Chapter V

CRITICALITY SAFETY IN PROCESSING PLANTS

A. Plant Features with Criticality Potential

Processing plants contain a multiplicity of work stations, and areas for both long-term and short-term storage. Criticality safety considerations go beyond the analysis of each of these in terms of subcritical individual units or storage arrays. The progression of fissile material through a plant involves transfers and special handling during which unusual conditions may be encountered. It is important that these operations be governed by procedures and be carried out by well-trained personnel.

Consider a plant for processing highly enriched uranium as solids, such as fabrication of weapon components or fuel elements for reactors. It is essential to avoid the effect of massive fissile units falling together or encountering other units as the result of an accident with transfer equipment. Minimum spacing between units can be maintained by the use of birdcages, provided there are appropriate procedures for loading and unloading them.

In a plant for scrap recovery or processing irradiated fuel, the operations involving fissile solutions must be carefully planned. It is noteworthy that all criticality accidents that have occurred in processing plants have involved solutions. Mishaps that have led to these accidents include solution leakage, precipitation, dissolution of solids, instrument failure, and transfer among vessels. Avoidance of these mishaps calls for continued cooperation of criticality safety and operating personnel.

In general, both physical and administrative criticality safety practices must be tailored to specific plant conditions. This requirement inevitably will require judgment. Special evaluation also may be required because there is no "standard" plant for which universal criticality safety recipes can be defined.

B. Administration

Provisions of Standard *ANSI/ANS-8.19, Administrative Practices for Nuclear Criticality Safety*, are of major significance in processing plants. This Standard recognizes that criticality safety requirements must contribute to the physical and economic functions of a plant in a balanced manner. Accordingly, it places no requirement on the form of plant organization. Instead, requirements of the Standard are expressed in terms of management, operational supervisors, and a criticality safety staff provided by management.

The Standard emphasizes that effective criticality control, like other branches of safety, requires the positive support of management and implementation by supervisors with assistance of the criticality safety staff. It identifies associated responsibilities, calls for effective training of personnel and concise operating procedures, and has sections on process evaluation, material control, and planned response to criticality accidents.

C. Training

The training program for persons involved in operations with fissile material should make safety considerations, including criticality safety, an integral part of a program that provides necessary job skills. Standard *ANSI/ANS-8.20, Nuclear Criticality Safety Training*, applies to personnel associated with operations where there is the potential for a criticality accident. Provisions of the Standard are consistent with the precept that safety education will be most meaningful and readily assimilated if it is clearly relevant to operations. It follows that local supervision should participate in criticality safety training, or conduct it with the support of criticality safety specialists. Appropriate training of supervisory personnel is implied.

The Standard calls for training in the recognition of criticality alarms and the proper response to them. Training should be supported by discussion of selected criticality accidents. Stratton's history of nuclear accidents[30] describes each in sufficient detail to be helpful for this purpose. Accounts of real accident experience in training talks can help keep the audience awake.

D. Criticality Alarms and Response

Criticality alarms have twice initiated lifesaving evacuation of areas in which accidents occurred.[30] The value of such systems is therefore clear in areas for processing significant amounts of fissile material. Guidance for the design, installation, and maintenance of such systems may be obtained from Standard *ANSI/ANS-8.3, Criticality Accident Alarm System*. This document directs that an accident alarm system must be considered for any area containing more than a threshold quantity of fissile material. The Standard calls for an easily recognized signal for immediate evacuation in case of an alarm. It recommends that the response of the alarm system to radiation be tested at least monthly, each signal generator be tested at least once every three months, and an evacuation drill be performed at least annually.

The existence of an alarm system carries with it certain responsibilities. The system must be maintained to provide confidence that it will function if needed, and to minimize the frequency of false alarms. False alarms can have a negative impact on safety by creating a potential for injury as a result of precipitous response. False alarms also tend to destroy confidence in the system. Unannounced drills are not endorsed.

The response to an alarm is to be governed by an emergency plan with elements given in the administrative Standard *ANSI/ANS-8.19*. Further features of an emergency plan are being considered.

Elements of the emergency plan include procedures for evacuation to specified assembly stations, actions after assembly, and treatment of injured and exposed persons in accordance with advance arrangements.

Personnel must be trained in their proper response to the alarm including the use of evacuation routes and designated assembly points. Emergency plans must be kept current; evolution of a plant can influence the procedures to be followed in the event of an alarm.

E. Material Control

One criticality accident occurred because a concentrated fissile solution in a polyethylene cylinder was mistaken for a dilute solution.[30] This occurrence emphasizes the value of labeling or other positive identification of fissile material in helping to avoid routing errors within a plant. Also of value are posted limits at work stations and storage areas. If observed, for example, in the transfer of material along a glove-box line, posted limits can prevent inadvertent overloading of a box.

Labeling and posted limits cannot take the place of up-to-date procedures used by well-trained personnel, but should make errors less likely. Computerized accounting procedures, such as proposed for safeguards, should contribute further to the reduction of transfer errors.

