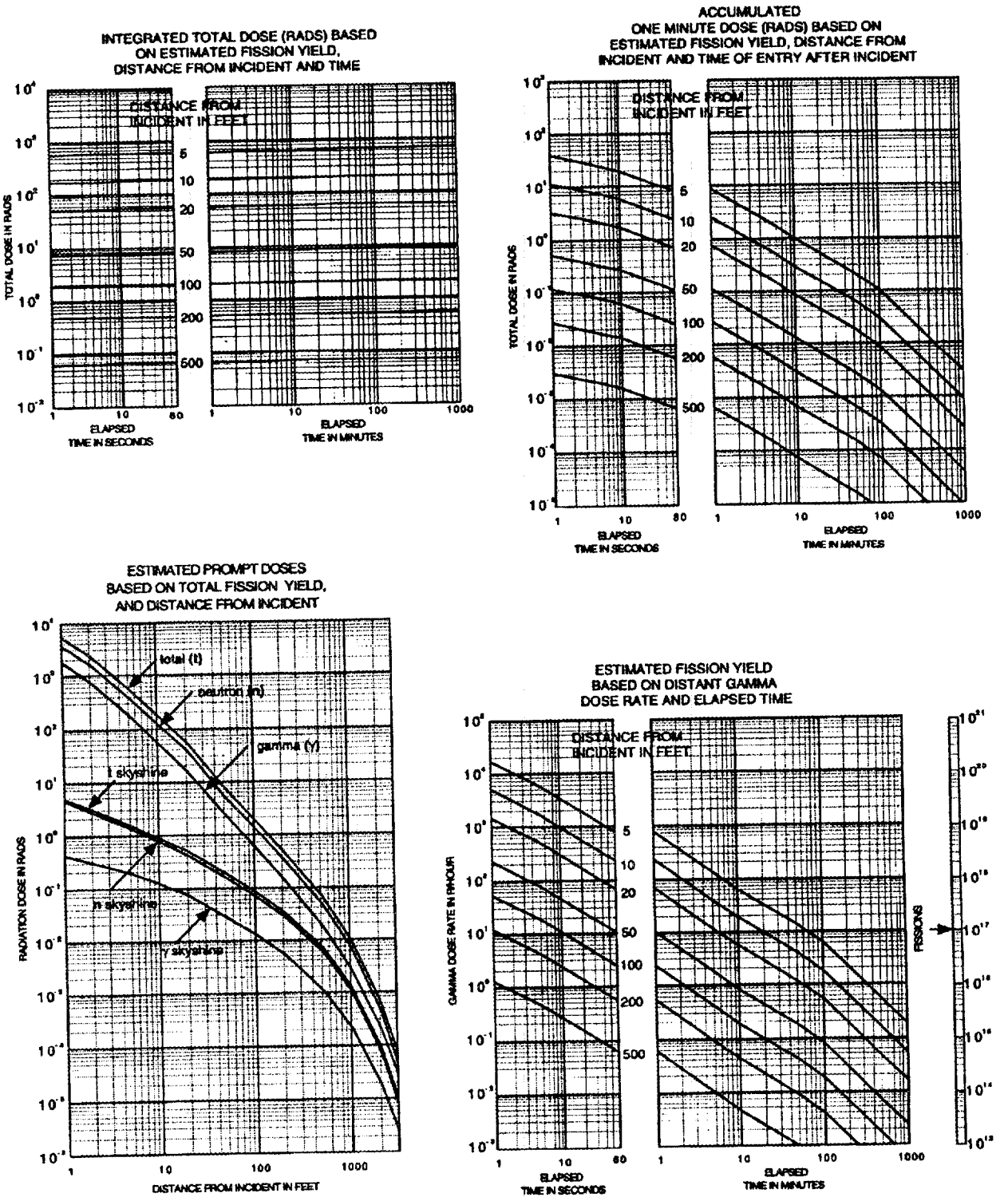


Figure 15 Damp $U(5)O_2$ @ $H/^{235}U = 200$

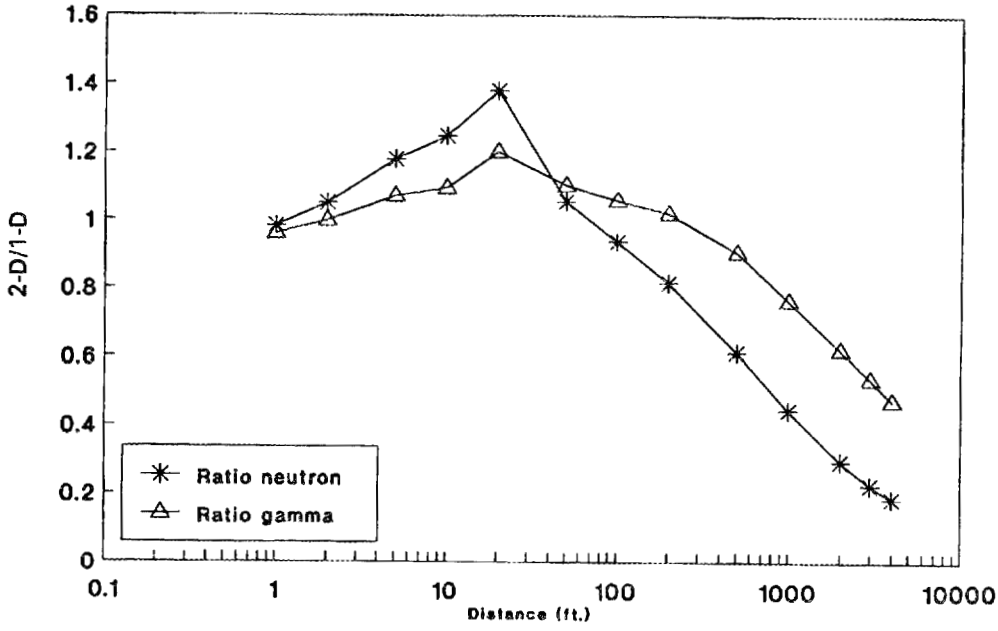


Figure 16 Comparison of 1- and 2-D uranyl fluoride slide rules

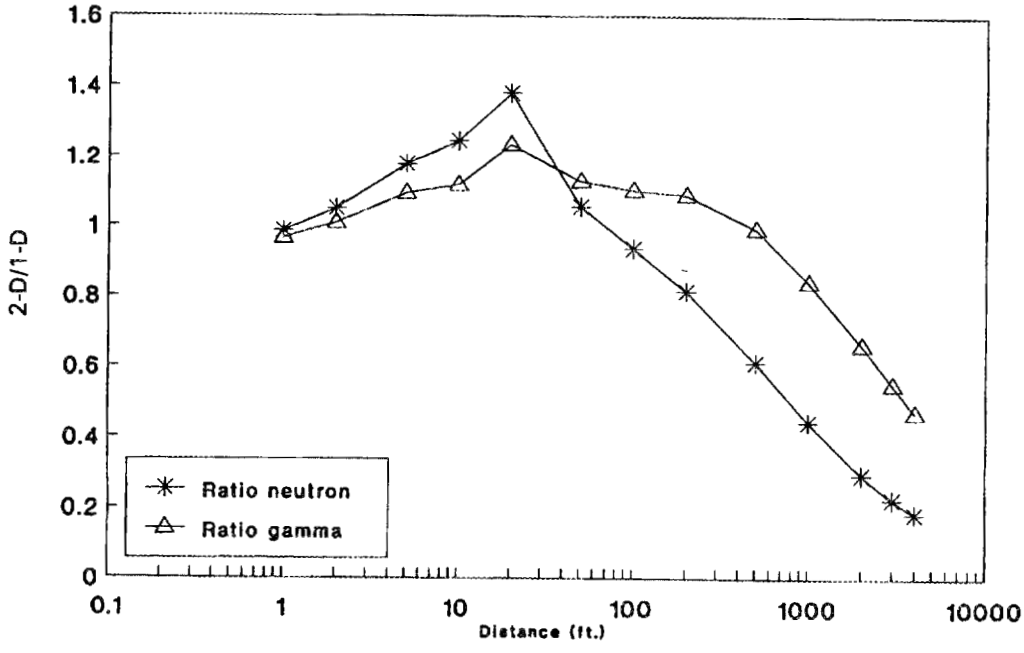


Figure 17 Comparison of 1- and 2-D damp UO_2 slide rules

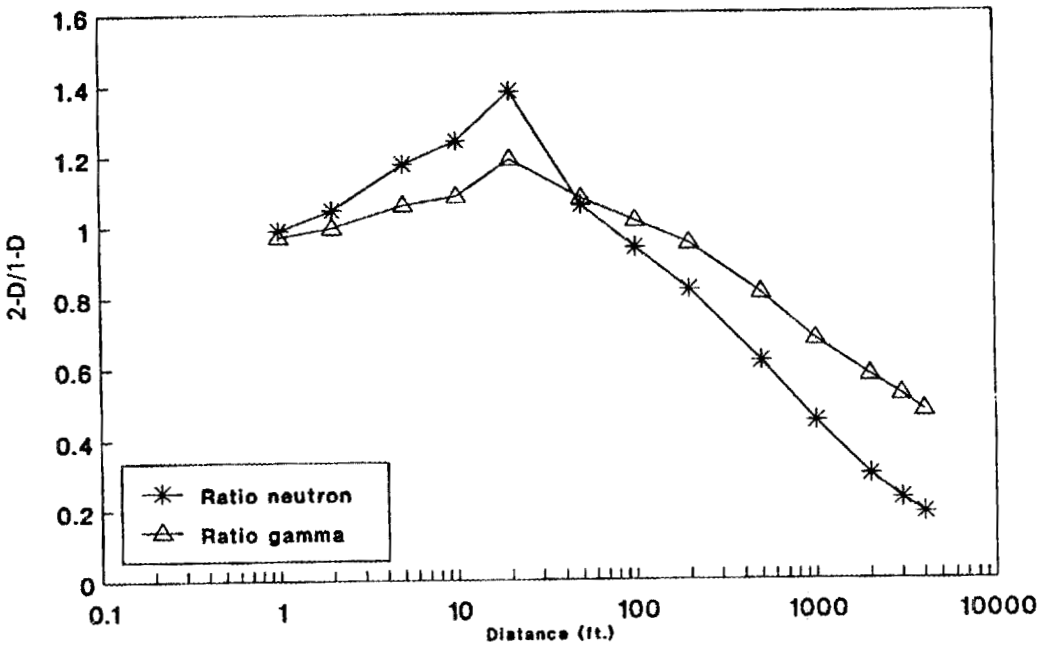


Figure 18 Comparison of 1- and 2-D uranyl nitrate slide rules

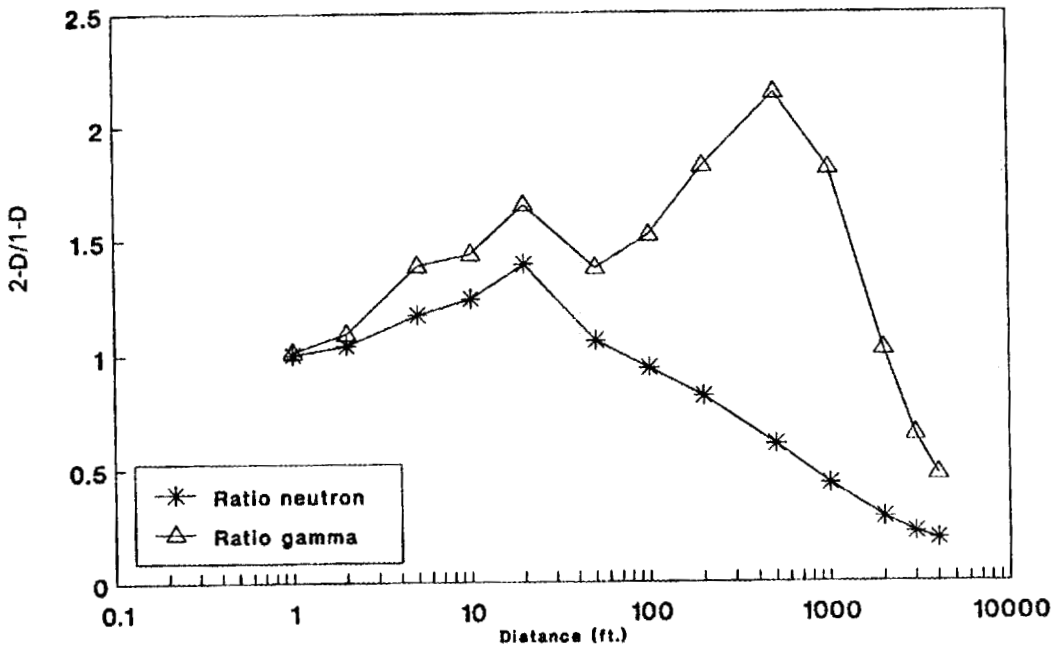


Figure 19 Comparison of 1- and 2-D U metal slide rules

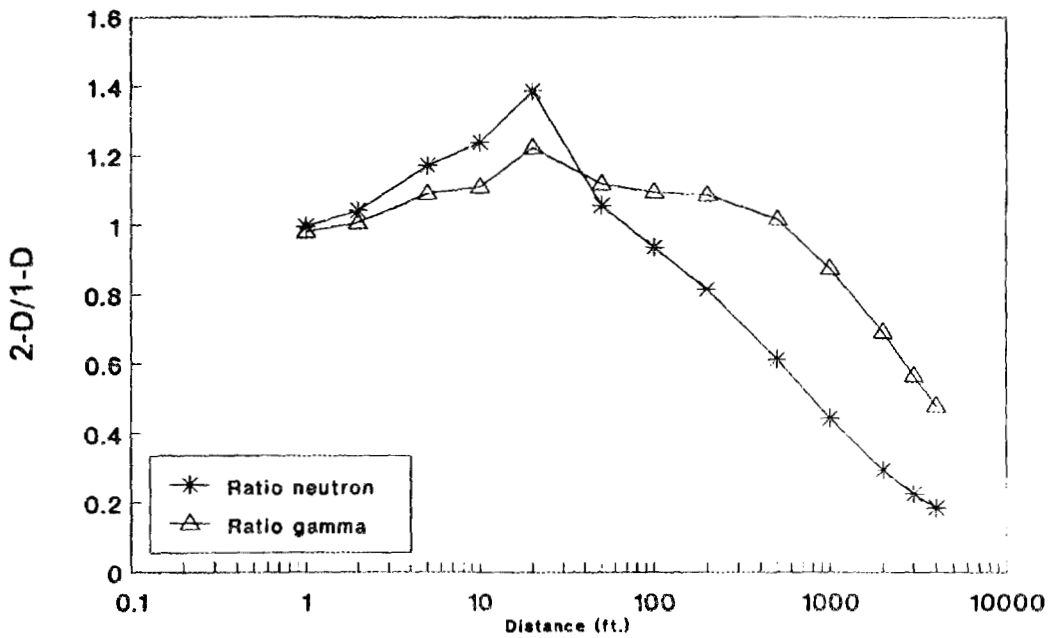


Figure 20 Comparison of 1- and 2-D damp U_3O_8 slide rules

The 1-D slide rules which were performed largely for demonstration purposes are included in this report since for certain situations they could be more appropriate than the 2-D slide rules. For scenarios where the criticality alarm system is located on the upper levels of a building or perhaps some 10 ft (3.0 m) or so above the floor, the ground effects built into the 2-D results would not be appropriate.

The slide rules presented in Figs. 11–15 are nonfunctioning versions of the actual slide rules which are contained in Volume 2 of this document. The slide rules contained in Volume 1 are normalized to a fission yield of 10^{17} fissions. The functioning slides in Volume 2 allow for the effect of various fission yields to be easily determined.

6 GENERATION OF FISSION-YIELD PREDICTION MODELS

In the planning for the updated slide-rule tasks performed in this work, it was envisioned that a valuable addition to the capabilities of the original slide rule would be the inclusion of estimates of fission yield information for various accident criticality scenarios. This would enhance the tools used in planning activities since an estimate of the magnitude of the accident could be generated as well as possible emergency responses to various accident scenarios. In keeping with the philosophy of the original slide rule, the yield information selected for inclusion is neither state-of-the-art nor complete for every system. The yield information is intended to be approximate and useful for planning purposes and emergency response. The information should also be useful to analyze trends in yield information as a function of the various system parameters selected for inclusion.

