Log Number: 4.2CMH04
Topic Area Number: 4.2
Oral Presentation

Computational Physics and Engineering Division

Slide Rule for Rapid Response Estimation of Radiological Dose from Criticality Accidents

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Submitted to the
ICNC'99
Sixth International Conference on Nuclear Criticality Safety,
September 20–24, 1999,
Palais des congrès . Versailles . France

*Managed by Lockheed Martin Energy Research Corporation under contract DE-AC05-96OR22464 for the U.S. Department of Energy.

SLIDE RULE FOR RAPID RESPONSE ESTIMATION OF RADIOLOGICAL DOSE FROM CRITICALITY ACCIDENTS

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Abstract

This paper describes a functional slide rule that provides a readily usable "in-hand" method for estimating nuclear criticality accident information from sliding graphs, thereby permitting (1) the rapid estimation of pertinent criticality accident information without laborious or sophisticated calculations in a nuclear criticality emergency situation, (2) the appraisal of potential fission yields and external personnel radiation exposures for facility safety analyses, and (3) a technical basis for emergency preparedness and training programs at nonreactor nuclear facilities. The slide rule permits the estimation of neutron and gamma dose rates and integrated doses based upon estimated fission yields, distance from the fission source, and time-after criticality accidents for five different critical systems. Another sliding graph permits the estimation of critical solution fission yields based upon fissile material concentration, critical vessel geometry, and solution addition rate. Another graph provides neutron and gamma dose-reduction factors for water, steel, and concrete shields.

Introduction

To perform safety analyses and to develop and maintain a program of emergency preparedness and response for nonreactor nuclear facilities that process fissile materials, it is necessary to hypothesize credible magnitudes of nuclear criticality accidents, potential personnel hazards, and safe corrective actions in the event of a nuclear criticality accident. In an effort to provide general technical information that relates to these requirements, a rapid, "in-hand"method has been developed for estimating pertinent information needed to guide response team actions and to help characterize some types of criticality accidents. The concept uses a series of sliding graphs that function similar to that of a slide rule. This hand-held functional tool was developed with the premise that visual demonstration of trends (e.g., dose versus time or distance) is helpful to response personnel and that the use of a nonelectronic estimator is prudent in the moments immediately following a criticality event. The characterization of and potential dose from a criticality event depends on numerous parameters: type and form of the fissile material in the system, plausible system conditions, time and distance from the critical event, and available shielding between the dose point and the criticality source. Using these parameters and a suitable range of parameter values, the slide rule is designed to provide estimates of the following:

1. magnitude of the fission yield based on knowledge of the particular system parameters and/or personnel or field radiation measurements,
2. neutron- and gamma-dose at variable unshielded distances from the accident,
3. the skyshine component of the dose for use in situations where shielding around the event causes the skyshine component to be the dominant dose component,
4. time-integrated radiation dose estimates at variable distances from and time after the accident,

5. 1-min gamma radiation dose integrals at variable distances from and time after the accident,
6. dose-reduction factors for variable thicknesses of steel, concrete, and water.

Used in the field or in training simulations, the slide rule enables rapid estimation of unknown data based on known data available to the emergency response personnel. 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 reentry personnel, estimation of future radiation field magnitudes, and fission yield estimates.

Originally conceived and developed for use at the U.S. Department of Energy's Y-12 plant [1], the slide rule concept has recently been expanded and updated to support the U.S. Nuclear Regulatory Commission (NRC) Response Technical Manual [2,3]. Five critical systems of unreflected spheres that provide general characteristics of the solution, powder, and metal operations likely in facilities licensed by the NRC were considered:

1. low-enriched (4.95 wt% ^{235}U) aqueous uranyl fluoride solution with an H/X of 410 (solution density = 2.16 g/cm^3);
2. damp, low-enriched (5 wt% ^{235}U) uranium dioxide with an H/X of 200;
3. high-enriched (93.2 wt% ^{235}U) uranyl nitrate solution with an H/X of 500 (solution density = 1.075 g/cm^3);
4. high-enriched (93.2 wt% ^{235}U) uranium metal with density = 18.85 g/cm^3; and
5. damp, high-enriched (93.2 wt% ^{235}U) uranium oxide (U_3O_8) plus water with an H/X of 10 and a U_3O_8 density = 4.15 g/cm^3.