Provisions for handling fissile material during inventories must be as carefully planned as for regular plant operation. This need is emphasized by the three criticality accidents that resulted from misdirection of solutions during inventory.[30]

An occasional requirement that should be anticipated is the emergency storage of fissile material that can accumulate as the result of interruptions of normal operations. Mishaps such as faulty processing or equipment failure may interrupt the flow of solutions, and accidents or other disruptions may prevent material from leaving the plant.

In a plant layout, the convenience of proper operations should be considered. To be avoided, for example, is transfer of material through a working area when another convenient route is available, and unnecessary processing of different fissionable materials in the same area. To illustrate, use of the same furnace for casting enriched and natural uranium, except during independent campaigns, could contribute to the confusion of feed items. An example of making mishaps inconvenient is to transfer fissile material on a single plane, as with special carts. Transfer by crane over other fissile material would be objectionable.

F. Process Startup

Before initial operation of a plant, or of a module that is new or revised, confirmation of the proper condition of its components is mandatory. Confirmation includes testing of instrumentation, valves, seals, transfer devices, and ventilation and fire-protection equipment. At this point, adequacy of training should be established.[26]

It is also important to reassess criticality safety before startup. The initial assessment can be influenced by evolutionary changes during construction. Even though the effect of each change has been considered, the as-constructed configuration should be examined.

At this stage, it is appropriate to reconsider matters of judgment about the adequacy of the experimental basis for evaluating the criticality safety of operation. Judgment is involved in decisions concerning the appropriateness of directly applicable experiments, of experiments used for validating calculations, or of additional safety margins applied when validation is questionable. Any doubt usually can be resolved by means of neutron-multiplication measurements as outlined in Standard *ANSI/ANS-8.6, Safety in Conducting Subcritical Neutron-Multiplication Measurements in Situ.* These measurements, conducted during stepwise introduction of fissile material, would identify safely subcritical conditions. In general, they would simply provide reassurance that normal operation is acceptable. They must not cause personnel to relax concerning accident potential.

G. Maintaining Safety Provisions

During plant operation, continuous observation and periodic surveys are means of guarding against adverse effects of evolutionary change in conditions or practices. Has a vessel that could contain more than a critical volume been brought into a process area? Has equipment for fissile material been used for other material? Should features of fire protection be reviewed because of changed plant content? Have precautions against the consequences of natural disasters such as earthquake, flood, or tornado been relaxed over time? The list of questions does not stop here. In fact, it depends on detailed plant features, regulations, and the policy of plant management. Thus, the wish for a universal check list would be futile.

H. Examples of Plant Application

1. Dissolver for Water-Reactor Fuel

The safe geometry of a 100-liter dissolver for chopped $U(3.2)O_2$ fuel elements is to be explored. The shape of the dissolver should be simple and it is to be surrounded by a steam jacket. Full water reflection should be assumed to allow for water in the steam jacket and for incidental reflection.

Table 9 shows a limiting value of 26.4 cm for the subcritical diameter of a long cylinder of heterogeneous oxide.[86] This value is essentially the inside diameter of 10-inch Schedule-5S pipe. The diameter limit for solution is significantly greater. Because a cylinder of this diameter has a capacity of 55 liters per meter of length, the height of a 100-liter dissolver would be about 1.8 m. A design study will show whether this height meets functional requirements.

Should this long, small diameter prove to be undesirable, an alternative would be an annular tank surrounding a neutron-absorbing material to reduce neutron exchange within the configuration. If the absorbing material is water and the inside diameter is at least 30 cm, the annular thickness can be approximated by a reflected infinite slab specified in Table 9 to be 12.6-cm thick. If additional conservatism is desired, a thickness of 10 cm and an inside diameter of 40 cm may be assumed for the design study, the capacity of which is about 157 liters per meter. Accordingly, a vessel of 100-liter capacity would have near-equilateral external dimensions. Before adoption, the acceptability of the final design should be confirmed either by a validated calculation or by *in situ* neutron-multiplication measurements.[29]

Of course, this dissolver encompasses more than the simple container. In the first place, to accommodate irradiated fuel, it must be one component of a shielded fuel-handling system. The container must be modified for introduction of the chopped fuel, draining solution, and withdrawal of residual solids. Sparging to facilitate uniform dissolution also may prove desirable. The ultimate criticality safety evaluation must take into account auxiliaries and interaction with other components.

Further, there may be special requirements for campaigning fuels from different sources, for instance, the fuel up to 4 wt% ^{235}U in the following example of plant application. If the possibility of handling fuel at somewhat more than 3.2 wt% ^{235}U can be foreseen, it should be more effective to plan for it at this stage than to adapt to it later. Actually, the "conservative" annular thickness of 10 cm may prove to be suitable for fuel enrichments of nearly 5 wt% ^{235}U.