6.1 THEORY

Hansen has developed an equation relating the probability of initiating the first persistent chain reaction as a function of time t_1 after the system reaches delayed criticality.[3]

$$P(t_1) = \sqrt{\frac{2\alpha\tau S^2}{\pi\bar{\upsilon}^2\Gamma_2^2}}\left\{\frac{1 - \mathrm{erf}\left[\left(\frac{\alpha}{2\tau}\right)^{\frac{1}{2}} t_1\right]}{2}\right\}\left(\frac{2S\tau}{\bar{\upsilon}\Gamma_2} - 1\right)\exp\left(-\frac{\alpha t_1^2}{2\tau}\right) . \tag{1}$$

This equation was developed for a near-critical fissile system undergoing a ramp reactivity insertion. The quantity $P(t_1)dt$ is the probability that the first persistent neutron chain reaction begins in the time interval dt about t_1. The remaining terms are defined as follows:

α = ramp reactivity insertion rate (change in system multiplication factor, Δk, per second);
τ = neutron lifetime (s);
S = equivalent neutron source strength for the system (neutrons/s);
$\bar{\upsilon}$ = average number of neutrons emitted per fission;
Γ_2 = the parameter $\frac{\overline{\upsilon(\upsilon-1)}}{\bar{\upsilon}^2}$;
erf = the error function, which is given by

$$\mathrm{erf}(x) = \frac{2}{\sqrt{\pi}}\int_0^x \exp(-y^2)\,dy .$$

In this study, the value of $\bar{\upsilon}$ was taken to be 2.5 neutrons/fission. The value of Γ_2 was noted by Hansen to be about 0.8 for various fission nuclides. The above equation is conditional upon the following relationship:

$$S\tau \ll 1 .$$

This relationship should hold for most of the systems envisioned in this work. Physically this quantity describes the statistical variation in the expected time of the fission pulse after the achievement of delayed criticality with a uniform reactivity insertion rate of α. For this work, the important quantity was the average initiation time, since this

determines the amount of added reactivity and hence the average, expected fission yield. This expected fission yield as a function of the time after delayed criticality is given by Hansen[3] as

$$E(t_1) = \frac{1}{b}\left[\alpha^2 t_1^2 + 2\alpha\tau \ln\left(\frac{2\alpha\tau\, W(t_1)}{b}\right)\right]^{\frac{1}{2}}, \tag{2}$$

where $W(t_1)$ is given by

$$W(t_1) = \frac{\left(\frac{8\alpha\tau}{\pi \bar{\upsilon}^2 T_2^2}\right)^{\frac{1}{2}} \exp\left(-\frac{\alpha t_1^2}{2\tau}\right)}{1 - \mathrm{erf}\left[\left(\frac{\alpha}{2\tau}\right)^{\frac{1}{2}} t_1\right]} \tag{3}$$

This equation is based on the energy model of reactivity quenching described by Stratton[13] and others, which postulates that negative reactivity introduced by the first fission pulse is proportional to the total energy released in the pulse. The value b has the units of Δk per fission. Thus the value b characterizing the fissile system must be known to employ the above fission-yield estimate.

The constants that are required for use of this method include S, τ, and b. The value of S is taken to be 105.66 neutrons/s-kg of uranium for HEU solution, and 40.0 neutrons/s-kg U for 5 wt % enriched uranium solutions. The value of S for HEU solution was taken from measurements reported by Hankins,[14] while the 5 wt % value was calculated via ORIGEN-S.[15] These neutron sources include both spontaneous fission and production from alpha particle interaction with predominately oxygen in the water. The values of τ are based on calculations performed with the KENO V.a code.[16] These calculations were performed for uranium densities of 25 to 2500 grams U/L. The results of these calculations are shown in Fig. 21. The actual value of τ is determined by interpolation to the uranium density given for each system. The values for b as a function of system critical volume were taken from a separate study.[17] Shown in Fig. 22 are the results of that study where plots of derived values of b vs critical system volume for the CRAC[18] experiments. A least-squared curve fit of the logarithms of b and V results in the following expression:

$$b = 7.54 \times 10^{-18}\, V^{-1.47},$$

where V is in liters, and b has the units of Δk/fission.

The physical interpretation of the parameter, b, is that the negative reactivity feedback of the system can be characterized as being proportional to the energy released by fission. The parameter, b, is simply the proportionality constant. All systems cannot be adequately characterized in such a manner, but for many common systems this approach is felt to be adequate for the purposes herein. Obviously, a system that is autocatalytic (i.e., a positive reactivity feedback as a function of energy released) would not be adequately represented by this approach.

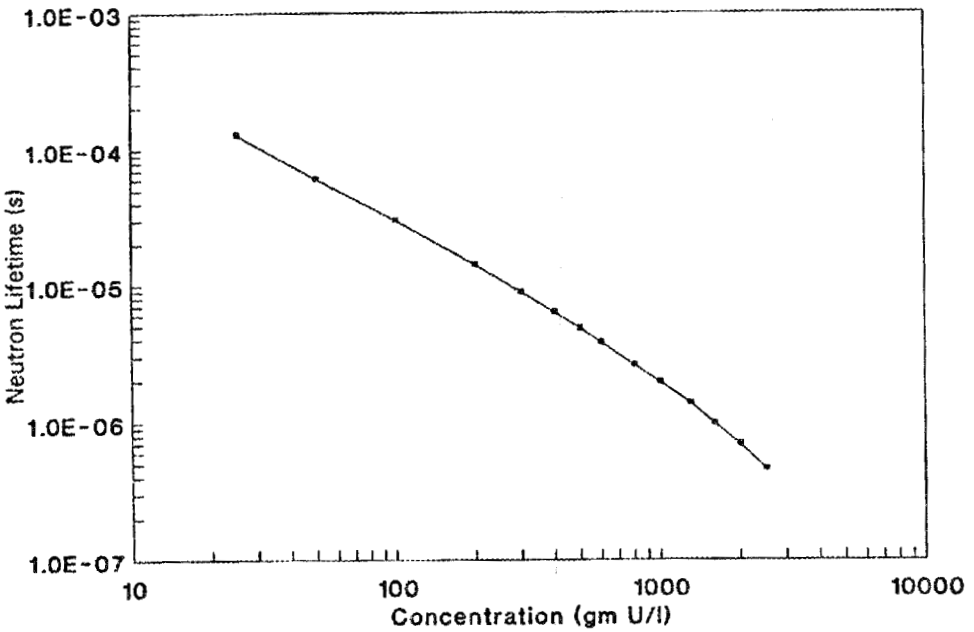


Figure 21 Neutron lifetime as a function of uranium concentration/density

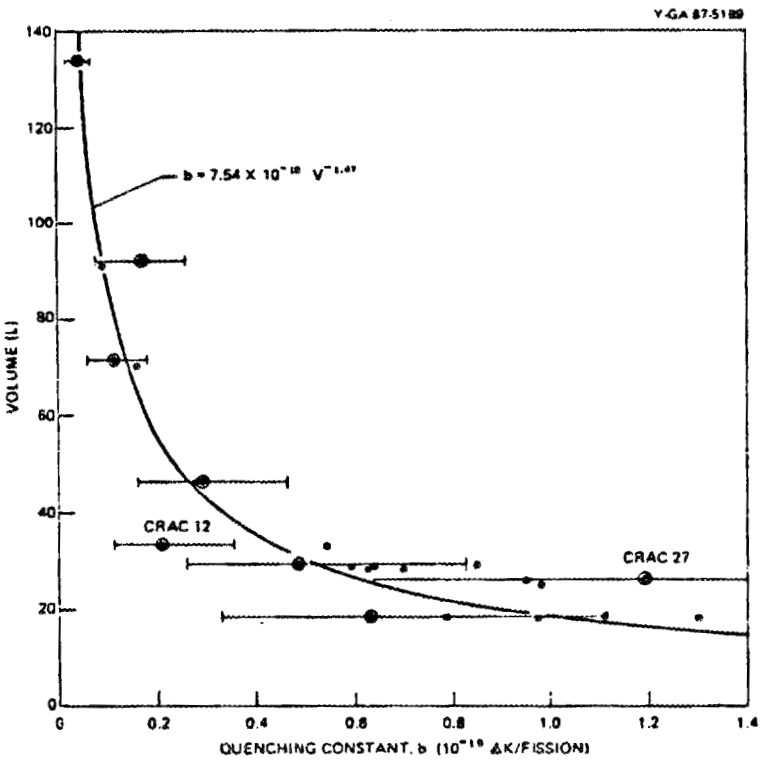


Figure 22 Values of quenching constant b vs solution volume V for CRAC experiments, and least-squares curve fit giving b as a function of V

6.2 COMPARISON WITH MEASUREMENTS

In order to judge the validity of the various approximations inherent in the methods described above, the fission yield tool was applied to the CRAC experiments. The 24 CRAC experiments considered covered a limited range of the parameters with HEU solutions of volumes from 19 to 134 L, tank diameters of 80 (31.52 in.) and 30 cm (11.82 in.), reactivity insertion rates of 0.014 to 0.786 $/s, and densities of 30.6 to 320 grams of U/L. While the full range of desired parameter space is not covered by these measurements, it is felt that the performance over these ranges should be indicative of the expected accuracies over small extensions of the parameter space. For low-enriched uranium (LEU) solutions, very little experimental information exists, and validation of the 5 wt % portion of the fission-yield tool is difficult. However, the results from calculations for LEU solutions with similar parameter variations can be used to increase the confidence in the results for LEU solutions.

Comparisons of the fission-yield predictions for the various CRAC experiments is shown in Table 3. The results from Ref. 17 where the statistical nature of fission yields is accounted for can be used to gauge the goodness of the agreements for these systems. The results from this previous study indicated that variations of a factor of about 1.7 from the average yield satisfactorily enveloped the expected statistical variations in the initial burst fission yields. The results shown in Table 3 fall within this range quite well.

While the amount of fission-yield data for validation of computations with HEU solutions is quite extensive, very little data exist for LEU systems. For testing purposes, the results from a series of calculations[19] using a 1-D coupled hydrodynamic-neutronic code were compared to the LEU results generated using the models described herein. The results are given for a few limited cases in Table 4. The agreement shown is typically within a factor of 2 to 3, which is quite good considering the approximations in the models presented here.

Table 3. Comparison of actual vs predicted first-pulse fission yields for CRAC experiments

CRAC exp. No.	Actual Yield 10^{16} fissions	Predicted yield 10^{16} fissions	Ratio actual to predicted[a]
5	6.3	4.8	1.3
6	6.7	5.1	1.3
7	4.0	3.9	1.0
8	8.0	3.8	2.1
9	4.4	6.4	0.7
10	4.3	3.6	1.2
12	3.7	1.6	2.3
13	5.2	5.1	1.0
14	4.0	4.0	1.0
19	3.5	3.2	1.1
20.4	5.9	3.5	1.7
21	3.1	3.2	1.0
22	4.2	3.1	1.4
23	4.2	5.2	0.8
25	3.9	3.2	1.2
26	3.9	4.2	0.9
27	3.5	5.9	0.6
28	3.7	5.0	0.7
29	3.3	3.1	1.1
39	18.0	12.6	1.4
40	13.0	19.9	0.7
41	8.0	6.5	1.2
42	17.0	9.7	1.8
43	13.0	10.2	1.3

[a] These ratios are to be compared with a predicted factor of 1.7 due to the statistical fluctuations in the fission yield magnitudes predicted in Ref. 17.

Generation

Table 4. Comparison of LEU fission yield predictions

Ref. 19 yield estimate 10^{18} fissions	Yield estimate this work 10^{18} fissions	Case description
70.0	43.0	2.5 wt % water, 10 $/s insertion rate
8.0	2.6	5.0 wt % water, 10 $/s insertion rate
1.2	1.0	10 wt % water, 10 $/s insertion rate

7 FISSION YIELD RESULTS

First-pulse fission yields for fully water-reflected supercritical systems were estimated using the above theory for a series of parametric variations in LEU nitrate solutions and oxides and HEU nitrate solutions that are representative of a large number of potential applications. Presentation of the results in a graphic format allows for easy interpolations within the individual parameter space and limited extrapolations outside the parameter ranges when the trends appear to be smooth. Graphic presentations of the computed fission yield estimates are provided in Figs. 23 and 24 for LEU systems and in Figs. 25 and 26 for HEU systems. One of two characteristic cylindrical-tank types (i.e., vertical-axis tank diameter or horizontal-axis tank length) is provided in each figure. This manner of presentation allows for quick fission-yield estimates (maximum or minimum) for different assumed values of water-reflected and water-moderated oxide or solution uranium densities, and critical volumes at a 45-gal (170.33-L)/min material addition rate. The estimated fission yields, as influenced by solution or damp oxide addition rates between 0.01 gal (0.038 L) and 200 gal (757 L)/min, are scaled for each of the four graphs to permit interpolation of the estimates.

Each graph provides four parameters: (1) cylindrical tank dimension in inches, either a vertical cylinder diameter or a horizontal cylinder length, (2) uranium density in grams of uranium per liter (ounces/gallon), (3) critical fissile material volume in gallons, and (4) first pulse fission yields that are representative of a fissile material addition rate of 45 gal (170.33 L)/min. Each fission yield graph may be scaled with a common fifth parameter, fissile material addition rate, that is provided on the functional slide rule. In Fig. 23 the user may estimate the critical volume of 5 wt % LEU solution at 600 g U/L in a 60-in. (152.4-cm)-diam tank to be about 160 gal (605.6 L). If fissile material were to be introduced into such an empty tank at 45 gal (170.33 L)/min (the assumed rate for the graph), it would take about 3.5 min to attain criticality, and the estimated first-pulse fission yield for such a criticality would be 4×10^{18} fissions. Because the first-pulse fission yield is not directly proportional to the fissile material addition rate, it is necessary to use the addition rate (gal/min) scaling graph to make first-pulse fission yield estimates for addition rates other than 45 gal (170.33 L)/min.

The results clearly show the trends toward larger fission yields for larger systems (resulting in greater volumes of material with smaller quenching constants), lesser densities (resulting in smaller intrinsic neutron source values per unit volume), and higher reactivity insertion rates (as influenced by solution reactivity worth and geometric-change reactivity worth). It can be observed that the graphs predict very large first-pulse fission yields for large system volumes and rapid material addition rates. It is appropriate to consider restraint in predicting first-pulse solution fission yields much in excess of a 5×10^{18} fission yield. Such a fission yield was observed, but from an intentionally designed, extremely large and rapid reactivity insertion in the destructive BORAX-I experiment accident.[20] Though this experiment was performed with plate-type material test reactor (MTR) aluminum clad fuel elements, the neutronics and radiation heating of the water moderator is much like a somewhat undermoderated uranium solution.

Barbry[4] provided the following empirical equation for estimating the total fission yield of a water-moderated critical system that remains critical for some time duration.

The reported metric unit version of the equation is

$$N_f(t) = \frac{V(\text{liters}) \times t(\text{secs})}{(3.55 \times 10^{-15}) + [(6.38 \times 10^{-17}) \times t(\text{sec})]},$$

where N_f (t) is given in fissions as a function of time, t seconds, and V is given in liters of solutions. The English unit version of the equation is

$$N_f(t) = \frac{V(\text{gallons}) \times t(\text{minutes})}{(1.563 \times 10^{-17}) + [1.686 \times 10^{-17}) \times t(\text{minutes})]}.$$