This paper provides a summary of the technical basis used to develop the slide rule data and discusses the scope of its use.

General Approach

In the original slide rule of Ref. 1, only the high-enriched metal and uranium nitrate systems were considered and the mathematical models were limited (e.g., use of inverse square rule, neglecting air attenuation, and use of approximate analytic expressions for the time-dependent sources) but carefully compared with available measurement data. The current work described here expanded the systems of interest to the five listed above and sought to improve the technical information in the slide rules by utilizing more rigorous models for the fission yield, decay, and radiation transport. However, only three of the five systems selected for the current slide rule had relevant experimental data that could be used to verify the analytic results.

A flowchart that characterizes the basic steps in the generation of the original and current 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. As a preliminary step in creating the current slide rule, initial scoping analyses were performed using one-dimensional (1-D) discrete-ordinates methods for static studies and point-depletion/decay methods combined with 1-D discrete-ordinates methods for time-dependent studies. These preliminary 1-D analyses enabled a ready comparison to the existing slide rule information and allowed efficient development of a production process for creating the slide rule data using two-dimensional (2-D) radiation transport models.

The influence of the air/ground interface, as well as the possible contributions due to radiation skyshine, are the major reasons for utilizing 2-D methods. The portion of the total dose due to skyshine included in the current slide rule is to allow the determination of accident characteristics where substantial shielding is present between the postulated accident and the desired location of radiation hazard information.

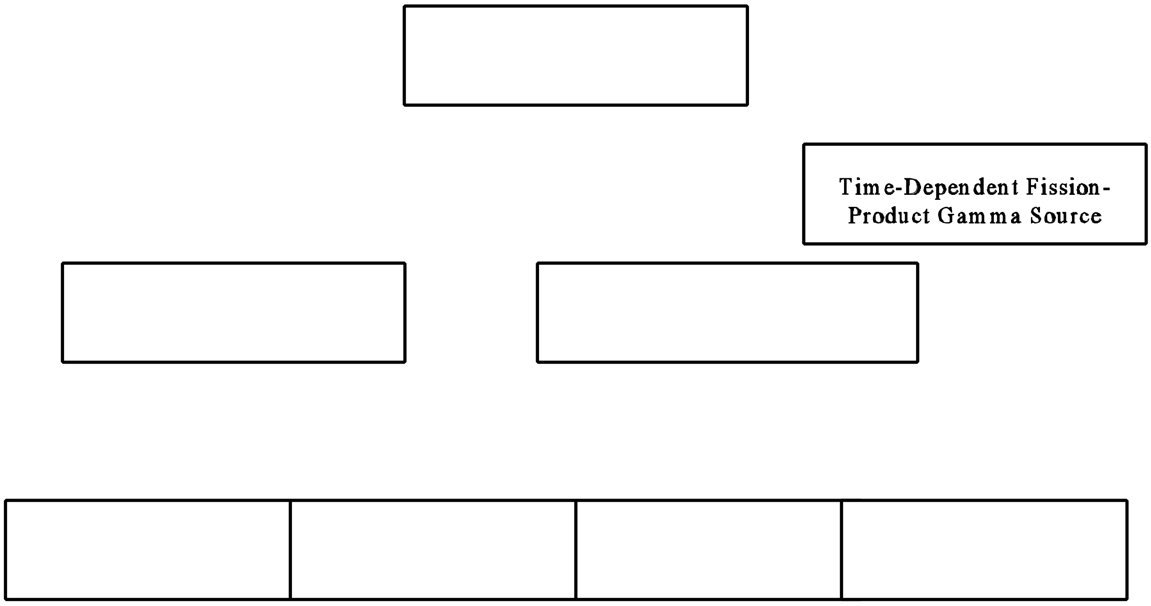
Time-Dependent Fission-
Product Gamma Source

Table 1 Critical system parameters[a]

Parameter	Uranyl fluoride (4.95%)	Damp UO_2 (5%)	Uranyl nitrate solution (93.2%)	U metal[b] (93.2%)	Damp U_3O_8 (93.2%)
Number density[c]					
U-235	1.3173E-4	2.6060E-4	1.3154E-4	4.5012E-2	6.4361E-3
U-238	2.5342E-3	4.9592E-3	9.6010E-6	2.6704E-3	4.6956E-4
N	–	–	2.8205E-4	–	–
O	3.1989E-2	3.6544E-2	3.4012E-2	–	5.0641E-2
F	5.3345E-3	–	–	–	–
H	5.3314E-2	5.2203E-2	6.5769E-2	–	6.4460E-2
Sphere 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]For the metal system, the following material number densities were also used: U-234 (4.8503-4) and U-236 (9.6182-5).