2. Storage of Low-Enriched Uranium Solution

Consider vessels for storing a variety of uranium solutions in which the ^{235}U enrichment will not exceed 4 wt% and the uranium density will remain below 750 g/L. A total capacity of 1890 liters (500 gal) is desired, and, because of the possibility of long-term storage and the difficulty of internal inspection, a single vessel packed with Raschig rings is not selected. The preferred arrangement is a planar bank of cylinders near a 12-m-long, 5-m-high concrete wall, with a narrow walkway between the cylinders and wall.

According to Table 8, the subcritical limit on cylinder diameter for U(4) solution is 30 cm; the next smaller commercial pipe size is 10-inch Schedule-5S (26.6-cm-i.d.). At a usable height of 4.6 m, the capacity per cylinder is 250 liters and 8 cylinders would be required. Construction and operational convenience would be met by a one-meter center spacing of cylinders and would result in additional space at the ends of the bank of cylinders.

A walkway of 0.7 m separates the cylinders from the concrete wall and reduces the effect of the wall to that of incidental reflection on each vessel. Because the 30-cm diameter limit is based on full water reflection, which is much more effective than incidental reflection, it is necessary to show that the effect of interaction among the cylinders is acceptable. According to validated KENO calculations,[147] $k_{eff} = 0.725$ for a single cylinder having only 2.5-cm-thick water reflection, and $k_{eff} = 0.785$ for the linear array spaced from the concrete wall, showing that interaction is adequately small. Thus, it is appropriate to proceed with the design of this arrangement and with detailed exploration of contingencies.

The low values of k_{eff} suggest the reasonableness of further investigation of a storage bank with significantly increased capacity. For example, a one-dimensional calculation of a 12-inch Schedule-5S pipe (31.5-cm-i.d.) instead of the 26.6-cm pipe resulted in a k_{eff} of 0.9. The capacity of 8 cylinders at the 4.6 m height would be increased to 750 gallons. Of course, a careful computational study and analysis of contingencies would be required before adopting this approach.

3. Solution in Tanks Packed with Boron-Containing Raschig Rings

In certain cases, as noted before, an alternative to geometrically subcritical tanks for solution storage is the use of large capacity tanks packed with borosilicate-glass Raschig rings. Typically, although one-quarter to one-third of the tank volume is sacrificed to the glass absorber, the tank may still accommodate large volumes of solution more efficiently than long, limited-diameter cylinders or thin slab-like containers. Other than for primary criticality control, Raschig rings in auxiliary tanks may protect against accidental criticality resulting from inadvertent diversion of fissile solution to those tanks.

American National Standard Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material, ANSI/ANS-8.5, defines appropriate conditions for criticality control. Restrictions exclude the use of alkaline solutions, HF, and hot, concentrated H_3PO_4. Temperature and radiation fields also are limited. The Standard defines chemical and physical properties that are typified by Pyrex type 7740 and Kimbal type KG-33 and limits the ring size to 3.81-cm-o.d. It specifies packing conditions and gives requirements for inspection and maintenance. Finally, maximum densities of fissile material in vessels of unlimited size are specified for three different volume percentages of glass. Typically, as the glass volume fraction ranges from 0.24 to 0.32, density limits range from 150 to 200 g/L for ^{233}U, from 270 to 400 g/L for ^{235}U-enriched uranium, from 115 to 180 g Pu/L for ^{239}Pu, and from 140 to 220 g Pu/L for plutonium containing more than 5 wt% ^{240}Pu.

Although it is unlikely that these reasonably generous limits would restrict a practical process, there could be unusual circumstances that would require greater glass fractions. Because computational models cannot closely approximate randomly packed Raschig rings,[148] the preferred guidance for increased limits would be experimental data near the desired conditions or computational results verified by *in situ* neutron multiplication measurements.[29] An example of an experimental system that is subcritical at a plutonium density greater than that permitted by the Standard is reported by Lloyd, Bierman, and Clayton.[137] The subcritical density of plutonium (8.3 wt% ^{240}Pu) in nitrate solution was 391 g/L when a 61-cm-diameter tank was filled to a depth of 99.1 cm. Raschig rings containing 4.0 wt% boron occupied 18.8% of the volume, and there was an effectively infinite water reflector on the tank walls and base.

Nurmi[149] reports the use of borosilicate-glass rings with enriched uranium solutions that have free fluoride-ion contents greatly exceeding the limit specified in the Standard. Because of this deviation, there is daily visual inspection and semiannual emptying of tanks for detailed examination. This is a more stringent maintenance schedule than that required by the Standard.

Another approach to environments that are hostile to borosilicate glass is suggested by experiments at Battelle Pacific Northwest Laboratories[137] with plutonium solutions in a tank packed with stainless steel Raschig rings containing 1.0 wt% boron. A 45.7-cm-diameter tank, water reflected on sides and bottom, was packed with 1.27-cm-o.d., 1.27-cm-long steel rings occupying 27.0% of the volume. At a depth of 99.1 cm, plutonium (8.3 wt% ^{240}Pu) solutions at densities of 275 g Pu/L with 480 g NO_3/L and of 412 g Pu/L with 602 g NO_3/L were subcritical.