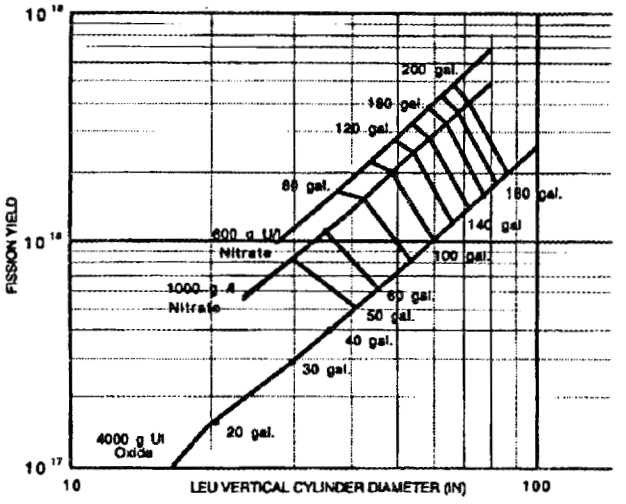


Figure 23 LEU vertical cylinder first-pulse fission yield estimate

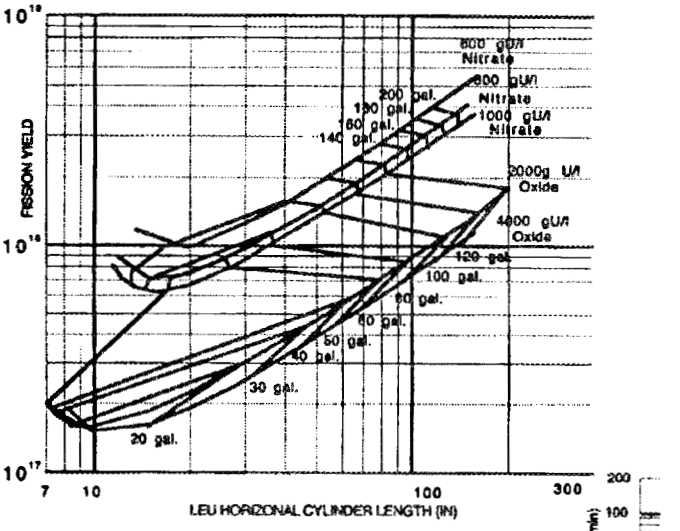


Figure 24 LEU horizontal cylinder first-pulse fission yield estimate

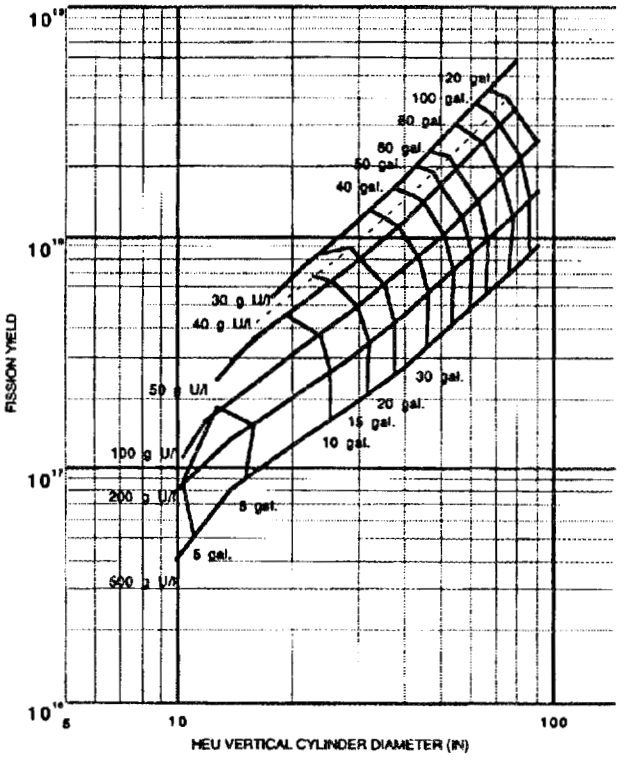


Figure 25 HEU vertical cylinder first-pulse fission yield estimate

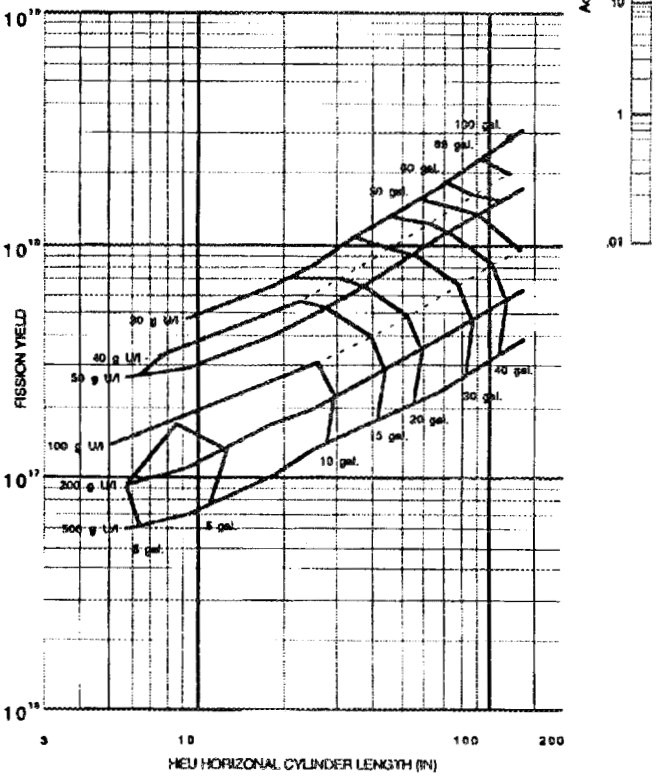


Figure 26 HEU horizontal cylinder first-pulse fission yield estimate

8 SUMMARY/CONCLUSIONS

This work has updated the results of the original slide rule published in limited form in the early 1970s. The general format and features of the original slide rule are retained, in that the various curves include a prompt dose vs distance relationship, a fission-product gamma dose rate vs distance and time relationship, a total dose vs time and distance relationship, and a 1-min total dose vs time and distance relationship. The original slide rule consisted of only two system types, highly enriched uranium solutions and metal, and contained a number of approximations, namely an assumed inverse-square relationship of neutron and gamma-ray doses with distance. The newly updated slide rule contains information for the following five systems:

1. unreflected sphere of 4.95 wt % enriched aqueous uranyl fluoride, $U(4.95)O_2F_2 \cdot H_2O$, solution having a hydrogen-to-^{235}U ratio of 410 (solution density = 2.16 g/cm^3),

2. unreflected sphere of damp 5 wt % enriched uranium dioxide, $U(5)O_2$ having a hydrogen-to-^{235}U ratio of 200,

3. unreflected sphere of 93.2 wt % enriched uranyl nitrate, $U(93.2)O_2(NO_3)_2 \cdot 6H2O$, solution having a hydrogen-to-^{235}U atom ratio of 500 (solution density = 1.075 g/cm^3),

4. unreflected sphere of 93.2 wt % enriched uranium metal sphere (metal density = 18.85 g/cm^3), and

5. unreflected sphere of damp 93.2 wt % enriched uranium oxide, U_3O_8 plus water, having a hydrogen-to-^{235}U atom ratio of 10 (uranium oxide density = 4.15 g/cm^3).

The new slide rule also includes not only the air/ground interface effects near the assumed accident, but out to 4000 ft (1219.2 m) as well. The possibility of a shielded criticality in which skyshine radiation can be important is also treated, with the inclusion of a separate skyshine contribution as a function of distance from the accident. Also, results of first-pulse fission yield estimate evaluations are presented as functions of vertical or horizontal cylindrical critical volumes (based upon the degree of fissile material moderation expressed in terms of uranium density and cylinder dimension) and material addition rates. The first-pulse fission yield estimates may then be used for determining appropriate mitigating measures for protection of personnel as an uncontrolled system approaches criticality.

Though the presentation of dose and dose rate information for less than one minute (i.e., 1 to 60 elapsed time in seconds) following the initial or prompt fission yield has no use for emergency response, the information is useful for emergency preparedness in the training of personnel to quickly respond to a criticality accident alarm or in the estimation of radiation fields at time of the accident or very shortly thereafter.

Included as an Appendix to this report is a suggested input for the NRC publication entitled "Response Technical Manual" (RTM). This section is included to allow for explanatory information regarding the use of the slide rules developed in this work. The specific aim of the RTM input is to make the slide rule more useful in emergency response situations.

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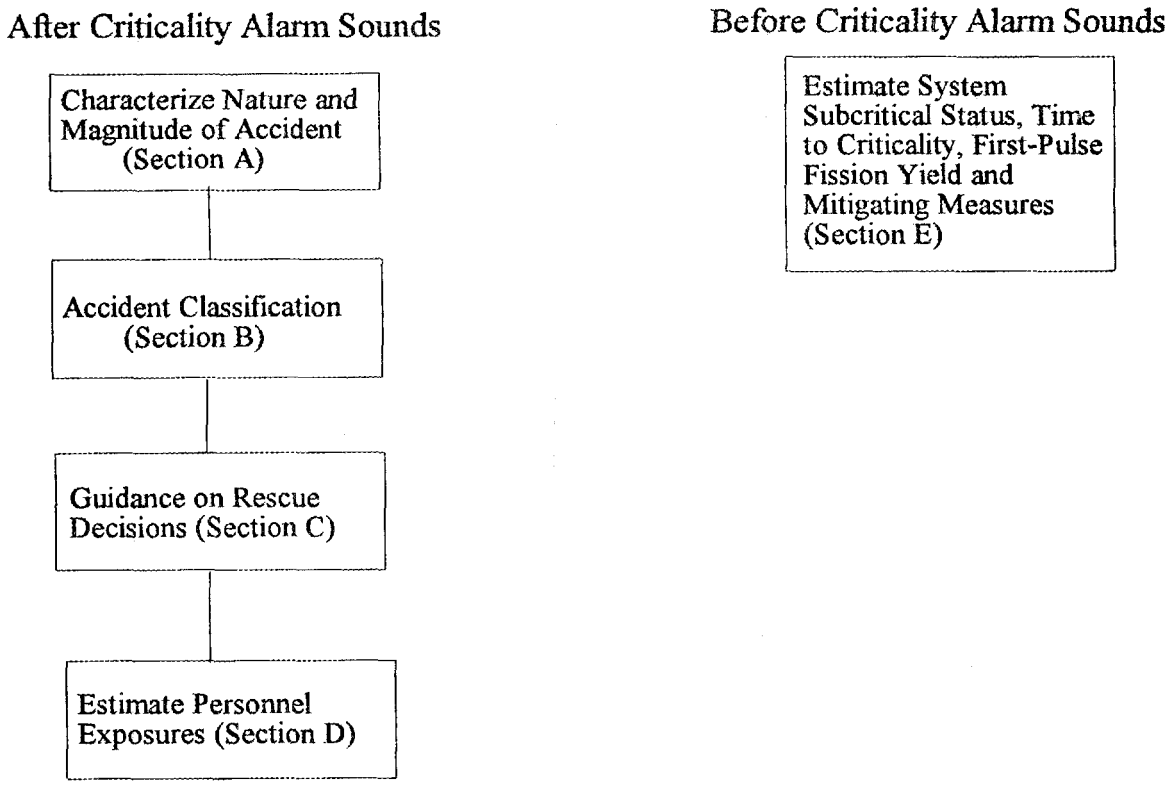
APPENDIX

INPUT FOR RESPONSE TECHNICAL MANUAL

This appendix is the suggested input for the NRC publication entitled "Response Technical Manual" (RTM). Because the updated slide rule described in the body of the report is an essential tool in the RTM procedures, this appendix is provided as an adjunct to the use of the slide rule. This RTM slide rule tool is designed to be useful in the field, as well as emergency centers, for responding to emergency situations involving nuclear criticality accidents. The use of these procedures with slight modifications can also be useful as a planning tool. In this scenario, the predicted fission yields from Step 3 of Section A can be used to postulate a nuclear criticality accident, and the slide rule could be used to estimate radiation levels and used as "detector readings" in the simulated accident. With this information, the entire RTM exercise could be carried out as a drill.

The slide rules referred to in this section as slides 1–6 are contained in Volume 2 of this document.

Chart 1. Fuel cycle and material facilities criticality accident assessment



Section A
Characterize Probable Accident Conditions

Purpose

To assess the likely physical size, location, magnitude or yield, and the ongoing nature of the criticality accident.

Discussion

The primary focus of this section is to guide you through an estimate of the magnitude and duration of the criticality accident. This information will be used in other sections to plan possible rescue operations, arrange intervention scenarios (if necessary) for the termination of the accident, and estimate personnel exposures from the accident.

These steps will be required in this assessment:

Step 1: Estimate material type involved and whether accident is terminated. Step 2: Establish monitoring activities for radiation levels. Step 3: Estimate yield magnitudes from the system physical information. Step 4: Confirm yield data with monitoring data. Step 5: Continue monitoring activities

Step 1

In order to properly assess the magnitude of the accident and predict dose rates over time, it is necessary to select a fissile material type that is likely involved in the criticality accident. Please answer the following questions to the best of your ability.

1. Which of the following materials is most likely the criticality accident material? (Check one.)

   _____(1) Low (< 6 wt %) enriched uranium solutions (select slide 4)
   _____(2) Low (< 6 wt %) enriched uranium damp compounds (select slide 5)
   _____(3) High (> 80 wt %) enriched uranium solutions (select slide 1)
   _____(4) High (> 80 wt %) enriched uranium metal (select slide 2)
   _____(5) High (> 80 wt %) enriched uranium damp compounds (select slide 3)

2. Has the criticality accident terminated? How do you know?

3. Does the accident appear to be pulsing (i.e. multiple bursts typically of smaller and smaller magnitude)? (See examples of various criticality dose rate traces in Figs. A.1 and A.2)

4. Based on the accident types in Figs. A.1 and A.2, does the accident appear to be a single pulse, or a multiple-pulse accident?

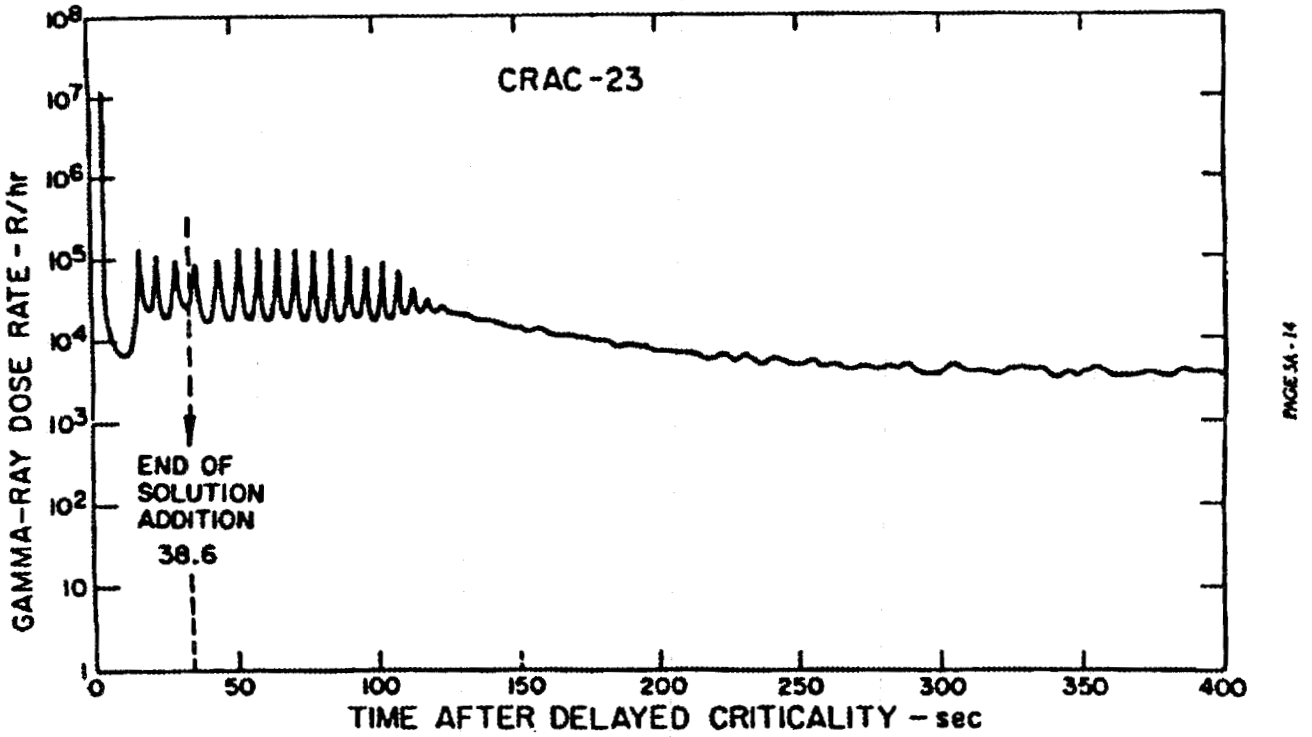


Figure A.1 Gamma-ray dose rate as a function of time for multiple-pulse system

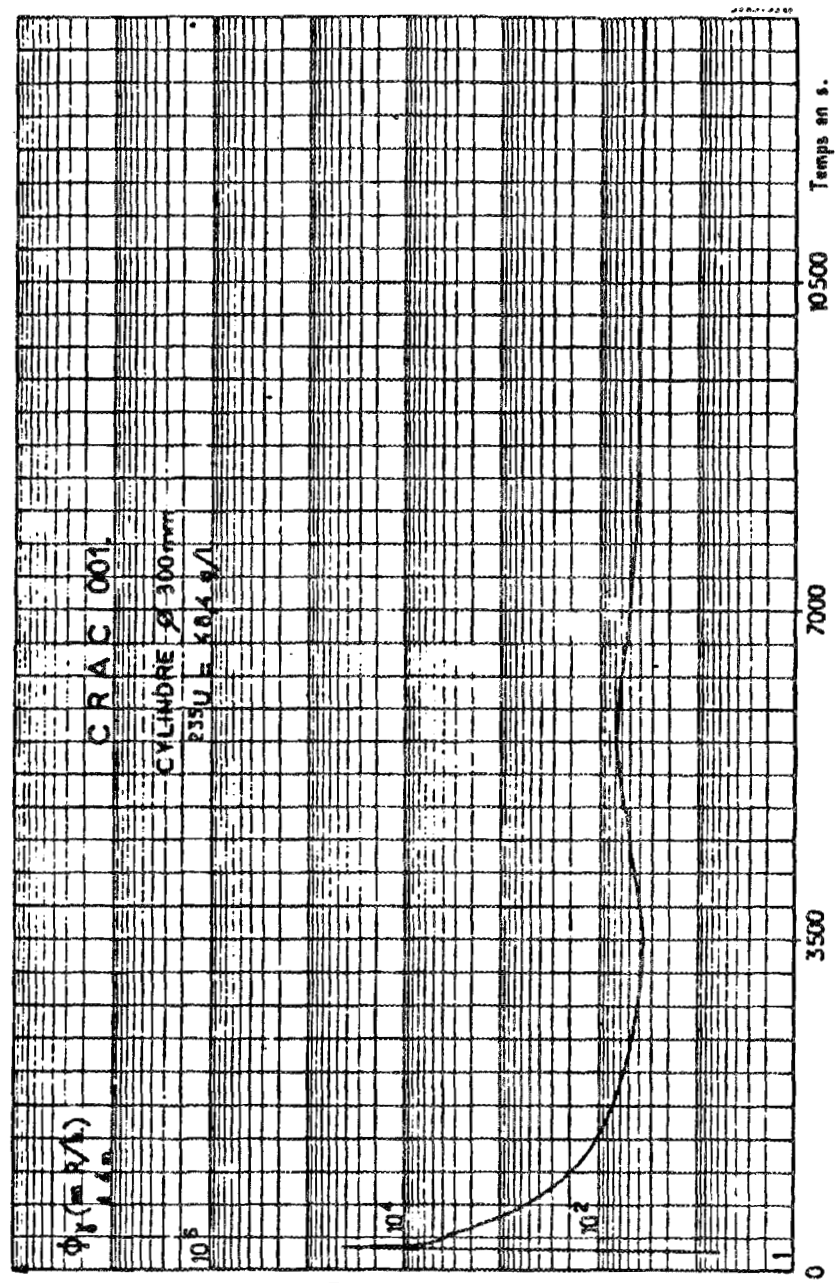


Figure A.2 Gamma-ray dose rate as a function of time for a single pulse system

Step 2

In order to locate the accident, a survey of the available criticality accident alarm system radiation monitors should be taken. Rapidly increasing levels (possibly readings off-scale) on most, if not all, monitors may be expected during the initial burst from the accident, followed by an abrupt decrease in radiation monitor readings. Typically, within a few seconds of the initial burst a gradual decrease in the radiation monitor levels is observed. This behavior of the radiation monitors can be used to approximate the location of the criticality accident. Monitors near the accident should show the largest radiation levels due to the typical inverse square relationship between dose and distance. A number of anomalous situations should be considered to determine if the monitors are functioning accurately prior to their use in the procedures that follow. A GM-type detector can be saturated by the initial burst; therefore the readings can immediately fall erroneously to zero. A sodium iodide (NaI) detector very near the accident can become activated and continue to indicate elevated levels for some time after the accident. Typically neutron detectors are fairly well behaved and should provide reliable radiation levels for the duration of an accident; however, neutrons are normally present only during and briefly after an actual criticality accident burst.