[c]Units of atom/barn-cm.

2-D Radiation Transport Model

The development of the final slide-rule information utilizes the 2-D discrete-ordinates code, DORT [7], which models the radiation transport of coupled neutron/gamma-ray interactions. Fixed sources and sources resulting from particle interaction within the medium are allowed. The principal application of DORT is to the deep-penetration transport of neutrons and photons. Since many physical systems associated with radiation transport 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 neutrons and photons.

In Table 1, the assumed composition of air used in the radiation transport is 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). The neglect of water 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 ground was modeled as a 1-ft (30.48-cm) layer of concrete (SCALE material REG-CONCRETE [6]) 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-time-delayed analyses were the leakage spectra.

Although the 1-D and 2-D model results were compared for an infinite-air model with very good agreement, Fig. 2 provides a visual representation of the ground effects found in the more detailed 2-D models. Reference 2 provides similar plots for all five systems and indicates that the impact of the ground effects change somewhat with the characteristics of the system. 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 system 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.

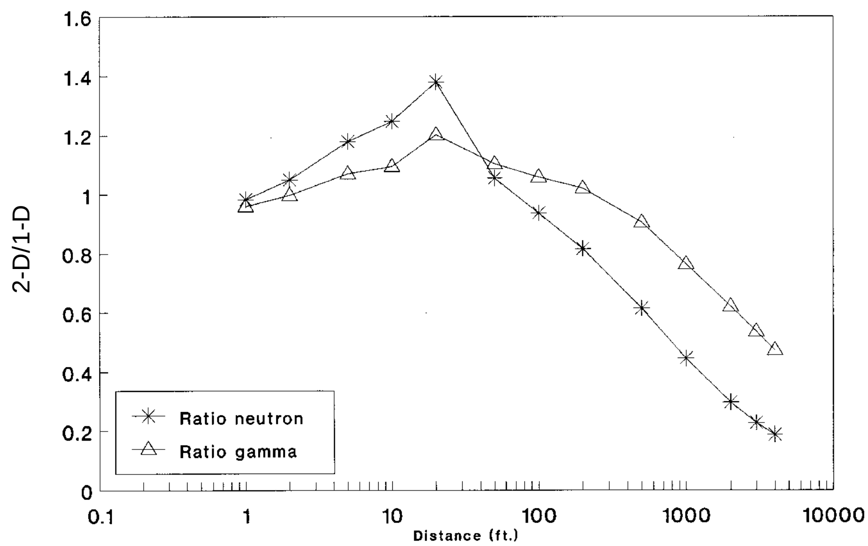

Figure 2. Comparison of 1- and 2-D uranyl fluoride slide rules.

Skyshine and Shield Effects

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 90E 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.

To enable the user to assess the effect of shielding components that would be typical of storage or process building and other nonreactor applications, additional 2-D radiation transport models were developed to evaluate the dose reduction due to a series of thin shields or due to a single shield located between the accident location and the point of interest. Figure 3 provides the effective prompt fission neutron- and gamma-radiation, dose-reduction factors for multiple layers of a specified shielding material located at 24-ft intervals from the criticality accident out to a distance of 240 ft. The thicknesses of the thin shielding materials considered were 1-in. (2.54-cm)-thick layers of steel, or 3-in. (7.62-cm)-thick layers of concrete, or 3-in. (7.62-cm)-thick layers of water. The purpose of evaluating multiple thin layers of shielding materials was to simulate the effects of walls and equipment that may be intervening between operating areas of a facility. Because the dose-reduction factors are based upon coupled neutron-gamma calculations, the influence of neutron-capture gammas is included in the gamma-radiation, dose-reduction factor. The slide rule also provides a figure similar to that of Fig. 3 for thin single shields of material (i.e., steel, concrete, or water) located approximately 10 ft from the criticality accident.