A further example includes data on plutonium-uranium nitrate mixtures in a 61-cm-diameter tank, water reflected on the sides and bottom and packed with glass Raschig rings containing 4 wt% boron.[137–138] The rings, which were 3.81-cm-o.d. and 4.32 cm in length, displaced 18.8% of the solution volume. At a depth of 90.4 cm, solution at a density of 180 g U/L (0.66 wt% ^{235}U in U) and 78.4 g Pu/L (5.7 wt% ^{240}Pu in Pu) containing 377 g NO_3/L was subcritical.

4. Solution Holdup Design

A cell in a U(93.2) reprocessing facility has a concrete floor area of 9 m^2 and analyses have shown that the neutron interaction between the process vessels and between the vessels and the floor is negligible. The floor with sidewalls will serve as a catch basin for solutions that may leak from the vessels. An overflow line is to be installed in the floor, draining to a poisoned catch tank, thereby limiting the thickness of solution. The maximum expected ^{235}U density in $UO_2(NO_3)_2$ is 250 g/L. A permitted solution height over the floor is to be determined. The configuration of the solution is conservatively approximated by an effectively infinite uniform slab with a thick concrete reflector on one side and incidental reflection on the other side.

From Table 1, the specified subcritical thickness of an infinite slab of $UO_2(NO_3)_2$ reflected by 30-cm-thick water is 4.9 cm. A thick water reflector on both surfaces is expected to be more effective than concrete reflection on one and incidental reflection on the other. It follows that the specified height of the overflow pipe should not exceed 4.9 cm. The chosen height should be measured from the lowest portion of the floor as established by an elevation survey.

APPENDIX

This Appendix provides a description of the calculational study leading to the curves presented in Figures 2 through 13 of this document. The motivation for this study was to provide quantitative examples illustrating the relationship between system reactivity (k_{eff}) and system geometry. Inferential in these curves are the partial derivatives of geometrical size (x) versus k_{eff}. Figures from report *LA-10860-MS, Critical Dimensions of Systems Containing* ^{235}U, ^{239}Pu, *and* ^{233}U, *1986 Revision*, were adapted to provide a basis for the illustration. This adaptation appears directly in Figures 2 through 13. The adaptation brings forward results from LA-10860-MS for experimentally determined critical systems. These data provide a reference to interpret the curves. The three well-established fissile nuclides ^{233}U, ^{235}U, and ^{239}Pu were selected for the construction of the examples. The ^{235}U was taken to be present as U(93.2). System compositions were taken to be metal-water mixtures and were selected to systematically span the entire range from limiting critical fissile density (7 to 13 grams per liter in water) to pure metal density (approximately 20 kilograms per liter). For these systems, the neutron spectrum varies systematically from a thermalized distribution for dilute fissile densities to a slightly softened fission spectrum for the pure metal systems. Three system geometries were selected to complete the set of examples: spherical, infinite circular cylinder, and infinite planar slab. In each case, the fissile-bearing region is surrounded by a tight-fitting pure water reflector of effectively infinite thickness. These are classic geometries which occur repeatedly in the literature of criticality safety. The first documented occurrence of these geometries and the associated characteristic curves, known to the editors, is found in the report *CP-400, Chain Reaction of Pure Fissionable Materials in Solution.*[150]

The metal-water systems used in the examples have no direct experimental analog. Uranium metal and plutonium metal are not, in a chemical sense, soluble in water. However, the metal-water mixtures are neutronically approached in an asymptotic sense for dilute fissile systems. In such systems the atomic ratio of the hydrogen to the fissile atomic species is very high (above 1000). In these systems, the other nuclear species needed for a chemical solution, such as nitrogen and fluorine, are also very dilute and have a minimum perturbing effect. Hence, these dilute systems approach the idealized metal-water mixture. Over the remaining range, however, the chemical constituents, such as nitrogen and fluorine, represent a serious perturbation from the idealized metal-water mixture. Hence, any comparison between calculational and experimental results requires a careful and accurate determination of the impact of the presence of these other nuclear species.

Caution should be exercised in the application of the curves presented in Figures 2 through 13. First, the reader should recognize that the curves represent calculational results. Second, the reader should note that these calculations do not conform to current validation and verification criteria. No attempt has been made to document a rigorous compliance with such criteria. That is, software and platform verification and the comparison of calculational results with experimental results have not been carried out as described in Chapter I, Section B-4 of this document, **The Role of Calculational**

Validation. Instead, we comply with the traditional criteria for reporting scientific results by providing sufficient detail to allow for independent reproducibility and confirmation of results.