Use the table below to record radiation monitor readings from two to three locations exhibiting the largest radiation levels after the initial burst. Enter the radiation monitor readings as soon as possible after the initial burst or when the readings come back on scale.

Time of criticality alarm: AM, PM

(1) Time after alarm (min)	(2) Assumed distance from accident (ft)	(3) Assumed shielding thickness (in.)	(4) Measured dose rate (rad/h)	(5) Estimated fission yield (see Step 4)

Pick the monitor(s) with the highest reading(s) and plot at least 5 points from over a one-hour time frame. Compare this plot or an actual monitor strip chart output with the accident graphs shown in Figs. A.1 and A.2.

Select the accident mode which more closely resembles the plot from your accident readings.

Accident mode is: Figure A.1 - Pulse mode _____
Figure A.2 - Single mode _____.

Step 3

In this step you will use information about your system in order to estimate approximate yields for your system. These estimated yields will be compared in Step 4 to another method of estimating yields in order to arrive at a best estimate of the magnitude of the accident.

Please fill in your best estimate of the following system parameters. Guidance is given for each as well as default or typical values should you be unable to obtain a parameter.

(1) What is the probable system density of uranium in g-U/L? _____
(For low-enrichment systems choose between 600 and 4000 g-U/L, for high-enrichment systems choose between 30 and 500 g-U/L. The smaller densities should produce high yield estimates.)

(2) Characterize the uranium enrichment as HEU or LEU _____
(HEU has enrichments > 80%, LEU has enrichments < 6%)

(3) What is the expected fissile material feed rate in gal/min? _____
(Choose in the range 0.1 to 200 gal/min. The higher feed rates give higher estimates of yields.)

(4) What is the diameter (for a vertical tank) or the length (for a horizontal tank) of the suspected criticality system in cm? _____

For the selected system parameters given above, read the system yield from the appropriate yield curves shown in slide 6. Record value below:

Estimated yield based on system characteristics: _____ fissions.

Step 4

This step will use the radiation monitoring data from Step 2 to obtain an independent estimate of the accident fission yield obtained in Step 3. The accompanying slide-rule figure selected in step 1 allows the analyst to correlate measured dose or dose rate information at a specific time and location to the estimated fission yield of the accident. For the typical radiation monitor that measures the dose rate, the curve used to correlate these measurements with the projected yield is the ESTIMATED FISSION YIELD BASED ON DISTANT GAMMA DOSE RATE AND ELAPSED TIME curve. To read the slide rule, the slide-rule graph is moved until the specified time after burst, location from the burst, and the measured dose rate align. The estimated fission yield is then read from the small graph labeled ESTIMATED FISSION by observing the location of the arrow.

This process should be repeated for some four to six measurements and the resulting fission yields averaged. Use the table in Step 2 to record your results.

Enter the resulting average fission yield: _____ fissions.

Compute 0.5 × (step 3 yield) + 0.5 × (step 4 yield) = _____ final yield estimate.

Step 5

After the initial estimation of the fission yield is complete, it should only be necessary to record the radiation monitor reading periodically to ensure the continuing decrease of residual radiation from the accident. It is suggested that monitor readings only need to be recorded perhaps every 1 to 2 hours at this phase.

Section B
Classification Assessment

Purpose

To verify the licensee's classification of the accident.

Discussion

This section provides methods for determining the appropriate classification of an accident at a nuclear power reactor or at a fuel cycle or material facility.

Differences in classification should be discussed with the licensee only if there is a clear conflict in classification. Questioning the licensee in other cases could slow the accident response.

Step 1

Assess the classification of the accident using one of the methods below. The method chosen will depend on the type of facility and the classification method the facility uses.

Reactor accident
- NUREG-0654 quick assessment..... Method B.1
- NUREG-0654 full guidance..... Method B.2
- NUMARC/NESP-007 assessment (barrier approach)..... Method B.3

Fuel cycle and material facilities accident..... Method B.4

END

Sources: NUREG-0654, NUMARC/NESP-007

Method B.4
Fuel Cycle and Material Facilities Classification Guidance

Purpose

This method is used to assess the classification of an accident at a fuel cycle or material facility.

Discussion

Emergency plans for fuel cycle and material facilities are not yet standardized. As licenses are renewed at facilities requiring emergency plans, a standardized classification system will be adopted. Some facilities do not have emergency plans because of the small quantity of material they handle. These classification descriptions would not apply to the facilities that do not have emergency plans.

Step 1

Use the classification descriptions in Table B-8 to determine the emergency classification of the accident. Note that there are no Unusual Event or General Emergency classifications for non-reactor facilities.

Step 2

Compare the classification with the licensee's classification. If the licensee's classification does not appear to be correct, review the licensee's classification procedure before discussing your finding with the licensee. Resolve any differences in the interpretation of the plant conditions.

Step 3

If, after attempts to resolve any differences, it appears that the licensee is potentially underclassifying a General Emergency, ask the licensee to reevaluate.

Step 4

If the classification is determined to be General Emergency, assess protective actions using Section G.

END

Table B-8. Event classification for fuel cycle and material facilities

Class/Description	Offsite Consequences	Anticipated Responses
Alert		
Events may occur, are in progress, or have occurred that could lead to a release of radioactive material but the release is not expected to require a response by offsite response organizations to protect persons offsite.	Possible minor releases well below EPA PAG exposure levels. Environmental sampling and some offsite monitoring may be required.	Licensee emergency response personnel secure operations, stop any releases and perform monitoring. State and local organizations notified, inspectors dispatched. Fire department, ambulance and law enforcement respond as required to support onsite response. NRC notified, Regional Operations Center activated and inspectors or site team dispatched. HQ may activate Operations Center. DOE medical support and/or monitoring may be requested.
Site Area Emergency		
Events may occur, are in progress, or have occurred that could lead to a significant release of radioactive material and could require a response by offsite response organizations to protect persons offsite.	Significant release possibly approaching EPA PAG exposure levels. Radiation and contamination levels may require restricting areas offsite. Environmental sampling and offsite monitoring required.	Licensee emergency response personnel secure operations, stop the release, perform monitoring and regain control of radioactive material. State and local organizations notified, emergency personnel respond to site, assess situation, assist monitoring activities and advise the public as required. Fire department, ambulance and law enforcement personnel respond to mitigate consequences, restrict public access to affected areas and support onsite response as required. NRC notified, Operations Center activated and site team dispatched. DOE monitoring support requested. DOE medical support may be requested if required.

Section C

Guidance on Rescue Decisions During and After an Accident

Purpose

To determine when and if a rescue can be attempted and if so how long the rescuer has to complete the rescue based on the regulatory dose limits.

Discussion

This guidance should allow for planning of rescue operations. Once you have estimated the travel time to the rescue, estimates should be made of the rescue time. The slide rule can be used in this section to predict the accumulated dose for a series of 1-min time periods at various times after the accident and various locations from the accident. These 1-min doses can then be used along with the rescue time estimates to predict the total dose due to a rescue at various times.

Step 1

In column 5 enter the total time, in minutes, for the rescue team to travel through the high radiation area to arrive at the rescue scene.

In column 2 enter the average distance, in feet, between rescue personnel and the accident site during travel to the rescue scene.

In column 7 enter the estimated time needed for rescue, in minutes.

In column 3 enter the distance from the rescue scene to the accident in feet.

Step 2

In column 1 enter the desired range of rescue initiation times, in minutes, after the criticality alarm.

These times represent possible starting points for the initiation of a rescue. Early after the accident, doses may be too large to justify rescue initiation.

To arrive at the total dose for travel (column 4) and rescue (column 6), the slide rule selected in Section A, step 1, and the curve entitled ACCUMULATED ONE MINUTE DOSE (RADS) BASED ON ESTIMATED FISSION YIELD, DISTANCE FROM INCIDENT, AND TIME OF ENTRY AFTER ACCIDENT are used.

The total dose for travel, TDT, is obtained by:

(a) along the abscissa locate the ELAPSED TIME corresponding to the rescue initiation time in column 1;

(b) vertically scan upward until the effective travel location (column 2) distance is reached;

(c) the one-minute dose for travel, OMDT, value is then read from the plot (record value below);
(d) record travel time, TT, below and calculate TDT.

TDT = OMDT•TT = (_____)•(_____) = _____ rads (record in column 4)

The total dose for rescue, TDR, is obtained using the same slide-rule curve as described above, as follows:

(a) combine the rescue initiation time, RIT, and the travel time, TT, to arrive at arrival time of rescue, ATR;

ATR = TT + RIT = (_____) + (_____) = _____ minutes

(b) locate along the abscissa the ELAPSED TIME corresponding to ATR in minutes;

(c) vertically scan upward until the rescue location (column 3) distance is located;

(d) the one-minute dose for rescue, OMDR, value is then read from the curve (record below);

(e) record rescue time, RT, below and calculated TDR.

TDR = OMDR•RT = (_____)•(_____) = _____ rads (record in column 6)

Column 8, total dose estimate, is obtained by summing columns 4 and 6.

(1) Rescue initiation time (min)	(2) Effective travel location (ft)	(3) Rescue location (ft)	(4) Dose estimate for travel (rads)	(5) Travel time (min)	(6) Dose estimate for rescue (rads)	(7) Rescue time (min)	(8) Total dose (rads) estimate

Step 3

Based on a comparison of the doses as a function of rescue initiation time given in the table above and the following allowable doses for various rescue scenarios, determine an appropriate rescue initiation time for your situation.

Actions to save a life - 25 rads
Actions to protect facility - 10 rads

Section D

Determination of Exposures to Personnel

Purpose

To estimate the radiation doses received by plant personnel as a result of a nuclear criticality accident for early diagnostic and archival purposes.

Discussion

The individual personnel exposures are dependent upon the total fission yield as determined in Section A, their location at the time of the accident, the amount of shielding present, the time of the exposure, and the length of the exposure. The table supplied below should allow for collection of the appropriate data to estimate the doses for all personnel exposed to the accident.

Step 1

Fill in the table below as follows.

Column 1 - Self explanatory
Column 2 - Indicate the time of initial exposure (in minutes since the initial burst)
Column 3 - Record the length of exposure in minutes
Column 4 - Estimate the distance the individual was from the accident in feet
Column 5 - Estimate and enter the neutron dose reduction factor by noting the approximate amount of shielding material (in inches and by shield type) between the accident and the exposed individual. See the dose reduction curves given in Fig. A.3 to estimate the dose attenuation factor for a given thickness of shielding material.
Column 6 - Estimate and enter gamma-ray dose reduction factor (see column 5 directions).

(1) Name	(2) Time of initial exposure (min)	(3) Duration of exposure (min)	(4) Distance from accident (ft)	(5) Neutron dose reduction	(6) Gamma-ray dose reduction	(7) Estimated neutron exposure (rads) (see Step 2)	(8) Estimated gamma-ray exposure (rads)	(9) Estimated total dose (rads)

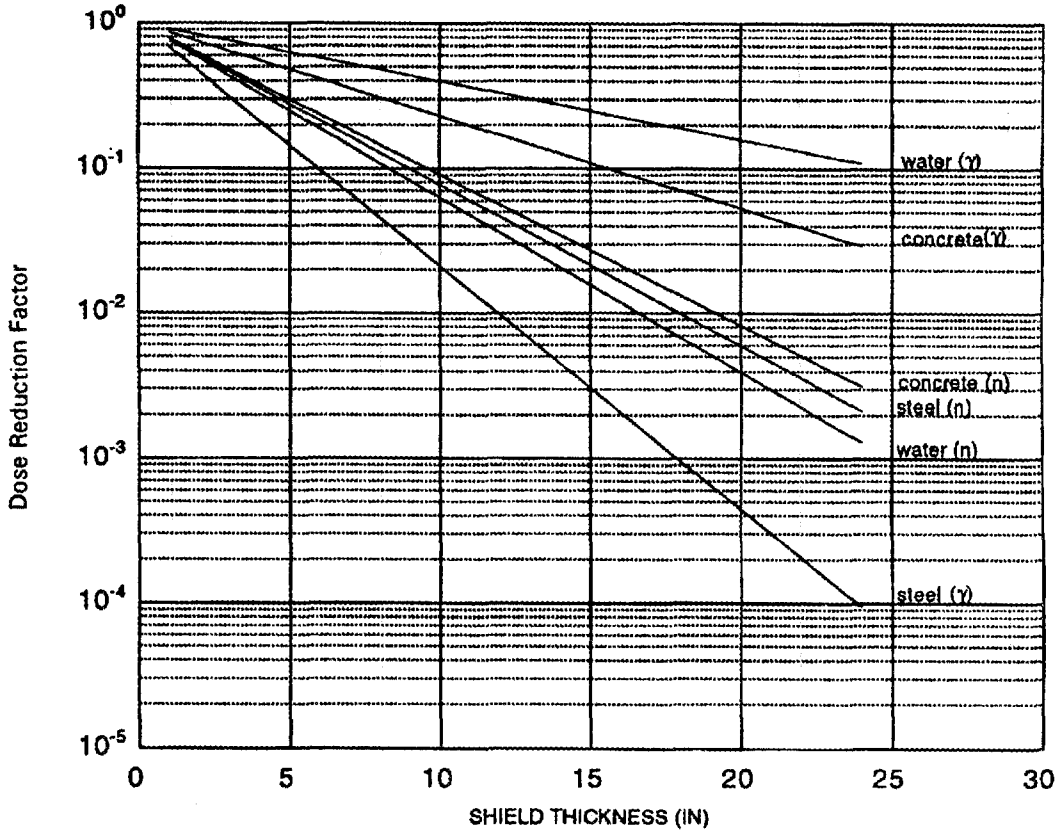


Figure A.3 Dose reduction for various shields

Step 2

Using the INTEGRATED DOSE (RADS) BASED ON ESTIMATED FISSION YIELD, DISTANCE FROM INCIDENT AND TIME curve from the slide rule selected in Section A, step 1, set the fission yield at the value obtained in Section A, and read the total dose-unshielded (TDU) corresponding to the recorded starting and duration times and the location given in the table above for each exposed individual. This dose assumes that no shielding was present between the accident and the exposed individual; if this is the case, the slide-rule result can be entered directly in the table above under ESTIMATED TOTAL DOSE (column 9). If shielding was present, the dose attenuation factor must be included as described below to account for the dose reduction due to shielding.

(1) The total dose-unshielded, TDU, should be recorded below.

(2) The neutron dose-unshielded, NDU, should be read from the curve labeled "neutron (N)" on the plot entitled ESTIMATED PROMPT DOSES BASED ON TOTAL FISSION YIELD, AND DISTANCE FROM INCIDENT. The distance should correspond to the entry in column 4 for each individual. (record below)

(3) The values of neutron dose reduction, NDR (column 5), and gamma-ray dose reduction, GDR (column 6), should be recorded below.

(4) Perform the following calculations for the gamma dose-shielded, GDS, neutron dose-shielded, NDS, and total dose-shielded, TDS:

GDS = (TDU – NDU)GDR = (_____ – _____) _____ = _____ rads
(record in column 8)

NDS = NDU•NDR = _____ • _____ = _____ rads
(record in column 7)

TDS = GDS + NDS = _____ + _____ = _____ rads
(record in column 9)

Section E

Estimate System Subcritical Status, Time to Criticality, First-Pulse Fission Yield and Mitigating Measures

Purpose

To assess the subcritical status of a fissile material solution or damp oxide system, to estimate the time for the uncontrolled approach to criticality, to estimate the first-pulse fission yield from the uncontrolled approach to criticality, and to provide information for taking mitigating protective measures prior to expected criticality.