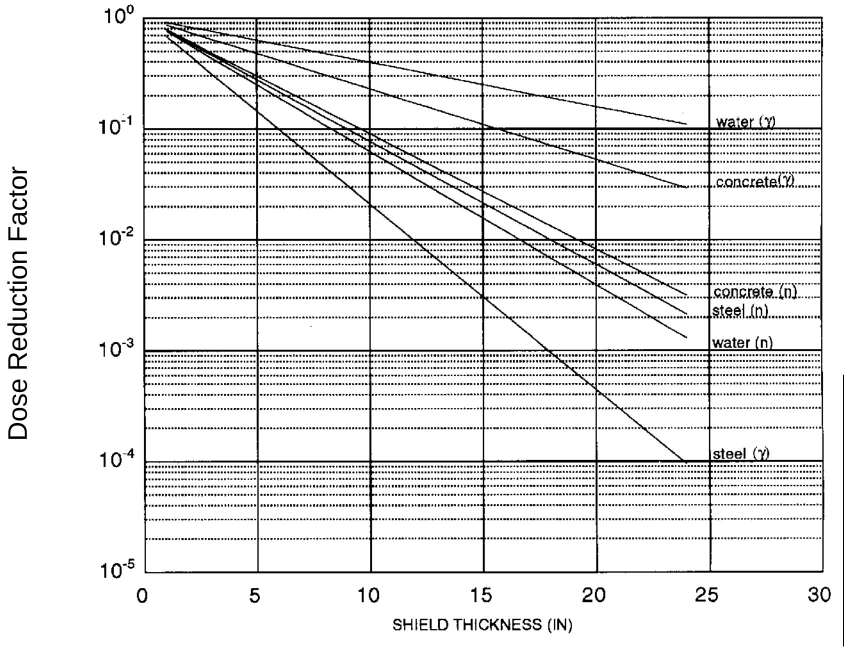

Figure 3. Dose reduction factors for various shield thicknesses.

Fission Yield Prediction Models

To obtain a tool for estimating the magnitude of the fission yields for potential accident situations, the present work [2] relied on the theory of Hansen [4]. The yield information provided by this methodology is approximate yet 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.

To facilitate the use of this theory it was necessary to determine values for the neutron lifetime (J), the equivalent neutron source strength (S) for the system, and a characteristic parameter (b) that describes the negative reactivity feedback provided to the system as the fission energy input to the system increases. The value of S is taken to be about 106 neutrons/s-kg of uranium for the high-enriched solutions [10] and about 40 neutrons/s-kg U for 5 wt % enriched uranium solutions. The value of S for the 5 wt % solutions was calculated via the ORIGEN-S module of SCALE [6]. These neutron sources include both spontaneous fission and production from alpha particle interaction with predominately oxygen in the water. The values of J are based on calculations performed with uranium densities of 25 to 2500 grams U/L using the KENO V.a module of SCALE [6]. The actual value of J 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 [11] and an expression developed for the determination of the b value as a function of system volume. 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.

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 [12]. The 24 CRAC experiments considered covered a limited range of the parameters with high-enriched 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. The work of Ref. 11 indicated that a factor of about 1.7 from the average yield satisfactorily enveloped the expected statistical variations in the initial burst fission yields. A comparison of this fission yield model with the CRAC experiments indicates the predicted results are within a factor of 1.7.

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. Instead, independent verification was sought by comparing the model predictions with the results from a series of calculations [13] performed with a 1-D coupled

(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. 5 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 vertical 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 [14]. 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.

The complete functional slide rule is provided in Ref. 3 and will be demonstrated at the paper presentation. As readily noticed within this paper there is an abundance of mixed English and metric units used throughout 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-gal tank, 16-in.-diam pipes/tubes, 2-gal/min 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 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.

Conclusion

An effective and efficient tool has been prepared for assisting in the evaluations necessary during emergency response to criticality accidents in nonreactor facilities. The slide rule has been designed to be applicable to five generic types of systems involving ^{235}U. Expansion of the slide rule to other generic systems involving MOX or plutonium is possible.