The value of k_{eff} was calculated for a specific nuclide type, density, and system dimension (x). The dimension x corresponds to a spherical diameter, an infinite cylinder diameter, or an infinite slab thickness. For each nuclide type, density, and system geometry, four to five values of x were selected which resulted in calculated k_{eff}'s in the range 0.5 to 1.2. In addition the value of k_∞ for an infinite metal-water mixture was calculated. To determine the value of x for a particular value of k_{eff}, the appropriate set of calculational results were fitted to a continuous curve having the following algebraic form.

$$k_{\text{eff}}(x) \quad = \quad k_\infty \left(1 - e^{-\alpha x}\right)^{\beta x^\gamma}$$

In the above expression α, β, and γ are fitting parameters. This form provides a monotonically increasing k_{eff} versus x which asymptotically approaches k_∞ for large x. The curves shown in Figures 2 through 13 were generated by fitting a spline through the calculated values of x for each selected fissile density. The calculational results were produced using the MCNP Monte Carlo code (see Ref. 13). The cross-sections were based on ENDF/B-V cross-section evaluations provided by the XTM group at Los Alamos. Specifically, the MCNP nuclide identifiers (ZAIDs) shown in Table 16 were used.

Table 16

Nuclides, Cross-Section Evaluations, and Atomic Weights Used for Calculational Results

Nuclide	ZAID	Atomic Weight
^{1}H	1001.50c	1.00782475
^{16}O	8016.50c	15.99491480
^{233}U	92233.50c	233.03962900
^{235}U	92235.50c	235.04392497
^{238}U	92238.50c	238.05078549
^{239}Pu	94239.55c	239.05215781
Avogadro's number (atoms/b-cm)		0.602204345

The lwtr.01t version of the $S(\alpha,\beta)$ scattering model was used for the water in the metal-water mixture and for the water in the reflector.

Table 17 gives values of the mass densities assumed for water and for the metal state of each nuclide. Table 18 gives the number densities of hydrogen and oxygen used for the 15.2-cm water reflector. Tables 19 through 21 give the number densities calculated for 22 selected fissile mass densities for the three fissile nuclides ^{233}U, ^{235}U, and ^{239}Pu. Finally, Tables 22 through 24 list the final calculated geometrical dimensions (x values) used to produce the curves shown in Figures 2 through 13.

Table 17

Mass Densities Assumed for Water and Fissile Metal

Material	Mass Density (g/cm^3)
Water	0.997801
^{233}U Metal	18.05
^{235}U Metal (93.2 wt% ^{235}U)	18.76
^{239}Pu Metal	19.74

Table 18

Calculated Number Densities for the 15.2 cm Water Reflector

Nuclide	Number Density (atoms/b-cm)
^{1}H	0.066725294
^{16}O	0.033362647

Table 19

Fissile Mass Densities and Calculated Number Densities for ^{233}U Metal-Water Mixtures

^{233}U Mass Density (kg/L)	Number Density (atoms/barn-cm)		
	^{233}U	^{1}H	^{16}O
0.005	.000012921	.066706810	.033353405
0.006	.000015505	.066703114	.033351557
0.007	.000018089	.066699417	.033349709
0.008	.000020673	.066695720	.033347860
0.009	.000023257	.066692024	.033346012
0.010	.000025841	.066688327	.033344163
0.011	.000028425	.066684630	.033342315
0.012	.000031010	.066680934	.033340467
0.013	.000033594	.066677237	.033338618
0.014	.000036178	.066673540	.033336770
0.020	.000051683	.066651360	.033325680
0.030	.000077524	.066614393	.033307197
0.050	.000129206	.066540459	.033270230
0.100	.000258413	.066355625	.033177812
0.200	.000516826	.065985955	.032992978
0.500	.001292064	.064876948	.032438474
1.000	.002584128	.063028602	.031514301
2.000	.005168257	.059331909	.029665955
5.000	.012920642	.048241833	.024120916
10.000	.025841285	.029758372	.014879186
14.000	.036177799	.014971603	.007485802
18.050	.046643517	.000000004	.000000002

Table 20

Fissile Mass Densities and Calculated Number Densities for U(93.2) Metal-Water Mixtures

^{235}U Mass Density (kg/L)	Number Density (atoms/barn-cm)			
	^{235}U	^{238}U	^{1}H	^{16}O
0.005	.000012810	.000000923	.066706212	.033353106
0.006	.000015373	.000001107	.066702396	.033351198
0.007	.000017935	.000001292	.066698580	.033349290
0.008	.000020497	.000001477	.066694764	.033347382
0.009	.000023059	.000001661	.066690947	.033345474
0.010	.000025621	.000001846	.066687131	.033343565
0.011	.000028183	.000002030	.066683315	.033341657
0.012	.000030745	.000002215	.066679498	.033339749
0.013	.000033307	.000002399	.066675682	.033337841
0.014	.000035869	.000002584	.066671866	.033335933
0.020	.000051242	.000003691	.066648968	.033324484
0.030	.000076863	.000005537	.066610805	.033305403
0.050	.000128105	.000009229	.066534479	.033267240
0.100	.000256209	.000018457	.066343665	.033171832
0.200	.000512419	.000036915	.065962035	.032981018
0.500	.001281046	.000092286	.064817147	.032408574
1.000	.002562093	.000184573	.062909001	.031454500
2.000	.005124186	.000369145	.059092707	.029546354
5.000	.012810464	.000922863	.047643827	.023821914
10.000	.025620928	.001845726	.028562361	.014281180
14.000	.035869299	.002584017	.013297187	.006648594
17.484	.044796448	.003227126	.000000004	.000000002