Discussion

The evaluation sequence of this section is provided to interpret slide 6 for estimating the fissile material solution or oxide system subcritical status, to estimate the time to fissile material system criticality, to estimate the magnitude of the first-pulse fission yield and to provide input for consideration in the development of mitigating measures. The tools of this section specifically apply to fully water-reflected aqueous solutions of homogeneous high-enriched uranium (~93 wt % ^{235}U) nitrates and aqueous solutions of homogeneous low-enriched uranium (~5 weight percent ^{235}U) fluorides and dioxides in vertical or horizontal cylindrical geometries. However, these tools may be applied to other minor variations in uranium enrichments (i.e., ≥ 80 wt % ^{235}U or ≤ 6 wt % ^{235}U) and solution compounds and oxides with relatively minor variances in estimated results. Though the fissile material systems were evaluated as fully water-reflected systems, the estimated first-pulse fission yields determined in this section may be used as the input fission yield for interpretation of information using slides 1, 3, and 5 of the nuclear criticality slide rule.

Even though a nuclear criticality has not occurred, a fissile material solution or damp oxide system may have lost the necessary nuclear criticality safety controls to prevent criticality. An example could include the loss of process controls thereby permitting the discharge of unsafe concentrations/densities of fissile material solution into an unfavorable geometry vessel. Another example could include the loss of moderation-control containment of dry uranium oxide thereby permitting the sorption of water or oils (neutronically similar to water) into a deposit of dry oxide. Even though the sorption of water or oils into a dry oxide deposit is somewhat antithetical of adding fissile material solution to an unfavorable geometry, the adding of water or oils to an existing subcritical geometry adds nuclear reactivity, via neutron moderation, similar to adding nuclear reactivity via the addition of fissile material solution to an unfavorable geometry.

The following information is required to perform the estimates of this section. Record the required information below.

Enrichment of the solution or oxide: _____ **weight percent of** ^{235}U in total uranium.
Density of uranium in the solution or oxide: _____ **grams of uranium per liter**.
Anticipated critical geometric configuration and dimension: (complete one)
Vertical cylinder of approximate fixed dimension equal to _____ **inches in diameter**, or
Horizontal cylinder of approximate fixed dimension equal to _____ **inches in length**.
Addition rate of fissile material equal to _____ **gal/min**.
Existing volume of fissile material in the anticipated critical geometric configuration equal to _____ **gal**.
Date ___/___/___ and **24-hour clock time** ___:___ of current volume estimate.

Conversion factors and equalities are provided at the end of this section to assist in the conversion of units to match the required units of these estimates.

The following steps are required to estimate the subcritical status of fissile material collecting into a critical geometry, to estimate the time to accumulate a critical geometric volume of material, to estimate the first-pulse fission yield of such a critical system, and to apply the information in the development of mitigating measures prior to the criticality.

Step 1: Select fissile material (FM) uranium enrichment and form.
Step 2: Select anticipated critical geometric configuration (ACGC).
Step 3: Select appropriate first-pulse fission (FPF) yield figure to use for estimates.
Step 4: Record uranium density anticipated to become critical.
Step 5: Record existing volume (EV) of fissile material in the ACGC.
Step 6: Determine the estimated critical volume of the ACGC.
Step 7: Record the FM addition rate (AR) into the ACGC.
Step 8: Determine time to accumulate enough FM to become critical.
Step 9: Determine preliminary first-pulse fission (PFPF) yield at plotted value for 45 gal/min FM AR.
Step 10: Determine estimated first-pulse fission (FPF) yield according to a FM AR fission yield multiplier.
Step 11: Apply the estimated FPF yield as the "ESTIMATED FISSION" input value in the appropriate dose curves of the criticality slide rule.
Step 12: Define appropriate mitigating measures to cope with anticipated criticality accident.

Step 1: Select fissile material (FM) uranium enrichment and form.

What enrichment and form of material best describes the fissile material that is accumulating into an uncontrolled geometric configuration? (Check one)

____ Low (< 6wt%) enriched uranium solutions or damp oxides
____ High (> 80 wt%) enriched uranium solutions

Step 2: Select anticipated critical geometric configuration (ACGC).

Judgments will likely be required as to the geometric approximation of the approaching critical condition of fissile material. Fissile material solution collecting in a square bottomed floor sump may be approximated as a vertical-axis cylinder having a diameter that provides an equivalent horizontal cross-sectional area to that of the floor sump. Limited length troughs or ditches may be approximated as a horizontal-axis cylinder of equivalent length. Systems that have primary dimensions (i.e., diameter or length) that are larger than those provided in Figs. 23 through 26 may be approximated via extrapolation of the curves; however, the estimated fission yield undetermined errors will become increasingly larger.

Which geometric configuration best describes the ACGC? (Check one and record its diameter in inches)

____ Vertical-axis cylinder having a diameter of _____ inches.
____ Horizontal-axis cylinder having a length of _____ inches.

Step 3: Select appropriate first-pulse fission (FPF) yield figure to use for estimates.

Based upon answers to steps 1 and 2, select the appropriate figure to use for FPF yield estimates. (Check one)

____ **Low-enriched** uranium solution or oxide in a **vertical-axis** cylinder, Figure 23.
____ **Low-enriched** uranium solution or oxide in a **horizontal-axis** cylinder, Figure 24.
____ **High-enriched** uranium solution in a **vertical-axis** cylinder, Figure 25.
____ **High-enriched** uranium solution in a **horizontal-axis** cylinder, Figure 26.

In the following guidance provided in steps 4 through 9 references to the "figure" are specific to the figure selected in this step.

Step 4: Record uranium density of solution being introduced into a vessel.

What is the uranium density of the solution or oxide that is being introduced into the vessel? (Record the value below in units of **grams of uranium per liter**)

Density, _____ grams of uranium per liter.

Examine the figure selected in *Step 3* above for the location of uranium density curves relative to the density recorded above.

Step 5: Record existing volume (EV) of fissile material in the ACGC.

What is the existing volume of fissile material in the ACGC and the time of this determination? (Record volume in units of **gal** and **24-hour clock time** of this determination)

Existing volume, **EV** = _____ gal

Date: __/__/__, 24-hour clock time: ___:___.

Step 6: Determine the estimated critical volume (CV) of the ACGC.

By examination and interpolation of scales and values presented in the figure selected in step 3:

(a) Along the abscissa, locate the **cylinder dimension** (diameter or length in inches) identified in step 2.

(b) Vertically scan along the identified cylinder dimension to that region of the figure that approximates the fissile material **uranium density** (in grams of uranium per liter) identified in step 4.

(c) At the approximate intersection of the cylinder dimension with the uranium density value, estimate the **critical volume** (CV) in gal. (Record volume)

Critical volume, **CV** = _____ gal

Step 7: Record the FM addition rate (AR) into the ACGC.

Record FM AR into the ACGC, **AR** = _____ gal/min.

Step 8: Determine the time (T) to accumulate enough FM to become critical.

Solve the following mathematical relationship:

T = (CV - EV) / AR = (_____ - _____) / _____ = _____ minutes.

Step 9: Determine preliminary first-pulse fission (PFPF) yield at plotted value.

By examination and interpolation of scales and values presented in the figure selected in step 3:

(a) Along the abscissa, locate the **cylinder dimension** (diameter or length in inches) identified in step 2.

(b) Vertically scan along the identified cylinder dimension to that region of the figure that approximates the fissile material **uranium density** [in grams of uranium per liter] identified in step 4.

(c) At the approximate intersection of the cylinder dimension with the uranium density value, horizontally scan leftward to the figure ordinate to read the **PFPF** yield. (Record value)

Preliminary first-pulse fission yield, **PFPF** = _____ fissions.

Step 10: Determine estimated first-pulse fission (FPF) yield according to FM AR fission yield multiplier.

The figure is based upon solution or damp oxide addition rates of 45 gal/min. In the slide-rule format the user may make adjustments of the curves in each of the figures to accommodate material addition rates by using the "addition rate (gal/min)" pointer on the slide-rule scale to the right of Fig. 24. In lieu of a functional slide rule, fission-yield multipliers are provided below for various material addition rates between 1 and 100 gal/min.

Fissile Material Addition Rate (gal/min) to Fission Yield Multiplier (FYM) values					
FM AR (gal/min)	**FYM**	**FM AR (gal/min)**	**FYM**	**FM AR (gal/min)**	**FYM**
0.1	0.04	2	0.21	30	0.81
0.2	0.06	3	0.25	40	0.94
0.3	0.08	4	0.29	45	1.00
0.4	0.09	5	0.33	50	1.06
0.5	0.10	6	0.36	60	1.16
0.6	0.11	7	0.39	70	1.25
0.7	0.12	8	0.42	80	1.33
0.8	0.13	9	0.44	90	1.42
0.9	0.14	10	0.47	100	1.50
1	0.15	20	0.66	200	2.13

Select an appropriate fission yield multiplier from the above table based upon the fissile material addition rate (**AR**) recorded in step 7. (Record selected FYM value that may be interpolated)

The selected fission yield multiplier value, **FYM** = _____.

Determine the first-pulse fission yield as,

FPF = (**PFPF**) × (**FYM**) = (_____) × (_____) = _____ fissions.

Step 11: Apply the estimated FPF yield as the "ESTIMATED FISSION" input value in the appropriate dose curves of the criticality slide rule.

The appropriate dose curves for use with the FPF estimate are one of Figs. 11, 14 or 15. Each of those figures have their graphic data normalized to 10^{17} fissions. Therefore, without a functional slide rule that allows "ESTIMATED FISSION"scaling it is necessary to mathematically scale data read from the graphs. This scaling factor (SF) is determined by the following relationship.

Scaling factor, SF = FPF / 10^{17} fissions = (_____) / 10^{17} = _____ .

All information interpreted from the selected nonfunctional slide-rule figure must be multiplied by the SF to adjust to the first-pulse fission yield estimate.

Step 12: Define appropriate mitigating measures to cope with anticipated criticality accident.

Inspect the lower left graphic presentation in the selected Figs. 11, 14, or 15. Using the scaling factor, SF, from step 11, interpret and adjust data presented in the lower left graphic to determine at what distances personnel may receive excessively large accidental emergency radiation exposures. Define and implement appropriate mitigating measures (e.g., movement of personnel away from the location of the criticality, location of personnel behind radiation shielding materials, etc.).

CONVERSION FACTORS AND EQUALITIES

It requires approximately 10^{17} fissions to boil one liter of water that was originally at room temperature.

3.12×10^{10} fissions per second = 1 watt

1.123×10^{14} fissions = 1 watt-hr

1 gallon = 3.785 liters

1 liter = 0.264 gallons

1 liter / minute = 0.264 gallons / minute

1 cubic foot = 7.481 gallons

1 cm = 0.394 inches

The total fission yield[4] of a continuing solution criticality may be estimated by:

$$\text{Total Fissions} = \frac{[(\text{Solution volume, gallons}) \times (\text{Duration of criticality, minutes})]}{1.563 \times 10^{-17} + [1.686 \times 10^{-17} \times (\text{Duration of criticality, minutes})]}$$

$$= \frac{[(_____\text{, gallons}) \times (_____\text{, minutes})] \text{ fissions}}{1.563 \quad 10^{-17} + [1.686 \times 10^{-17} \times (_____\text{, minutes})]}$$

$$= _____ \text{ Fissions}$$

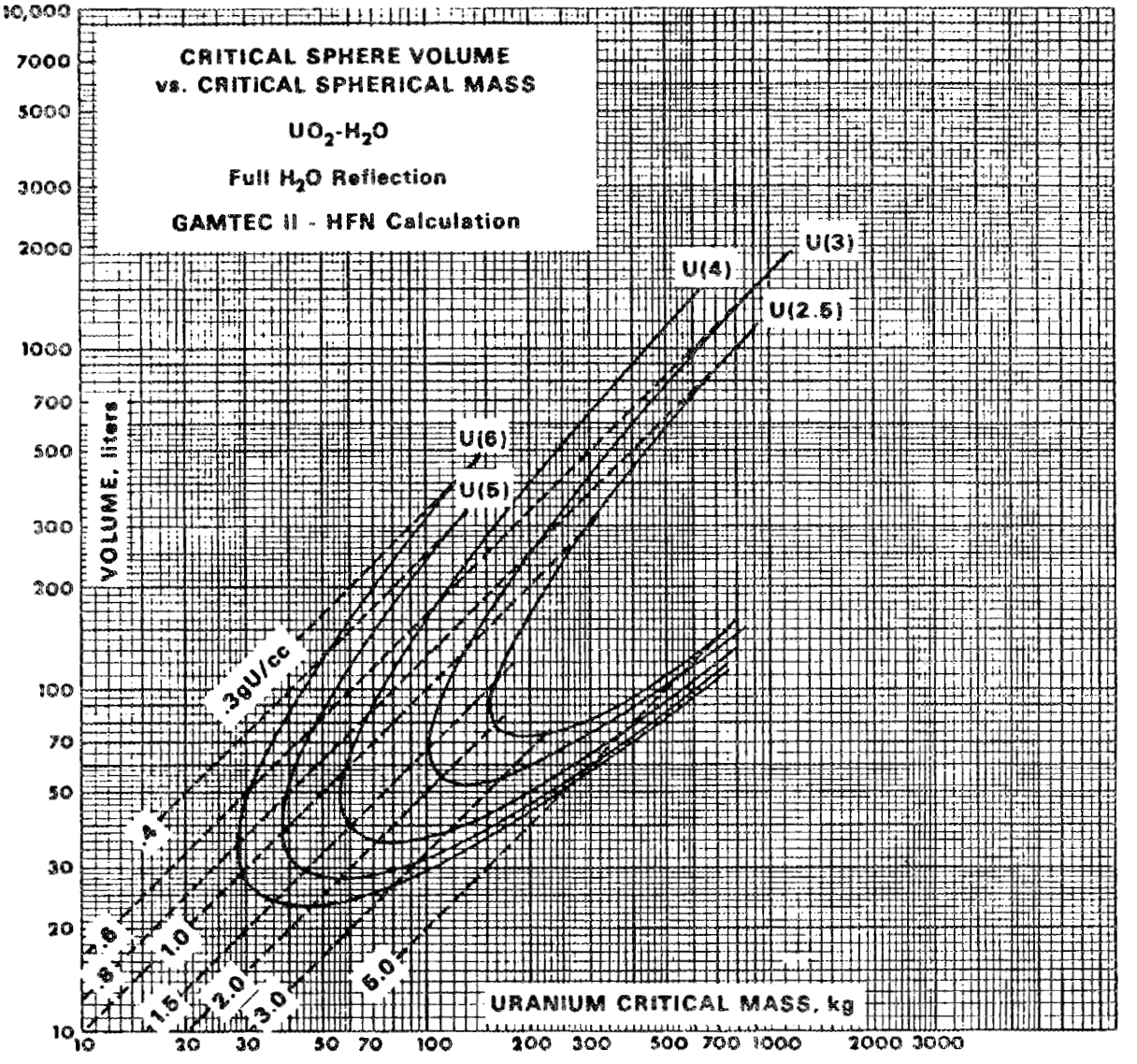
Shielding dose reduction factors may be determined from the following relationships:

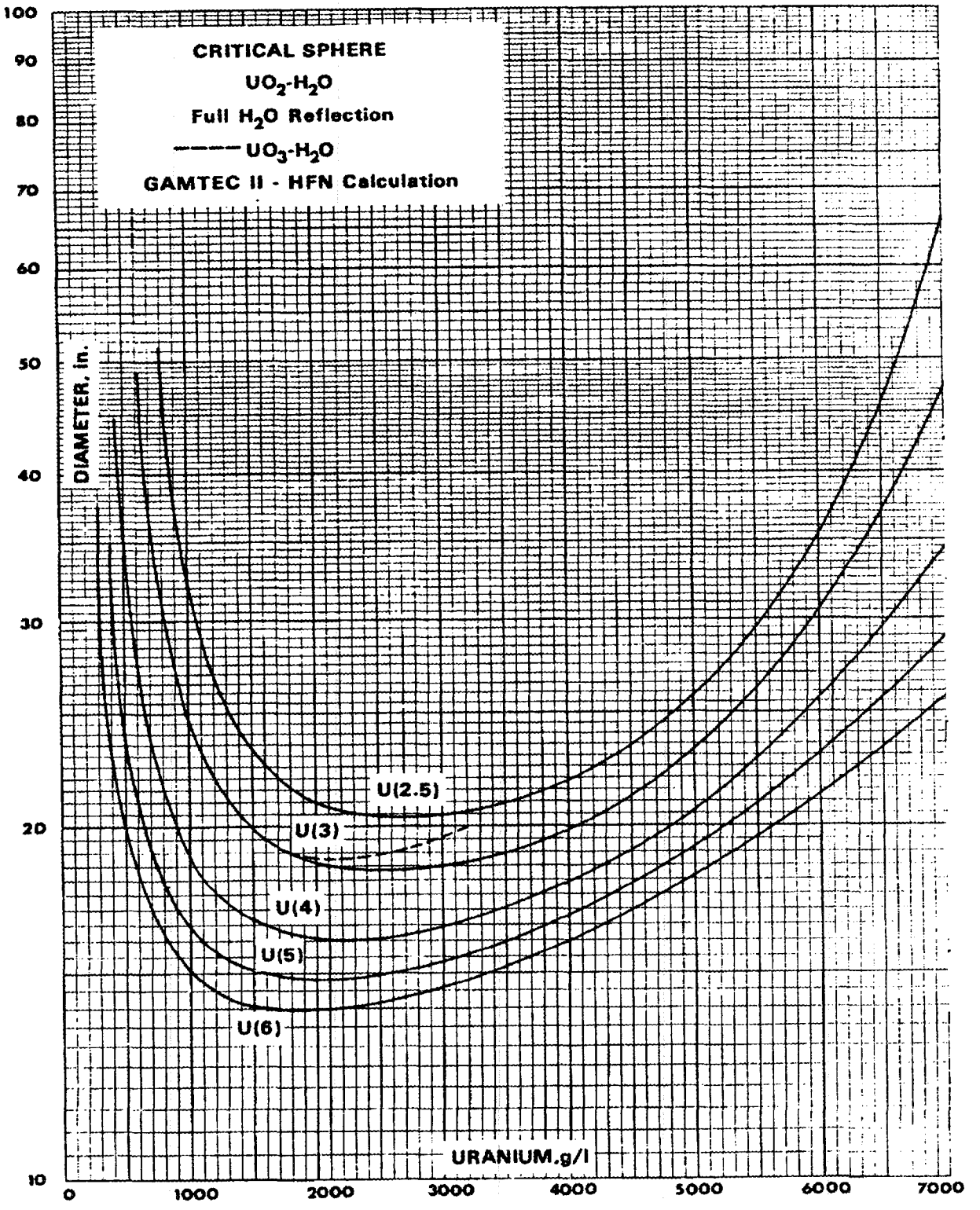
Steel Dose Reduction Factor: neutrons, $n = e^{-0.256 \times \text{steel thickness in inches}}$
gammas, $\gamma = e^{-0.386 \times \text{steel thickness in inches}}$