Acknowledgments

The authors are appreciative of the technical guidance and support of M. L. Thomas of the U.S. Nuclear Regulatory Commission during the technical development of the slide rule data and during the rather difficult preparation of the actual sliding graphs for publication. Development of the slide rule was funded by the U.S. Nuclear Regulatory Commission.

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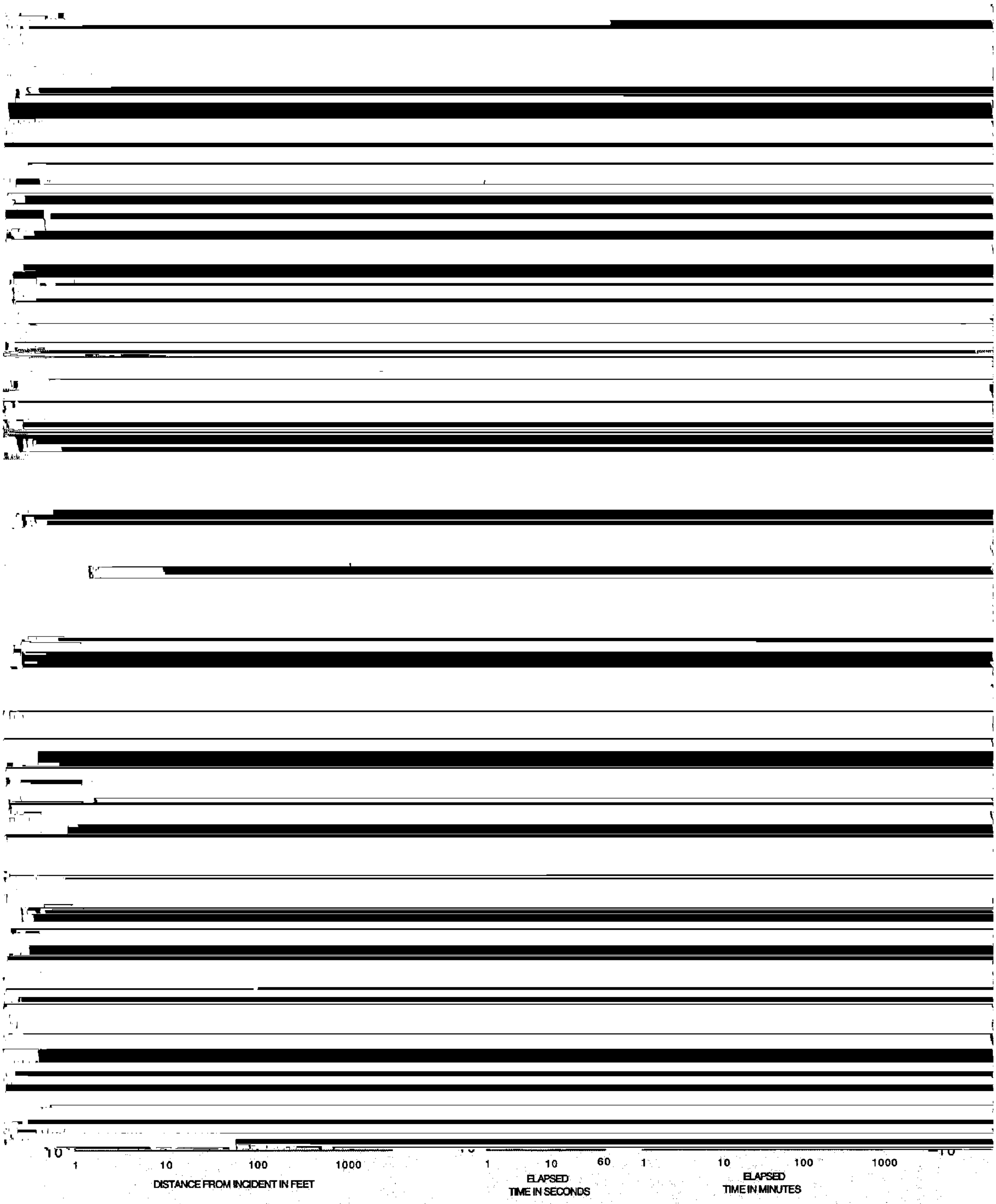

Figure 4. Damp $U(4.95)O_2F_2$ @ $H/^{235}U = 410$.