Table 21

Fissile Mass Densities and Calculated Number Densities for ^{239}Pu Metal-Water Mixtures

^{239}Pu Mass Density (kg/L)	Number Density (atoms/barn-cm)		
	^{239}Pu	^{1}H	^{16}O
0.005	.000012596	.066708393	.033354196
0.006	.000015115	.066705013	.033352506
0.007	.000017634	.066701632	.033350816
0.008	.000020153	.066698252	.033349126
0.009	.000022672	.066694872	.033347436
0.010	.000025191	.066691492	.033345746
0.011	.000027710	.066688112	.033344056
0.012	.000030230	.066684731	.033342366
0.013	.000032749	.066681351	.033340676
0.014	.000035268	.066677971	.033338985
0.020	.000050383	.066657690	.033328845
0.030	.000075574	.066623888	.033311944
0.050	.000125957	.066556283	.033278142
0.100	.000251913	.066387273	.033193637
0.200	.000503827	.066049252	.033024626
0.500	.001259567	.065035190	.032517595
1.000	.002519134	.063345086	.031672543
2.000	.005038267	.059964879	.029982440
5.000	.012595668	.049824257	.024912128
10.000	.025191337	.032923220	.016461610
14.000	.035267872	.019402390	.009701195
19.740	.049727697	.000000003	.000000002

Table 22

Calculated Dimensions for ^{233}U Metal-Water Mixtures

^{233}U Mass Density (kg/L)	k_{eff}	Sphere Diameter (cm)	Infinite Cylinder Diameter (cm)	Infinite Slab Thickness (cm)
0.005	0.8	–	–	–
	0.9	–	–	–
	1.0	–	–	–
0.006	0.8	–	–	–
	0.9	–	–	–
	1.0	–	–	–
0.007	0.8	–	–	–
	0.9	–	–	–
	1.0	–	–	–
0.008	0.8	317.12	238.93	149.53
	0.9	–	–	–
	1.0	–	–	–
0.009	0.8	105.78	78.38	47.43
	0.9	–	–	–
	1.0	–	–	–
0.010	0.8	76.12	55.63	32.51
	0.9	179.13	135.31	84.71
	1.0	–	–	–
0.011	0.8	62.29	45.22	25.58
	0.9	104.73	76.90	46.12
	1.0	–	–	–
0.012	0.8	54.16	38.85	21.67
	0.9	80.25	58.24	34.28
	1.0	177.95	135.68	84.06
0.013	0.8	48.55	34.78	18.92
	0.9	67.26	48.71	28.02
	1.0	112.46	83.69	50.54
0.014	0.8	44.71	31.68	16.96
	0.9	59.44	42.72	24.04
	1.0	88.23	64.94	38.59
0.020	0.8	32.93	22.72	11.20
	0.9	39.88	27.93	14.47
	1.0	49.50	35.19	19.10
0.030	0.8	26.10	17.50	7.84
	0.9	30.40	20.72	9.87
	1.0	35.65	24.74	12.36
0.050	0.8	21.26	13.82	5.46
	0.9	24.30	16.09	6.84
	1.0	27.85	18.72	8.45
0.100	0.8	17.76	11.07	3.60
	0.9	20.08	12.79	4.64
	1.0	22.71	14.75	5.83
0.200	0.8	15.74	9.50	2.50
	0.9	17.80	11.00	3.38
	1.0	20.09	12.69	4.38
0.500	0.8	14.20	8.25	1.60
	0.9	16.11	9.63	2.37
	1.0	18.19	11.18	3.27
1.000	0.8	13.27	7.53	1.14
	0.9	15.11	8.87	1.84
	1.0	17.12	10.35	2.68
2.000	0.8	12.34	6.85	0.79
	0.9	14.07	8.11	1.39
	1.0	15.99	9.50	2.15
5.000	0.8	10.75	5.77	0.43
	0.9	12.32	6.89	0.87
	1.0	14.04	8.12	1.47
10.000	0.8	9.04	4.71	0.25
	0.9	10.40	5.66	0.54
	1.0	11.88	6.71	0.97
14.000	0.8	7.94	4.08	0.18
	0.9	9.19	4.91	0.41
	1.0	10.52	5.83	0.75
18.050	0.8	7.00	3.58	0.14
	0.9	8.11	4.31	0.33
	1.0	9.31	5.11	0.60