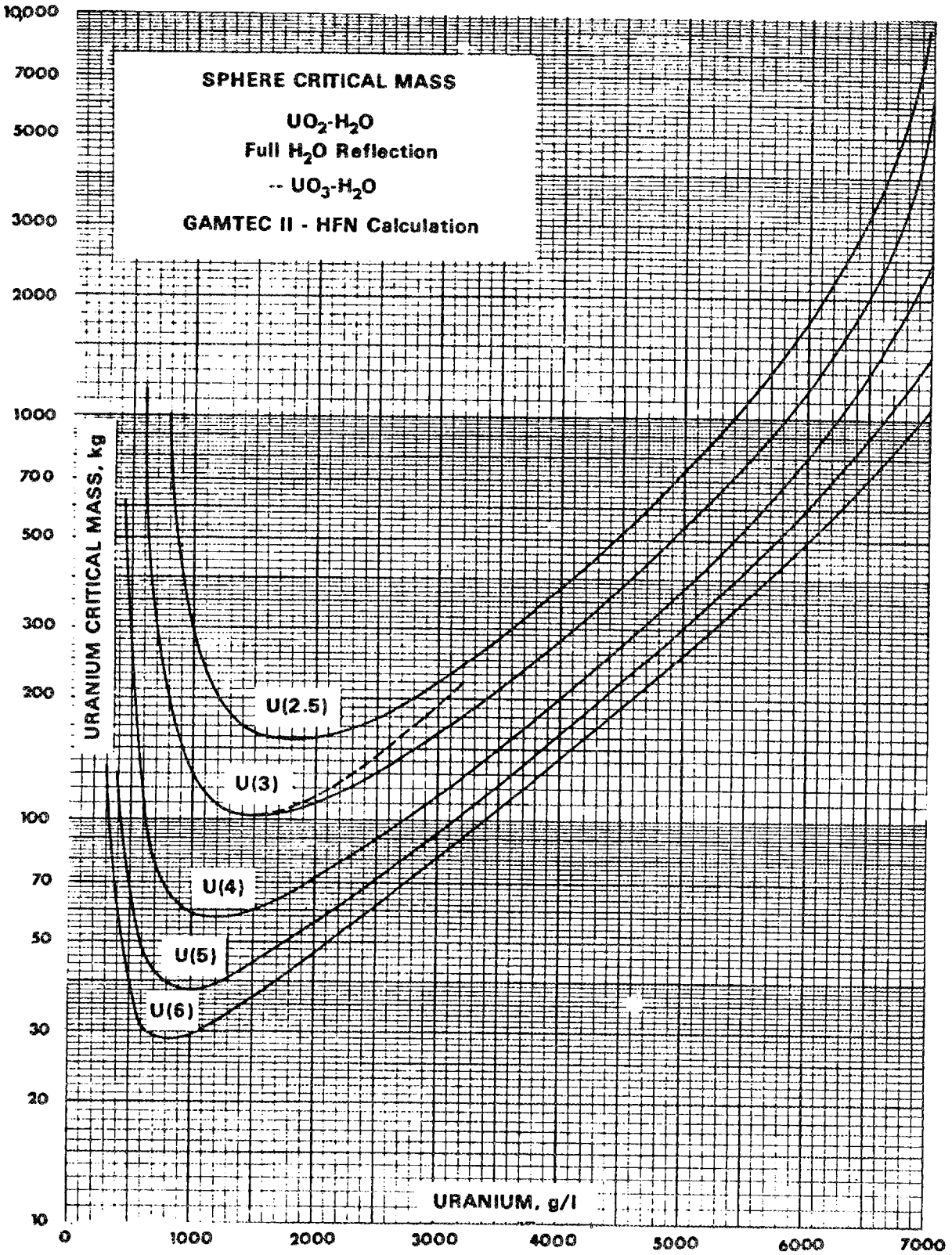
Concrete Dose Reduction Factor: neutrons, $n = e^{-0.240 \times \text{concrete thickness in inches}}$
gammas, $\gamma = e^{-0.147 \times \text{concrete thickness in inches}}$

Water Dose Reduction Factor: neutrons, $n = e^{-.277 \times \text{water thickness in inches}}$
gammas, $\gamma = e^{-0.092 \times \text{water thickness in inches}}$

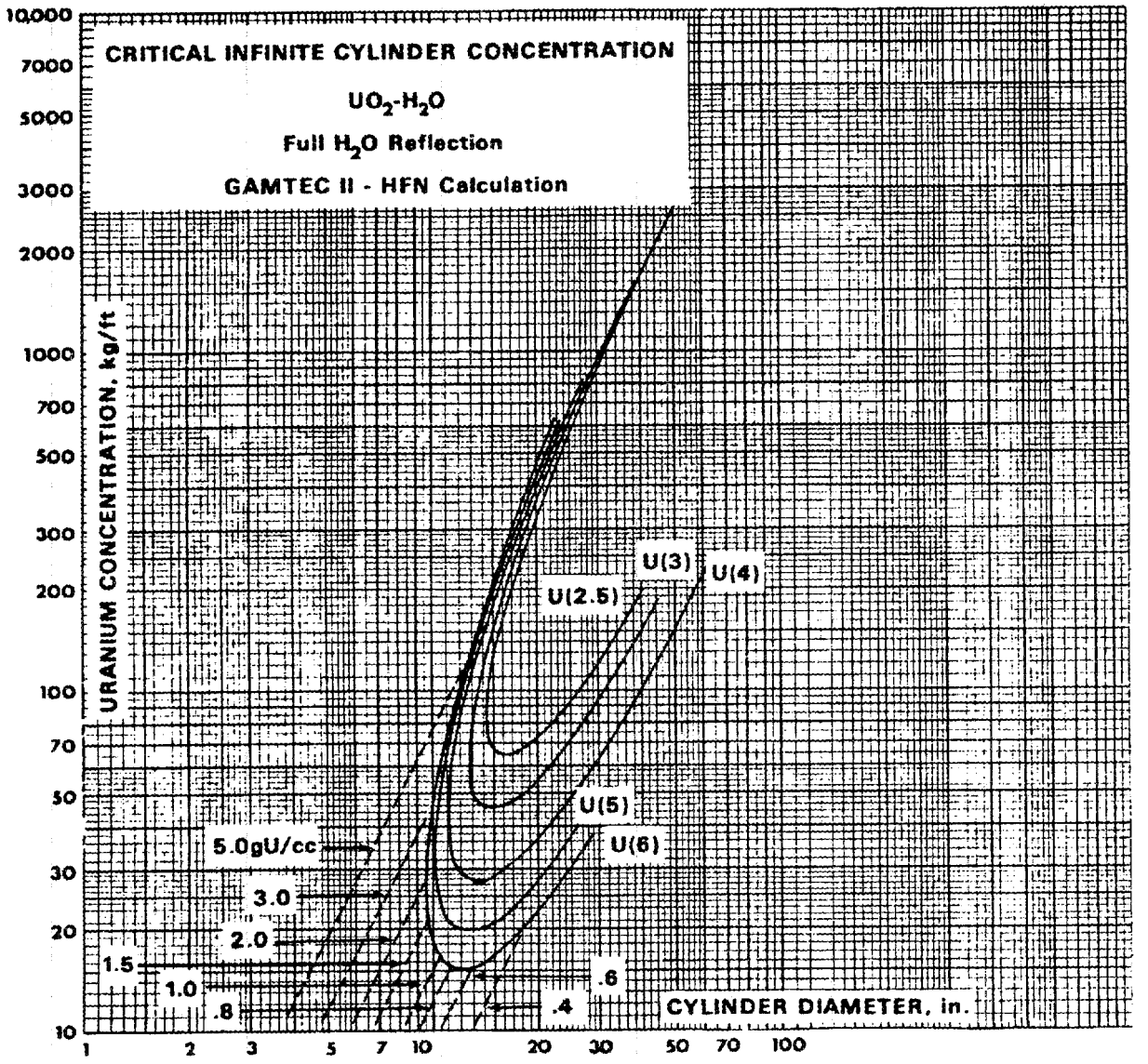
Selections from R. D. Carter, G. R. Kiel, and K. R. Ridgway, *Criticality Handbook*, ARH-600, Vol. II, Atlantic Richfield Hanford Co., May 23, 1969.



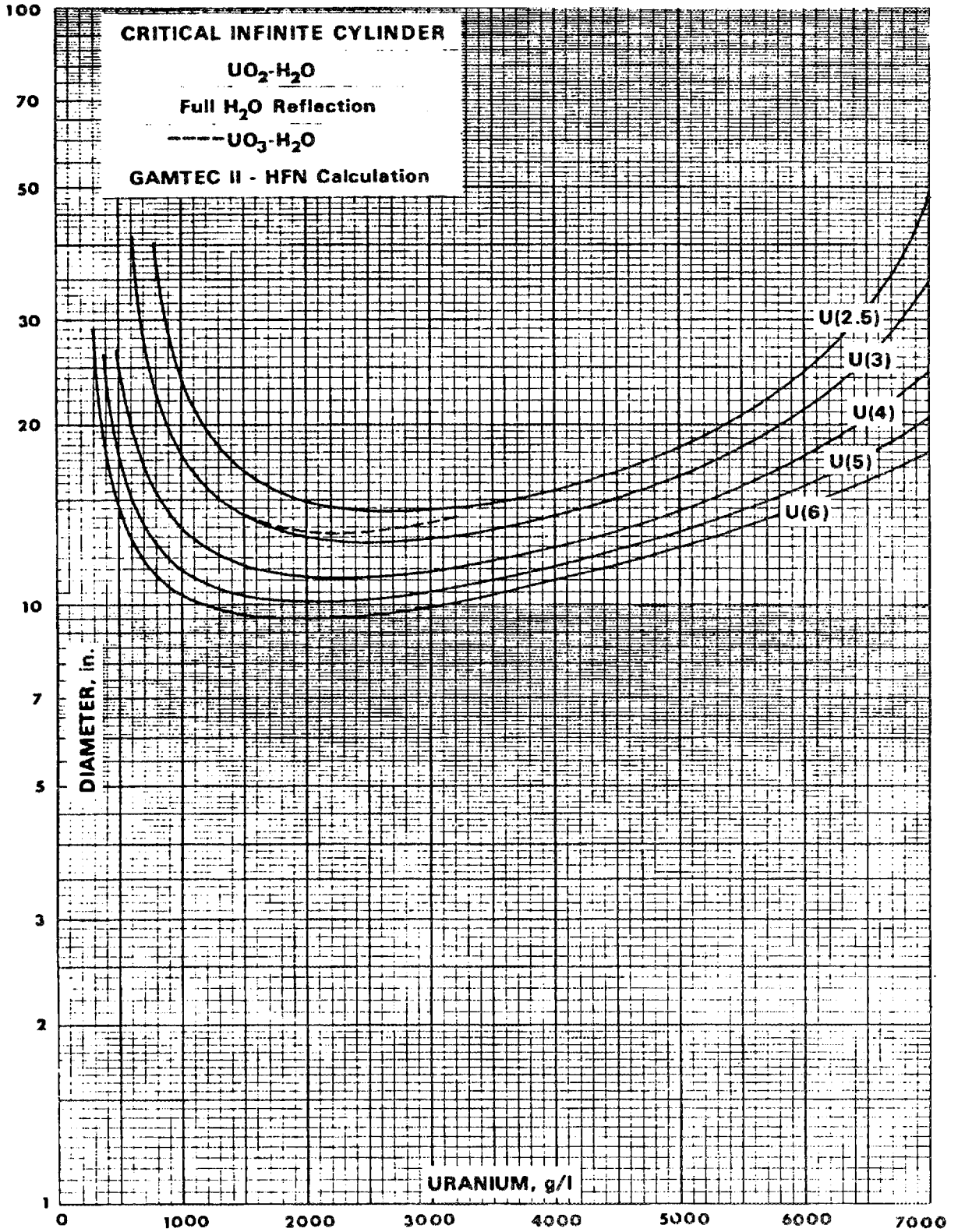

CRITICAL SPHERE
UO_2-H_2O
Full H_2O Reflection
UO_3-H_2O
GAMTEC II - HFN Calculation
DIAMETER, in.
URANIUM,g/l
U(2.5)
U(3)
U(4)
U(5)
U(6)
100
90
80
70
60
50
40
30
20
10
0
1000
2000
3000
4000
5000
6000
7000

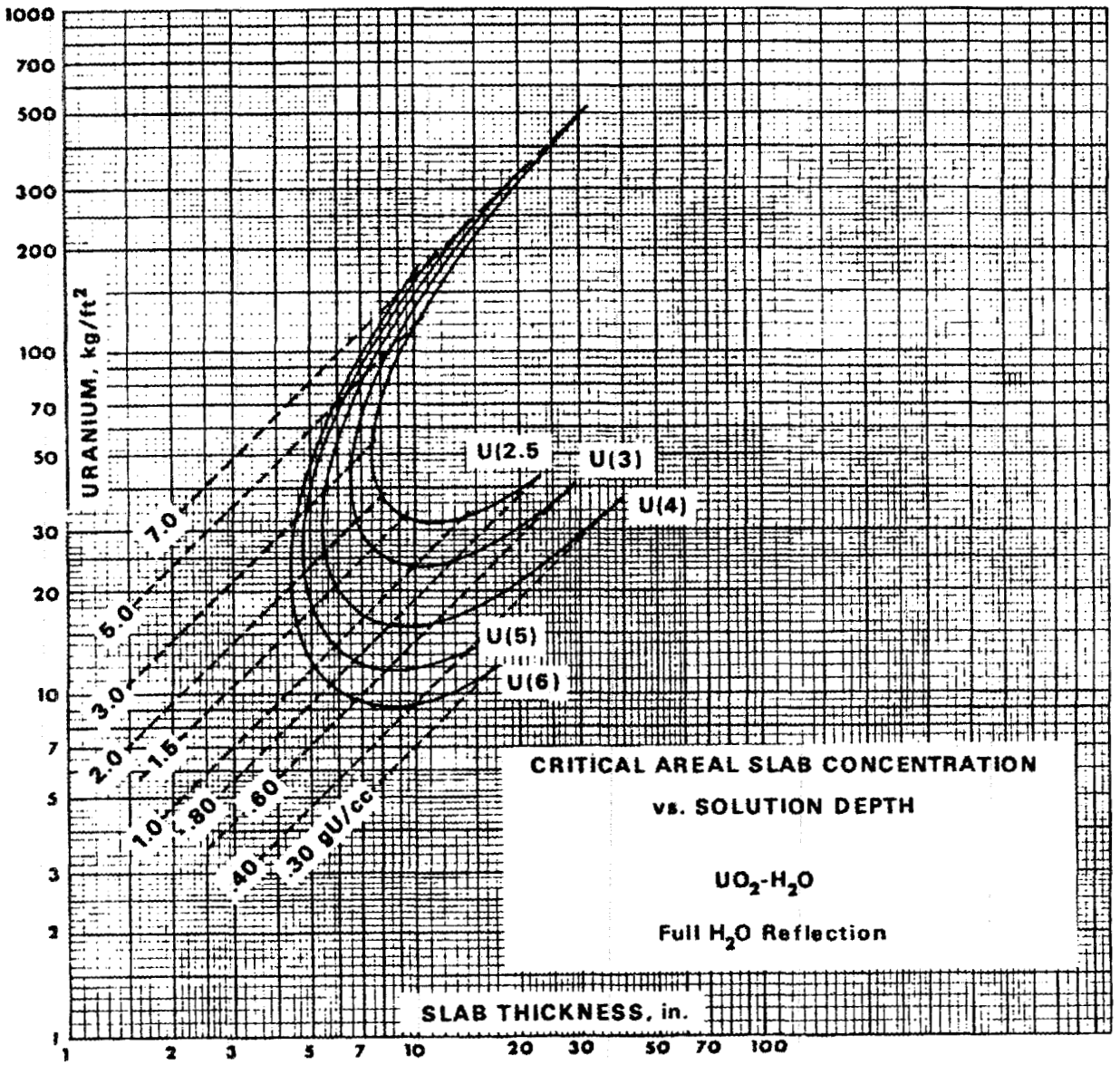

SPHERE CRITICAL MASS
UO_2-H_2O
Full H_2O Reflection
-- UO_3-H_2O
GAMTEC II - HFN Calculation
URANIUM CRITICAL MASS, kg
10,000
7000
5000
3000
2000
1000
700
500
300
200
100
70
50
30
20
10
U(2.5)
U(3)
U(4)
U(5)
U(6)
URANIUM, g/l
0
1000
2000
3000
4000
5000
6000
7000