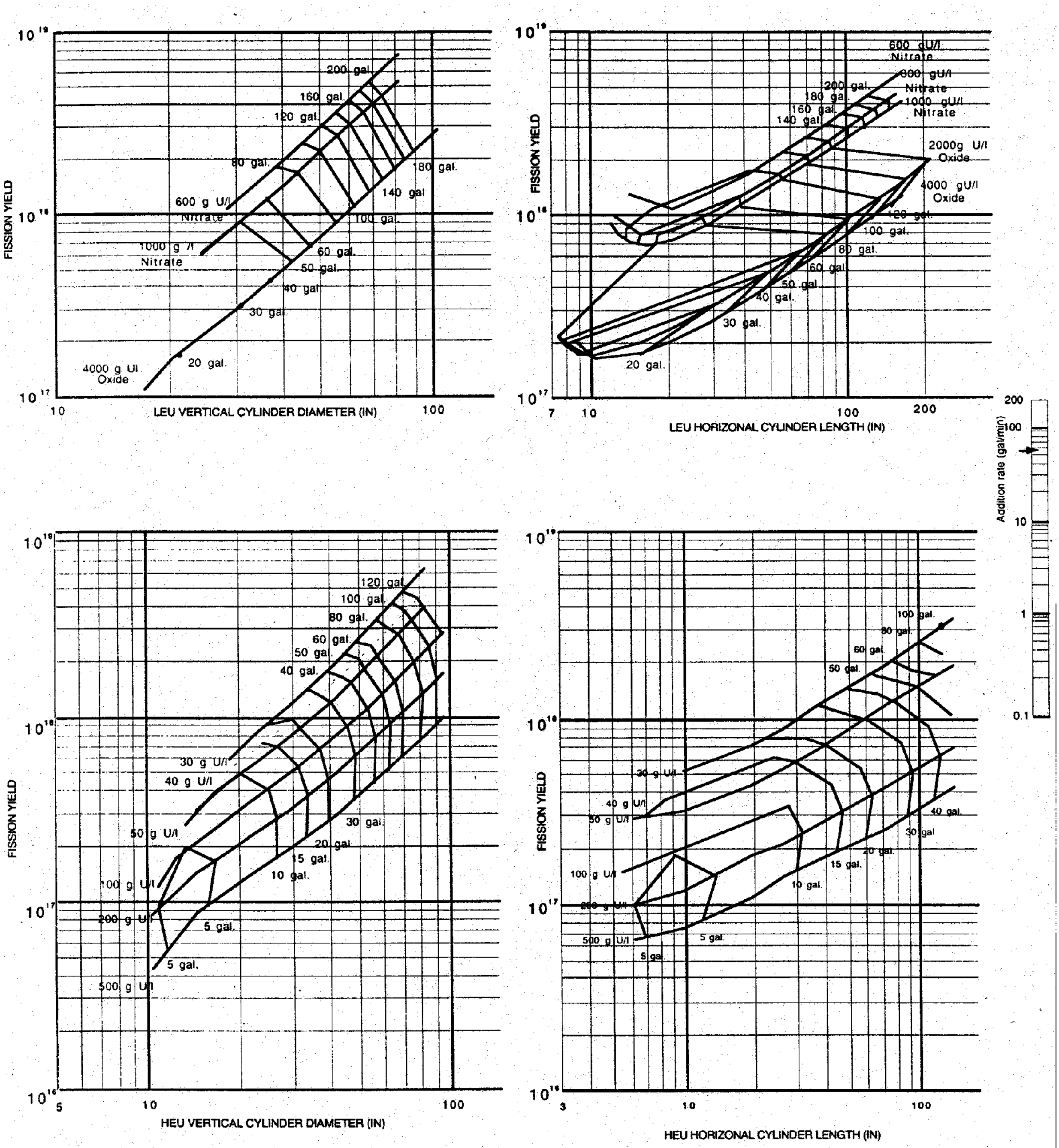

Figure 5. First pulse fission yield estimate for LEU and HEU in cylindrical geometries.