Table 23

Calculated Dimensions for U(93.2) Metal-Water Mixtures

^{235}U Mass Density (kg/L)	k_{eff}	Sphere Diameter (cm)	Infinite Cylinder Diameter (cm)	Infinite Slab Thickness (cm)
	0.8	–	–	–
0.005	0.9	–	–	–
	1.0	–	–	–
	0.8	–	–	–
0.006	0.9	–	–	–
	1.0	–	–	–
	0.8	–	–	–
0.007	0.9	–	–	–
	1.0	–	–	–
	0.8	–	–	–
0.008	0.9	–	–	–
	1.0	–	–	–
	0.8	130.68	98.23	60.42
0.009	0.9	–	–	–
	1.0	–	–	–
	0.8	87.11	63.90	37.71
0.010	0.9	1273.95	833.96	562.43
	1.0	–	–	–
	0.8	69.41	50.49	29.03
0.011	0.9	137.49	103.27	62.92
	1.0	–	–	–
	0.8	59.51	42.83	24.20
0.012	0.9	95.99	70.72	42.25
	1.0	–	–	–
	0.8	53.16	38.10	21.03
0.013	0.9	77.93	57.02	33.35
	1.0	173.45	129.51	80.40
	0.8	48.64	34.63	18.77
0.014	0.9	67.72	49.04	28.09
	1.0	115.37	85.54	52.14
	0.8	35.51	24.62	12.37
0.020	0.9	43.94	31.03	16.47
	1.0	56.68	40.69	22.63
	0.8	28.17	19.01	8.77
0.030	0.9	33.37	22.94	11.20
	1.0	40.18	27.98	14.42
	0.8	23.20	15.20	6.28
0.050	0.9	26.85	17.96	7.96
	1.0	31.22	21.28	9.99
	0.8	19.60	12.44	4.37
0.100	0.9	22.51	14.56	5.68
	1.0	25.82	17.02	7.19
	0.8	17.78	10.96	3.30
0.200	0.9	20.36	12.88	4.45
	1.0	23.32	15.06	5.79
	0.8	16.63	9.99	2.47
0.500	0.9	19.15	11.83	3.58
	1.0	22.06	13.98	4.88
	0.8	16.20	9.55	2.06
1.000	0.9	18.81	11.47	3.19
	1.0	21.76	13.72	4.52
	0.8	15.72	9.17	1.71
2.000	0.9	18.41	11.14	2.83
	1.0	21.53	13.41	4.19
	0.8	14.30	8.15	1.24
5.000	0.9	16.78	9.98	2.22
	1.0	19.64	12.10	3.45
	0.8	12.31	6.90	0.85
10.000	0.9	14.37	8.39	1.60
	1.0	16.68	10.09	2.56
	0.8	11.08	6.12	0.67
14.000	0.9	12.92	7.44	1.28
	1.0	14.97	8.92	2.07
	0.8	10.08	5.52	0.56
17.484	0.9	11.76	6.72	1.09
	1.0	13.62	8.05	1.77

Table 24

Calculated Dimensions for ^{239}Pu Metal-Water Mixtures

^{239}Pu Mass Density (kg/L)	k_{eff}	Sphere Diameter (cm)	Infinite Cylinder Diameter (cm)	Infinite Slab Thickness (cm)
0.005	0.8	235.44	178.13	114.24
	0.9	–	–	–
	1.0	–	–	–
0.006	0.8	84.59	61.77	36.54
	0.9	344.79	264.35	171.68
	1.0	–	–	–
0.007	0.8	61.59	44.46	25.04
	0.9	100.75	74.83	44.68
	1.0	–	–	–
0.008	0.8	50.95	36.39	19.95
	0.9	72.49	52.66	30.46
	1.0	135.53	101.00	61.68
0.009	0.8	44.84	31.70	16.89
	0.9	59.55	42.95	24.16
	1.0	89.85	66.19	39.16
0.010	0.8	40.65	28.53	14.89
	0.9	52.20	37.36	20.49
	1.0	72.23	52.75	30.27
0.011	0.8	37.65	26.24	13.38
	0.9	47.26	33.55	18.02
	1.0	62.44	45.03	25.40
0.012	0.8	35.38	24.51	12.27
	0.9	43.69	30.77	16.28
	1.0	56.03	40.18	22.32
0.013	0.8	33.54	23.12	11.39
	0.9	40.96	28.74	14.91
	1.0	51.55	36.76	20.01
0.014	0.8	32.06	21.99	10.66
	0.9	38.81	27.11	13.87
	1.0	48.15	34.15	18.34
0.020	0.8	26.90	18.05	8.10
	0.9	31.67	21.61	10.33
	1.0	37.68	26.22	13.16
0.030	0.8	23.31	15.29	6.26
	0.9	27.04	18.04	7.98
	1.0	31.52	21.42	10.08
0.050	0.8	20.63	13.16	4.81
	0.9	23.74	15.49	6.22
	1.0	27.37	18.19	7.90
0.100	0.8	18.66	11.58	3.63
	0.9	21.46	13.68	4.89
	1.0	24.68	16.09	6.34
0.200	0.8	17.66	10.73	2.94
	0.9	20.37	12.73	4.14
	1.0	23.48	15.07	5.55
0.500	0.8	16.75	10.00	2.35
	0.9	19.40	11.93	3.52
	1.0	22.44	14.18	4.88
1.000	0.8	15.86	9.33	1.97
	0.9	18.38	11.20	3.06
	1.0	21.26	13.35	4.33
2.000	0.8	14.56	8.45	1.55
	0.9	16.84	10.12	2.50
	1.0	19.42	12.03	3.62
5.000	0.8	11.98	6.74	0.96
	0.9	13.80	8.05	1.61
	1.0	15.82	9.53	2.42
10.000	0.8	9.27	5.07	0.57
	0.9	10.65	6.04	0.99
	1.0	12.14	7.09	1.49
14.000	0.8	7.76	4.19	0.42
	0.9	8.91	4.98	0.73
	1.0	10.13	5.84	1.12
19.740	0.8	6.23	3.34	0.31
	0.9	7.13	3.95	0.53
	1.0	8.08	4.60	0.81

References

1. A. D. Callihan, W. J. Ozeroff, H. C. Paxton, C. L. Schuske, "Nuclear Safety Guide," US Atomic Energy Commission report TID-7016 (1957).