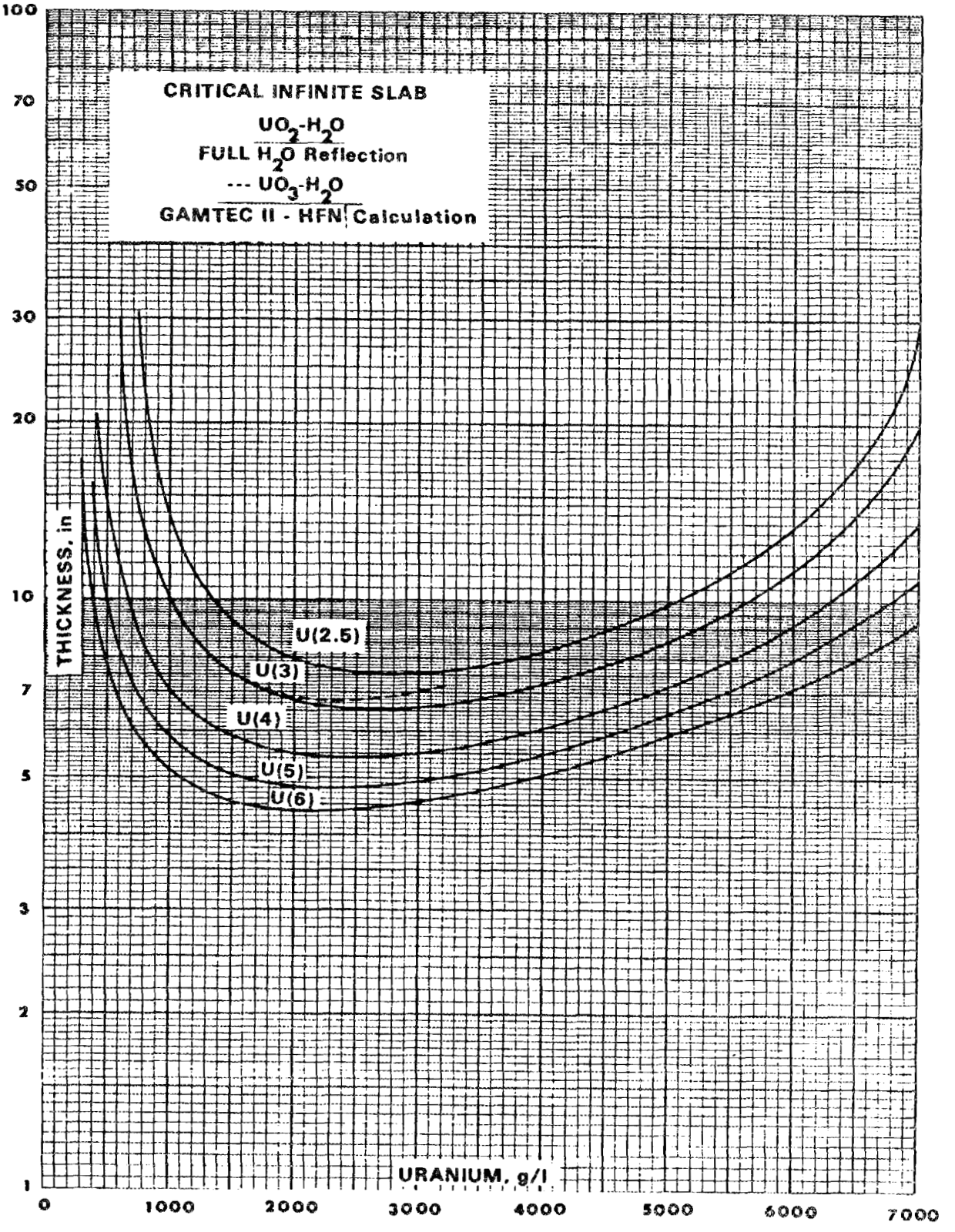
III.B.7-6 ARH-600



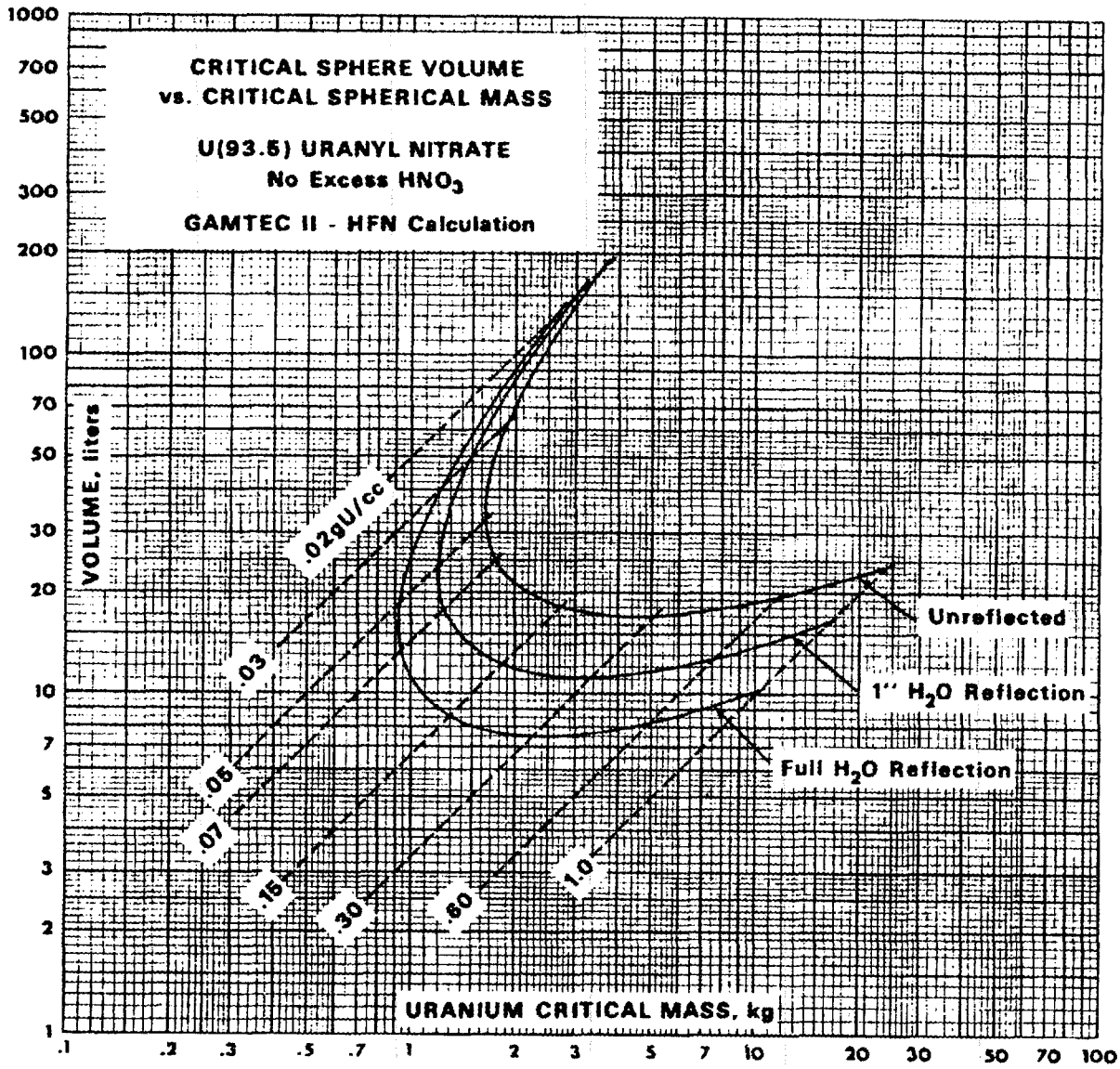
Revised:
3-19-76

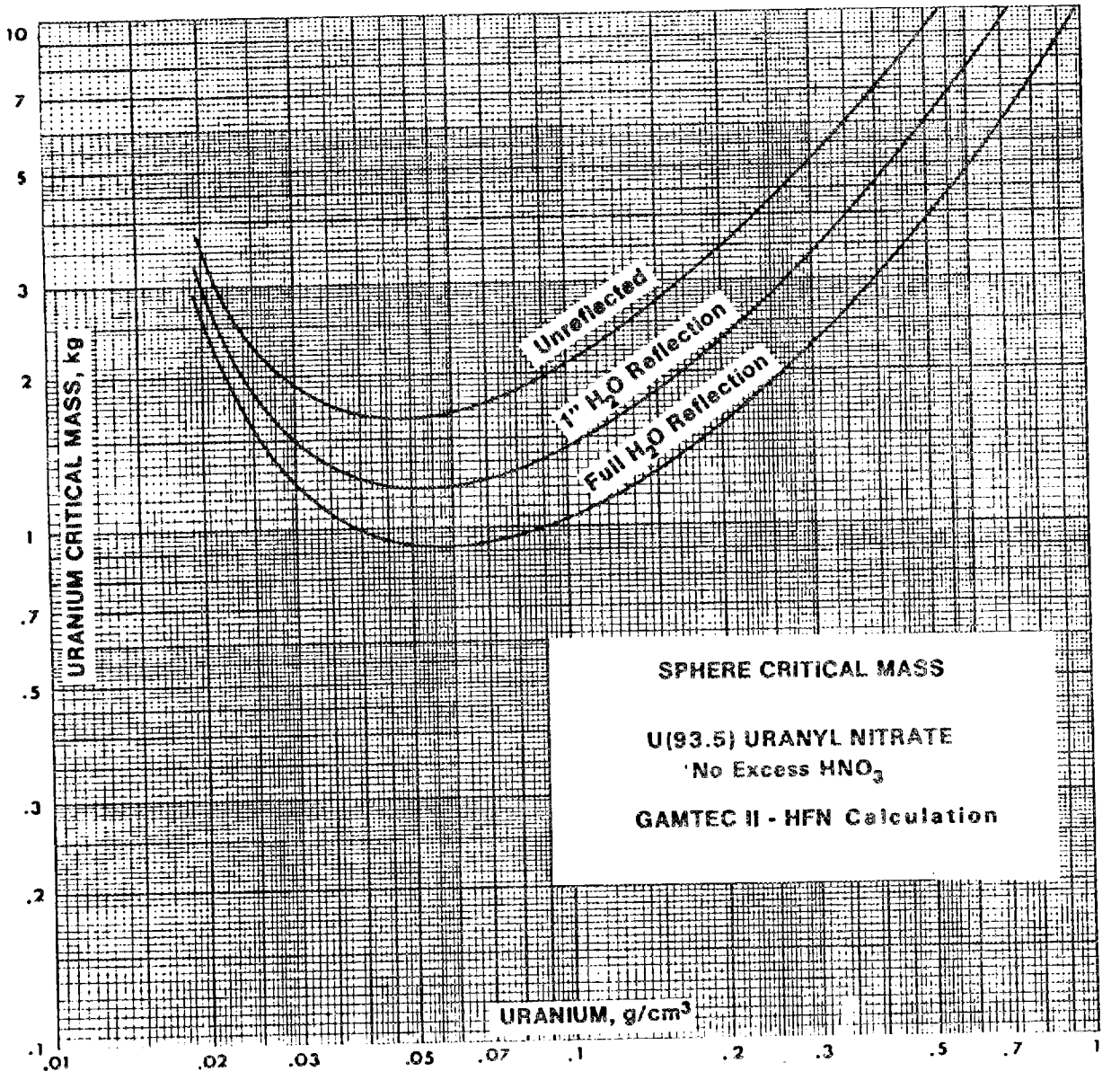

CRITICAL INFINITE CYLINDER
UO_2-H_2O
Full H_2O Reflection
-----UO_3-H_2O
GAMTEC II - HFN Calculation
U(2.5)
U(3)
U(4)
U(5)
U(6)
DIAMETER, in.
100
70
50
30
20
10
7
5
3
2
1
URANIUM, g/l
0
1000
2000
3000
4000
5000
6000
7000

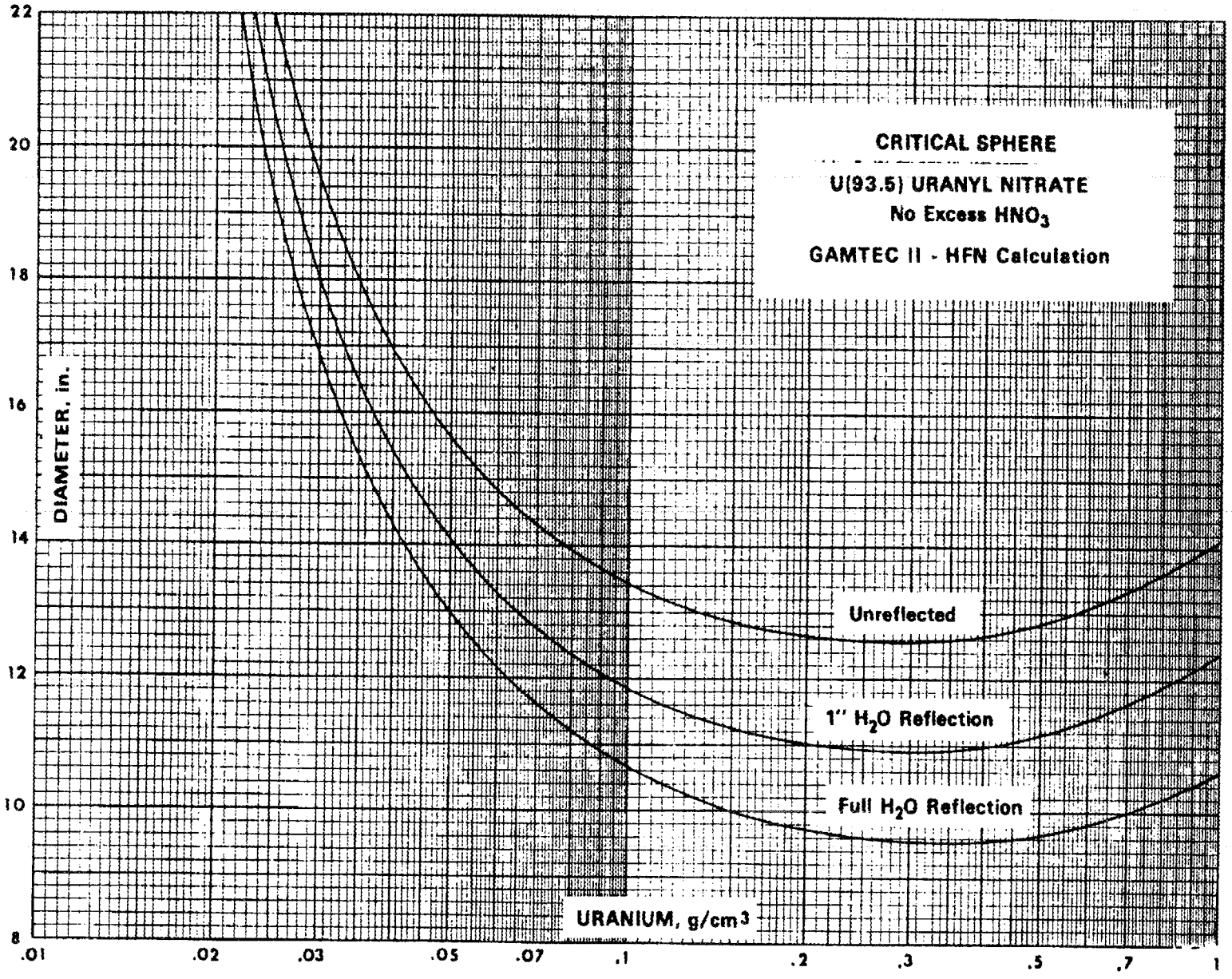

URANIUM, kg/ft²
1000
700
500
300
200
100
70
50
30
20
10
7
5
3
2
1
SLAB THICKNESS, in.
1
2
3
5
7
10
20
30
50
70
100
U(2.5
U(3)
U(4)
U(5)
U(6)
7.0
6.0
3.0
2.0
1.5
1.0
.80
.60
.40
.30 gU/cc
CRITICAL AREAL SLAB CONCENTRATION
vs. SOLUTION DEPTH
UO2-H2O
Full H2O Reflection

