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THE CRITICALITY ACCIDENT

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1. Empirical Model to Estimate Energy Release from Accidental Criticality, *A. R. Olsen, R. L. Hooper (BNW), V. O. Uotinen (B&W), C. L. Brown (BNW)*

Each new plant that fabricates, processes, or otherwise handles fissionable materials undergoes a safety analysis prior to startup. One part of the safety analysis is to assess the potential consequences of a postulated worst credible criticality accident. To perform this assessment, the total energy release—expressed as "total number of fissions" from the criticality excursion—must be estimated or assumed. This paper presents the results of study for development of an empirical model to estimate energy release from a criticality accident.

In a plant where the assumed worst credible criticality accident is in a fissile solution system, the present industry practice is to assume a preestablished upper limit of 4×10^{19} fissions. This value is based somewhat on the past history of criticality accidents and the fact that the highest excursion to date resulted in an estimated 4×10^{19} fissions.

Recently, a summary of the results of a series of criticality excursion experiments with highly enriched uranium solutions conducted in France by the Commissariat a l'Energy Atomique (referred to as the CRAC experiments), was published.[1] These experiments have provided the first firm basis for developing an empirical model for predicting the total energy release from a criticality excursion in a given solution system.

The criticality accident, as characterized by the model empirically derived from the experimental data provided by the CRAC experiments, is considered to be divided into (a) an initial fission burst followed by (b) a plateau period where the number of fission/sec, ignoring oscillations, decreases with increasing time in the plateau.

The model for the initial burst relates the total fissions in the burst to a function of the volume (liters) of solution at the time of the burst. The empirically derived equation is

$$F_B = 2.95(10)^{15}\, V_B^{0.82} \quad .$$

The estimated 95% upper confidence level is given by

$$\log(F_B^U) = 15.47 + 0.82 \log V_B^0 + 0.23 \left[1.04 + \frac{(\log V_B^0 - 1.73)^2}{4.07}\right]^{1/2} \quad .$$

Figure 1 presents the relationship. This is assumed to hold for the vessel diameter range (300 to 800 mm) and solution addition rates used in the CRAC experiments.

The model developed for predicting the number of fissions in the initial burst is mainly applicable to highly enriched uranium systems, since this was the material used in the CRAC experiments. However, this model is also applicable to plutonium and slightly enriched uranium systems, though the energy release predicted will be conservatively high because of the presence of ^{240}Pu and ^{238}U, respectively, and other isotopes that undergo spontaneous fission. Considering the CRAC experiments with a neutron source added, we estimate that for plutonium systems, the fissions in the initial burst could be a factor of 2 or more lower than predicted by the model. A reduction factor has not been estimated for slightly enriched uranium.

The model for the number of fissions in the plateau is given by

$$F_p = 3.2(10)^{18}\,[1 - t^{-0.15}]$$

with the 95% upper confidence level given by

$$F_p^U = 4.6 \times 10^{19}\,[t_i^{0.02} - 1] \quad ,$$

where t is the duration of the plateau in seconds and F_p is the number of fissions in the plateau. The model is developed from an empirically derived upper-envelope fission/sec plateau time plot and then integrated over the duration of the plateau (t). This relationship is presented in Fig. 2.

Thus, an estimate of the total number of fissions occurring during a criticality accident is obtained from $F_T = F_B + F_p$, the latter term generally being the main contributor.

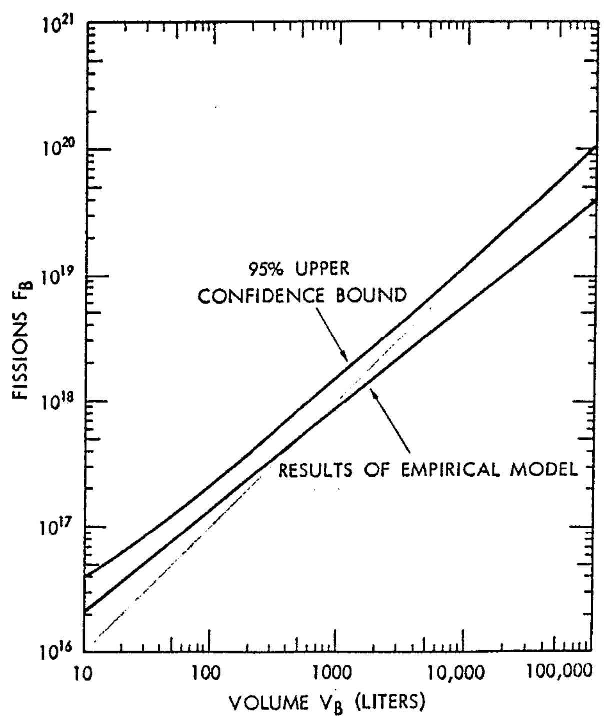

Fig. 1. Least-squares estimated relationship between the number of fissions in initial burst and the volume at pulse peak. (Based on CRAC experiments without external neutron sources. Upper curve gives the 95% upper confidence bound.)

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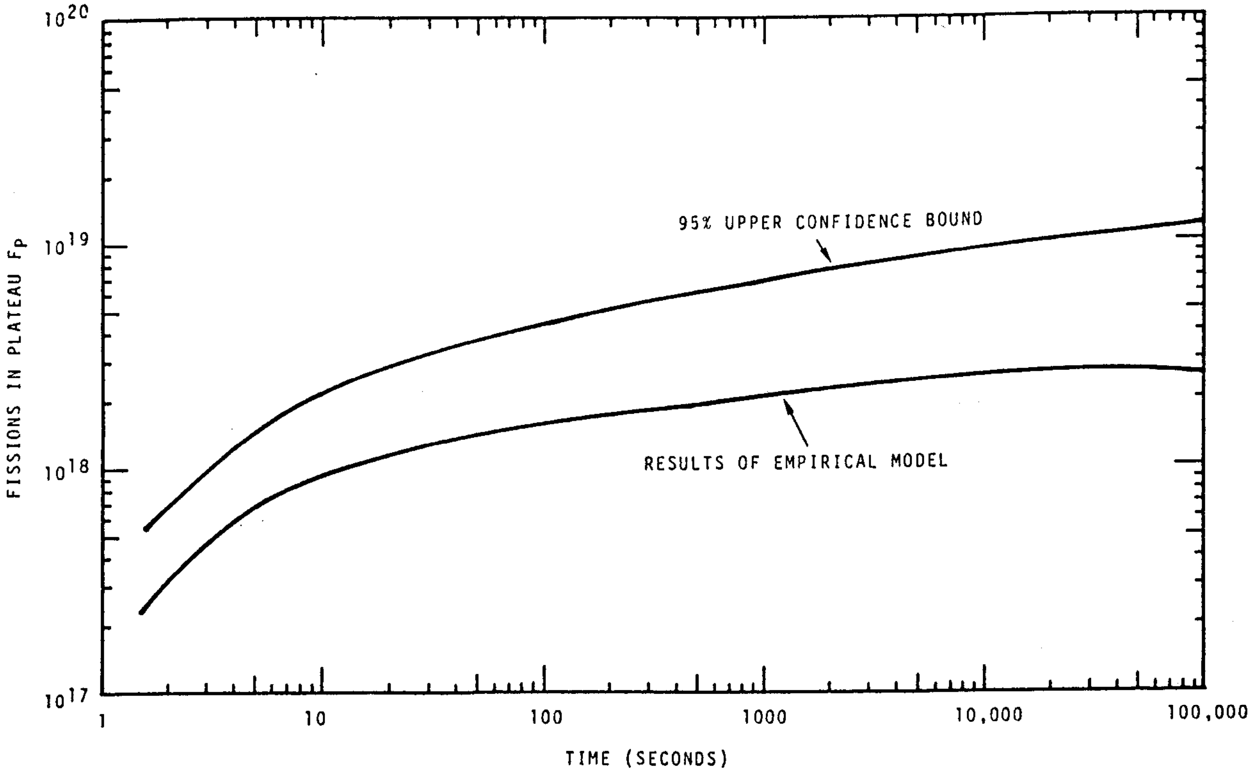

Fig. 2. Estimated total number of fissions occurring in plateau region of an excursion as a function of the duration of plateau. (The 95% upper confidence bound is presented as the upper curve.)