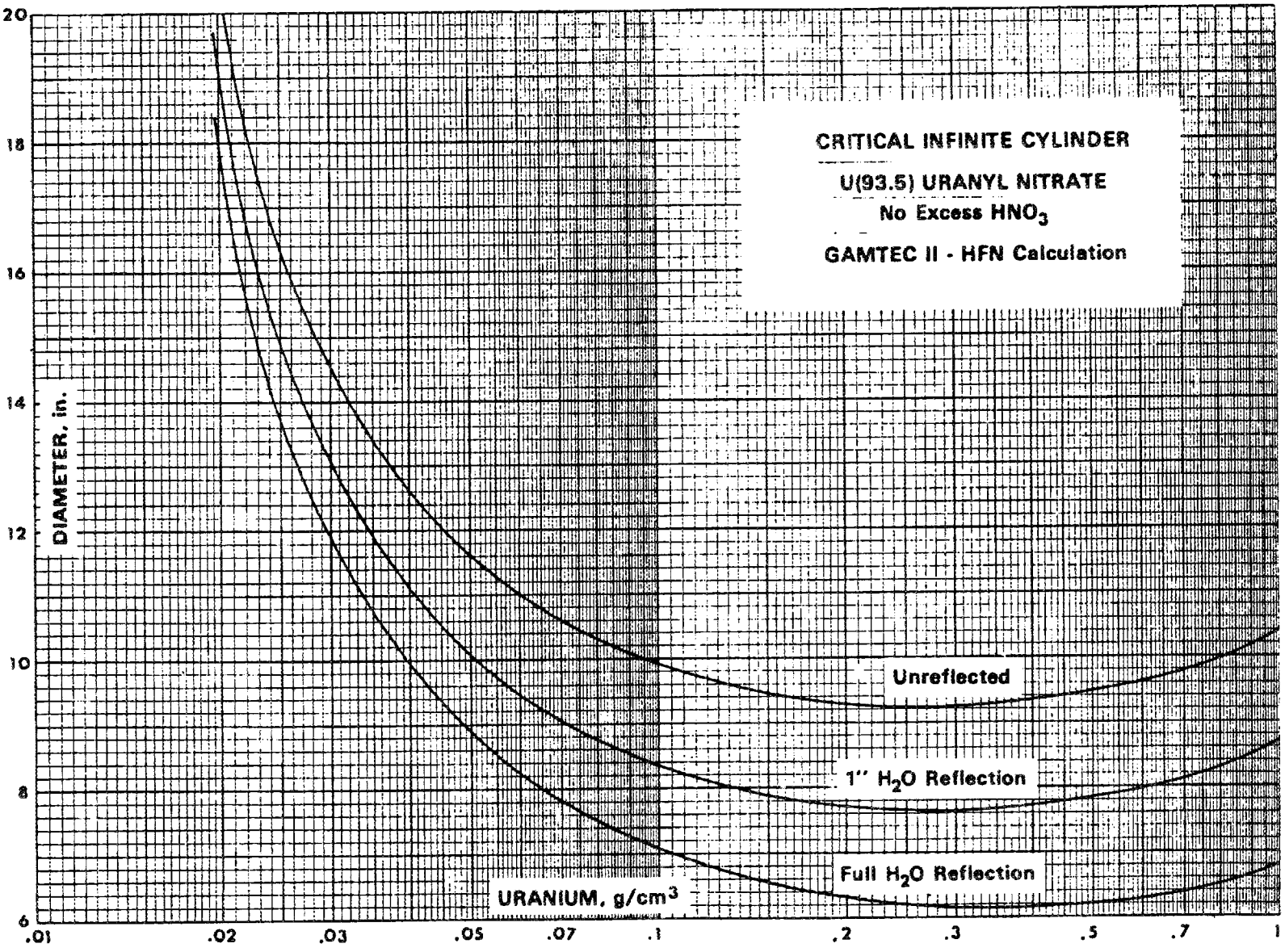

CRITICAL INFINITE SLAB
UO2-H2O
FULL H2O Reflection
--- UO3-H2O
GAMTEC II - HFN Calculation
THICKNESS, in
100
70
50
30
20
10
7
5
3
2
1
U(2.5)
U(3)
U(4)
U(5)
U(6)
URANIUM, g/l
0
1000
2000
3000
4000
5000
6000
7000

III.B.9(93.5)-1 ARH-600

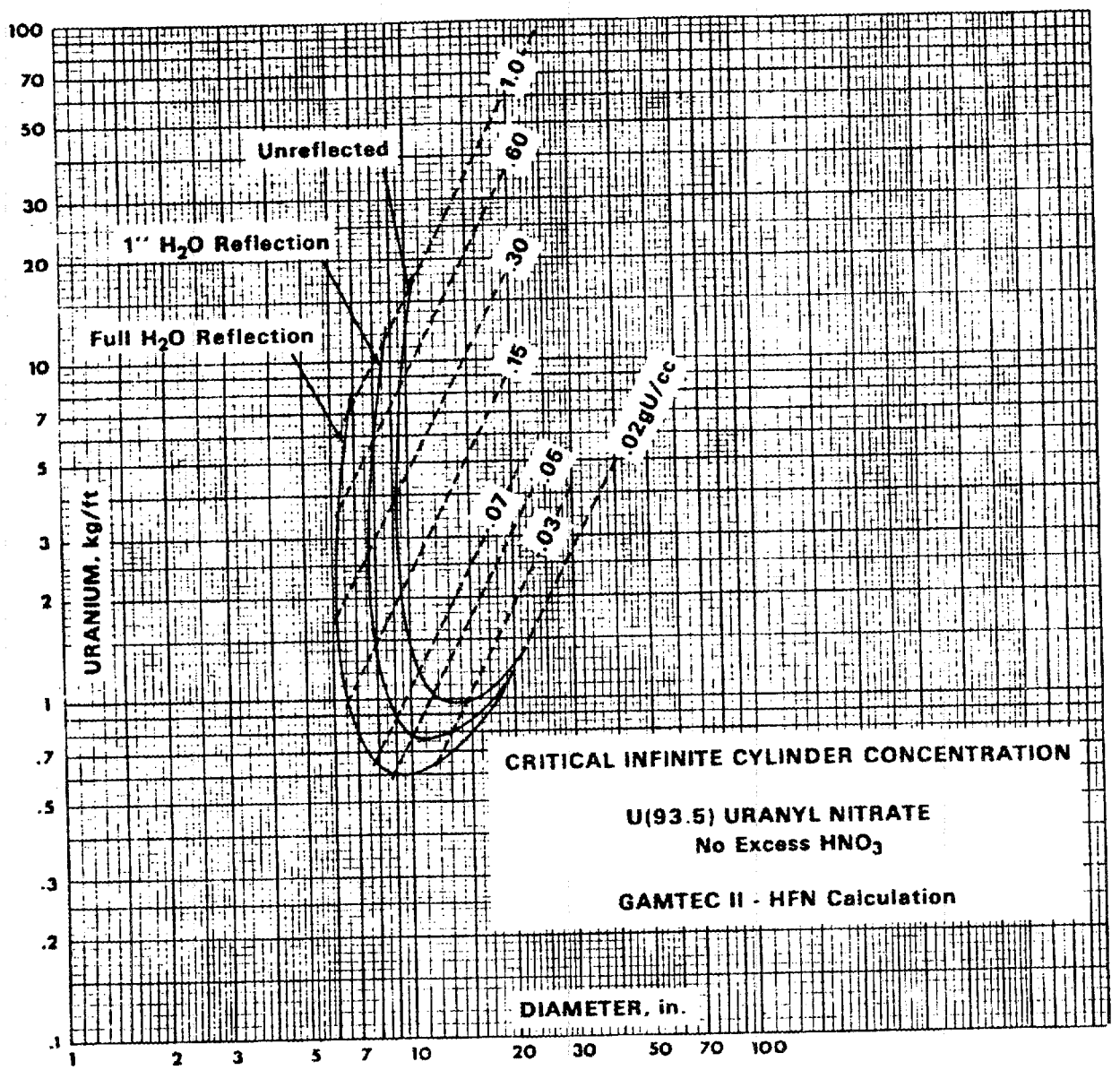


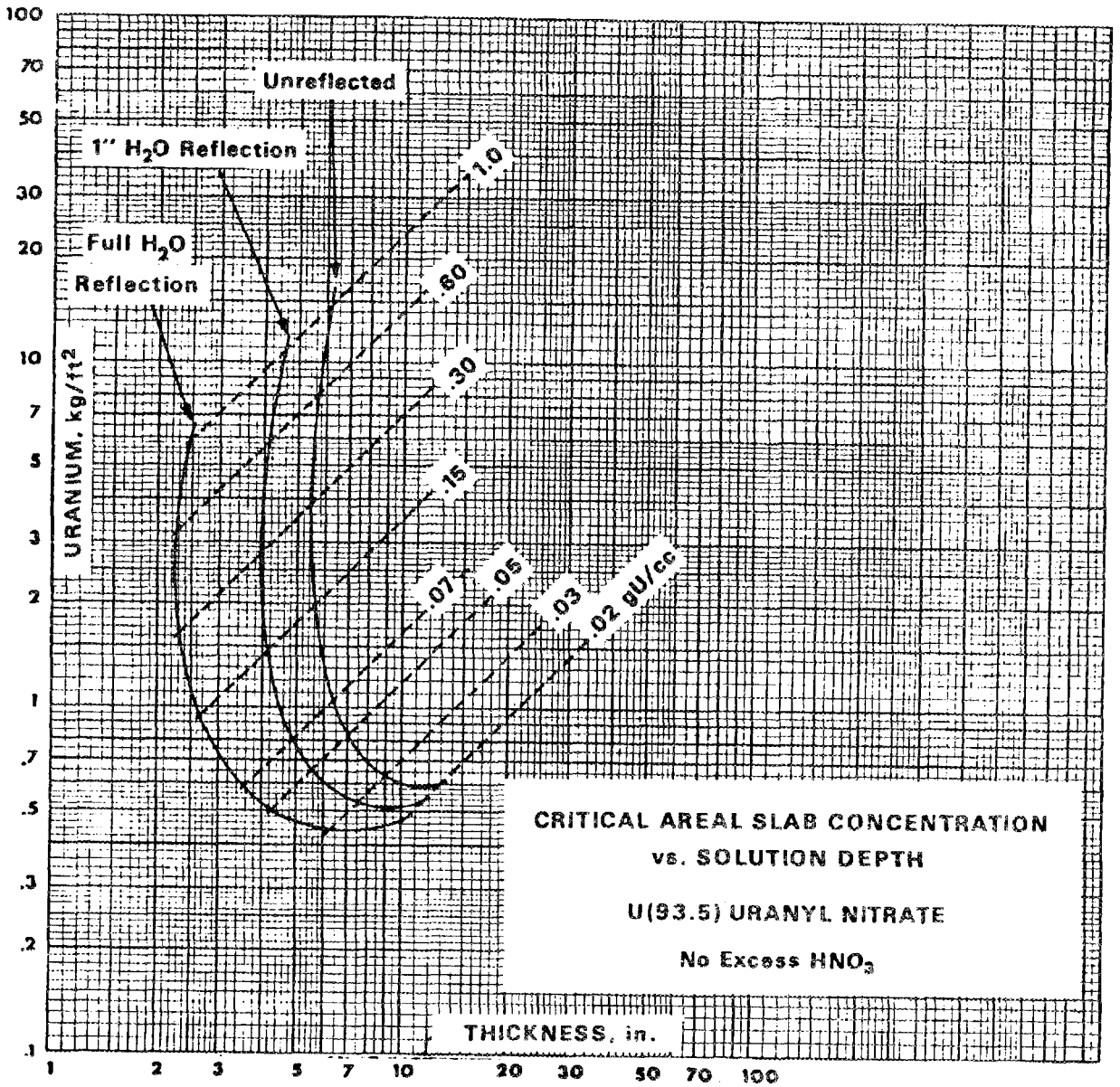

URANIUM CRITICAL MASS, kg
10
7
5
3
2
1
.7
.5
.3
.2
.1
Unreflected
1" H_2O Reflection
Full H_2O Reflection
SPHERE CRITICAL MASS
U(93.5) URANYL NITRATE
No Excess HNO_3
GAMTEC II - HFN Calculation
URANIUM, g/cm³
.01
.02
.03
.05
.07
.1
.2
.3
.5
.7
1

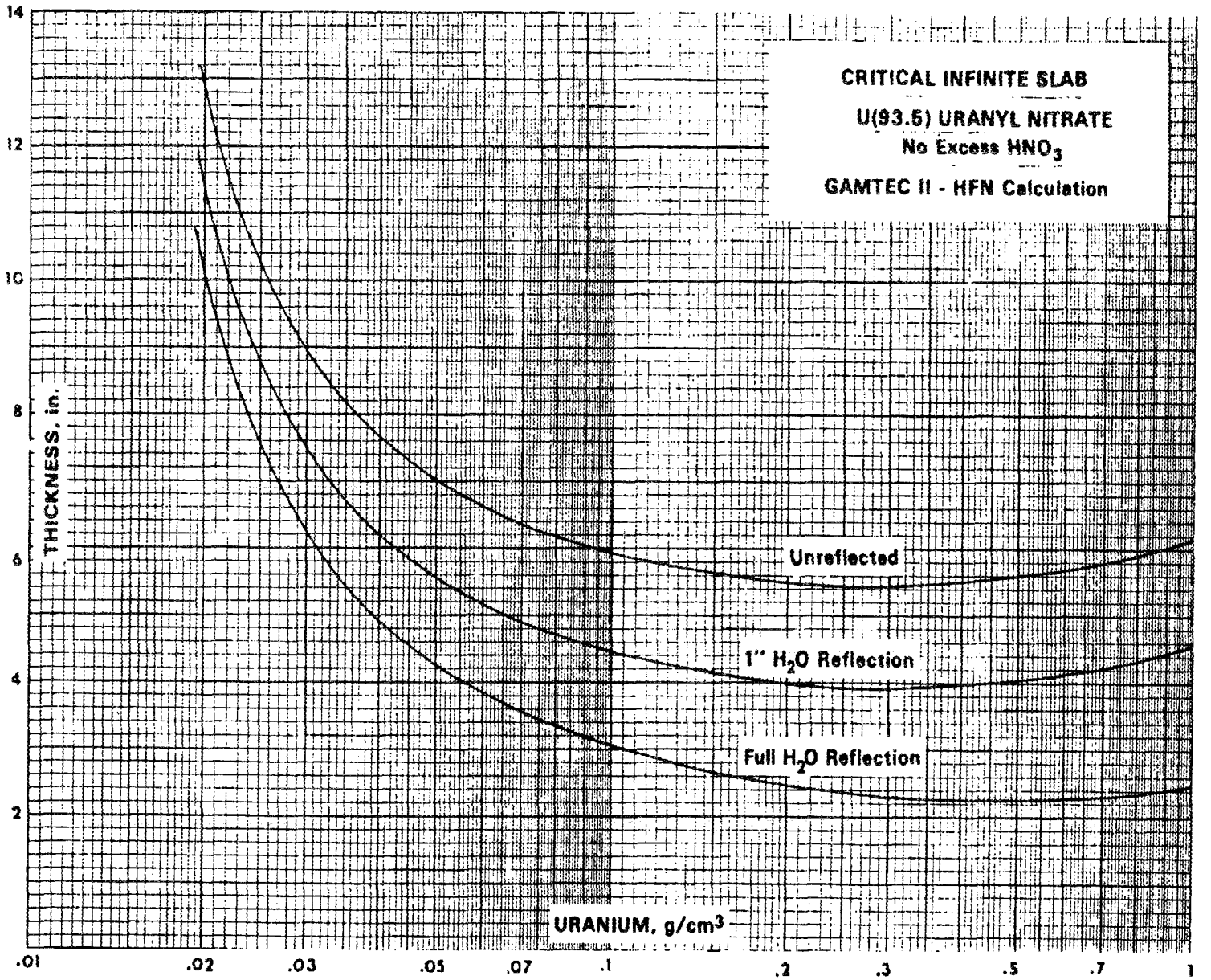

CRITICAL SPHERE
U(93.5) URANYL NITRATE
No Excess HNO_3
GAMTEC II - HFN Calculation
DIAMETER, in.
22
20
18
16
14
12
10
8
Unreflected
1" H_2O Reflection
Full H_2O Reflection
URANIUM, g/cm3
.01
.02
.03
.05
.07
.1
.2
.3
.5
.7
1


CRITICAL INFINITE CYLINDER
U(93.5) URANYL NITRATE
No Excess HNO_3
GAMTEC II - HFN Calculation
DIAMETER, in.
20
18
16
14
12
10
8
6
Unreflected
1" H_2O Reflection
Full H_2O Reflection
URANIUM, g/cm^3
.01
.02
.03
.05
.07
.1
.2
.3
.5
.7
1

III.B.7(93.5)-1 ARH-600




Unreflected
1" H_2O Reflection
Full H_2O Reflection
1.0
.60
.30
.15
.07
.05
.03
.02 gU/cc
URANIUM, kg/ft^2
THICKNESS, in.
CRITICAL AREAL SLAB CONCENTRATION
vs. SOLUTION DEPTH
U(93.5) URANYL NITRATE
No Excess HNO_3


CRITICAL INFINITE SLAB
U(93.5) URANYL NITRATE
No Excess HNO_3
GAMTEC II - HFN Calculation
THICKNESS, in.
14
12
10
8
6
4
2
Unreflected
1" H_2O Reflection
Full H_2O Reflection
URANIUM, g/cm^3
.01
.02
.03
.05
.07
.1
.2
.3
.5
.7
1