TABLE I

Comparison of Total Energy Release Predicted by Model and Actual Energy Release of Past Criticality Accidents

	Characteristics of Fissile Material System				Number of Fissions (x 10^{17})			
					Actual Excursion		Predicted By BNW Model	
Location	Form	Mass, kg Fissile	Volume, liter	Duration of Excursion (min)	Initial Burst	Total	Initial Burst	Total
Y12	$UO_2(NO_3)_2$ [a]	2.5	56	13	∿ 0.1	13.0	0.8	21.0
LASL	Pu/Organic	3.3	168	< 1	1.5	1.5	2.0	16.7
IF	$UO_2(NO_3)_2$ [a]	34.5	800	20	1.0	400.	7.1	28.1
IF	$UO_2(NO_3)_2$ [a]	8.0	40	< 1	∿ 0.6	6.0	0.6	15.3
Hanford	Pu Complex	1.5	∿ 60	2220	∿ 0.1	8.0	0.9	27.4
Wood River	$UO_2(NO_3)_2$ [a]	2.6	∿ 70	< 1	1.1	1.3	1.0	15.7
Windscale	Pu/Organic	2.5	∿ 100	< 1	0.01	0.01	1.3	16.0
ORNL	$Pu(NO_3)_4$	1.15	64	< 1	∿ 0.8	0.8	0.9	15.6
ORNL	UO_2F_2	18.3	55	< 1	0.5	0.5	0.8	15.5
ORNL	$^{233}UO_2(NO_3)_2$	∿ 1.0	5.8	< 1	0.11	0.11	0.1	14.8

[a]Uranium enrichment ∿ 93 wt% ^{235}U.

Application of the model to find the potential energy release from a criticality accident for use in a plant safety analysis could be accomplished by listing the vessels to be considered, estimating a potential duration for the excursion based on past accidents, calculating for each vessel the total number of fissions using the model presented, and then selecting the highest number of fissions.

To test the model, the conditions of the seven past industrial criticality accidents[2] and three critical-mass laboratory accidents were used to calculate initial burst and total energy release. Results are presented in Table I. The model is conservative in all but one instance.

1. P. LECORCHE and R. L. SEALE, "A Review of Experiments Performed to Determine the Radiological Consequences of a Criticality Accident," Y-CDC-12, Criticality Data Center (Nov. 1973).

2. W. R. STRATTON, "A Review of Criticality Accidents," LA-3611, Los Alamos Scientific Lab. (Sep. 1967).

2. Accidental Fissile Solution Excursions and Building Design Criteria, *G. Tuck (Dow-Colo)*

In the past few years, the AEC has required solution excursion analyses for new construction as well as plants already in operation. These hypothetical excursions should be chosen to be representative of the most severe excursion that could be reasonably expected in the particular systems involved since the results of the analysis will, in turn, determine the size and types of the hot exhaust filtering system and shielding thickness, etc. An example of the "most severe" excursion may be helpful. Consider a tank 3 ft in diameter by 4 ft high which is shown by a safety review to have solution. For the excursion parameter of maximum total fissions, the most severe condition is a tank that is critical when nearly full; hence, the full tank volume is used for the total fission estimate. The specifications of the process systems are usually such that the excursions chosen are limited by practical design considerations to tanks of 2- to 5-ft diam with a height of 1 to 2 diam and a solution transfer rate of $7\frac{1}{2}$ gpm or less.

All of the available solution excursion data are by no means directly applicable to excursion analysis. The data from accidental solution excursions have some limitations for our purposes. Of the five excursions in critical-mass laboratories, all involved volumes of <65 liter, three were in spherical geometry, and three involved the movement of control or scram rods. None of these characteristics is likely to be chosen for a reasonable most severe process plant excursion although the 65-liter volume is perhaps typical of the volumes involved in the smaller plant-type excursions.

All seven of the industrial excursions provide useful data. Two are representative of particular types: annular geometry and concentration buildup. Of the remaining, two involved volumes of <65 liter, two more for which the volumes are not given in the published excursion tables,[1] and one in an 800-liter volume. Summarizing the accidental solution excursion data, there appear to be two special cases, two typical smaller excursions at volumes of <65 liter but only one at the more reasonable process-area tank volume of 800 liter.

The French CRAC[2] data are directly applicable to the usual uranium solution process plant situations with respect to variables of the range of fill rates and solution concentrations. The data available are applicable for the first burst of a solution excursion for tanks of about 1-ft diam and for $2\frac{1}{2}$-ft diam up to an 18-in. height. Unfortunately, there are no data above 18 in. for the $2\frac{1}{2}$-ft-diam tank nor for the larger tanks which are more typical of the Rocky Flats areas. Of those experiments that lasted longer than 30 min, only one had much reactivity added after critical. Also, to use these data for plutonium solutions, the difference in neutron lifetimes, reactivity addition rates, neutron sources, and shutdown coefficient must be accounted for. The CRAC program was an extremely worthwhile project and these comments are intended only to point out the limitations of the CRAC data for plant accident-analysis purposes.

To extend the experimental excursion data to the range of tank sizes and to plutonium solutions typical of the Rocky Flats process areas, the CRAC data together with some of the KEWB[3] and industrial accident data were analyzed. This involved writing a simple excursion code utilizing the reactivity addition rate, the neutron lifetime, and a shutdown coefficient; deriving methods of obtaining these inputs; one series of calculations to validate the code and inputs with existing data; and another long series of calculations to evaluate the range of hypothetical plant excursions. It was found that for each tank diameter and size, there was a particular situation that produced the most severe case for each of the excursion parameters. These most severe cases were expressible in terms of simple equations and are given in Ref. 4. With these equations, one can start with the variables of tank diameter, height, and fill rate and easily estimate the excursion results needed for most building design criteria. It should be emphasized that the results obtained with these equations are not for a typical excursion but for a reasonable "most severe" excursion for the system involved.

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

2. PIERRE LÉCORCHÉ and R. L. SEALE, "A Review of the Experiments Performed to Determine the Radiological Consequences of a Criticality Accident," Y-CDC-12, Carbide and Carbon Chemical Corp. (Nov. 1973).

3. M. S. DUNENFELD and R. K. STITT, "Summary Review of the Kinetics Experiments on Water Boilers," NAA-SR-7087, North American Aviation (Feb. 1963).

4. GROVER TUCK, "Simplified Methods of Estimating the Results of Accidental Solution Excursions," *Nucl. Technol.*, 23, 177 (1974).

3. Analytical Studies of Criticality Accidents in Aqueous Solutions, *H. M. Forehand, Jr. (ANL-Idaho)*

A simple mathematical model has been constructed for simulating nuclear criticality accidents involving aqueous solutions.

The effect of radiolytic gas was demonstrated in the kinetics experiments on water boiler (KEWB)[1] program, a study designed to define the safety characteristics and dynamic behavior of the "water boiler" class of reactors. Radiolytic gas was observed to be the dominant shutdown mechanism for excursions with inverse periods above 100 $\sec^{-1}$.

In late 1968, the Service d'Etudes de Criticité of the French Commissariat à l'Energie Atomique initiated a program of systematic experimental nuclear excursions initiated intentionally to obtain realistic criticality accident data. This program was designated Conséquence Radiologiques d'un Accident de Criticité (CRAC).[2–5] Uranyl nitrate solutions of various concentrations were pumped at a steady flow rate into vertical cylinders (30- and 80-cm diam), simulating the accidental assembly