2. J. T. Thomas, editor, "Nuclear Safety Guide, TID-7016, Revision 2," US Nuclear Regulatory Commission report NUREG/CR-0095, ORNL/NUREG/CSD-6 (1978).

3. "Nuclear Safety Guide," US Atomic Energy Commission report TID-7016, Rev. 1 (1961).

4. *American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,* ANSI/ANS-8.1-1983, American Nuclear Society, LaGrange Park, IL (1983).

5. *American National Standard for Nuclear Criticality Control of Special Actinide Elements*, ANSI/ANS-8.15-1981, American Nuclear Society, LaGrange Park, IL (1981).

6. F. William Walker, Josef R. Parrington, and Frank Feiner, editors, *Nuclides and Isotopes, Fourteenth Edition*, GE Nuclear Energy, San Jose, CA, 12–13 (1989).

7. *Proceedings of the 4th Annual Scientific & Technical Conference on Nuclear Energy and Human Safety, Nizhni Novgorod, Russia, June 28 – July 2, 1993*, NE-93 (1993).

8. H. C. Paxton, "Glossary of Nuclear Criticality Terms," Los Alamos National Laboratory report LA-11627-MS (1989).

9. "Glossary of Terms in Nuclear Science and Technology," prepared by ANS-9, the American Nuclear Society Standards Subcommittee on Nuclear Terminology and Units, American Nuclear Society, LaGrange Park, IL (1986).

10. E. D. Clayton, "Anomalies of Nuclear Criticality," Pacific Northwest Laboratories report PNL-SA-4868 Rev. 5 (1979).

11. H. C. Paxton and N. L. Pruvost, "Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U, 1986 Revision," Los Alamos National Laboratory report LA-10860-MS (1987).

12. L. M. Petrie and N. F. Landers, "KENO V.a, An Improved Monte Carlo Criticality Program With Supergrouping," NUREG/CR-0200, Volume 2, Section F11, ORNL/NUREG/CSD-2/VI/R2 (1984).

13. J. F. Briesmeister, editor, "MCNP – A General Monte Carlo N-Particle Transport Code, Version 4A," Los Alamos National Laboratory report LA-12625-M (1993).

14. "MONK6 – A Monte Carlo Code for Criticality Safety Calculations," ANSWERS Service, AEA Reactor Services, Winfrith Technology Center, Dorchester, UK (1990).

15. S. Glasstone and M. C. Edlund, *The Elements of Nuclear Reactor Theory,* D. Van Nostrand Co., Inc., Princeton, NJ (1952).

16. A. M. Weinberg and E. P. Wigner, *The Physical Theory of Neutron Chain Reactors,* The University of Chicago Press, Chicago, IL (1958).

17. H. S. Isbin, *Introductory Nuclear Reactor Theory,* Reinhold Publishing Corp., New York, NY (1963).

18. G. I. Bell and S. Glasstone, *Nuclear Reactor Theory,* Van Nostrand Reinhold, New York, NY (1970).

19. J. R. Lamarsh, *Introduction to Nuclear Reactor Theory,* Addison-Wesley, Reading, MA (1972).

20. A. F. Henry, *Nuclear-Reactor Analysis,* MIT Press, Cambridge, MA (1975).

21. J. J. Duderstadt and L. J. Hamilton, *Nuclear Reactor Analysis,* Wiley, New York, NY (1976).

22. S. Glasstone and A. Sesonske, *Nuclear Reactor Engineering,* 3rd Edition, Van Nostrand Reinhold, New York, NY (1981).

23. R. A. Knief, *Nuclear Criticality Safety, Theory and Practice,* American Nuclear Society, LaGrange Park, IL (1985).

24. R. A. Knief *Nuclear Engineering, Theory and Technology of Commercial Nuclear Power,* 2nd Edition, Taylor & Francis/Hemisphere, Washington, DC (1992).

25. *American National Standard Administrative Practices for Nuclear Criticality Safety,* ANSI/ANS-8.19-1984, American Nuclear Society, LaGrange Park, IL (1984).

26. *American National Standard Nuclear Criticality Safety Training,* ANSI/ANS-8.20-1991, American Nuclear Society, LaGrange Park, IL (1991).

27. *American National Standard Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement,* ANSI/ANS-8.10-1983, American Nuclear Society, LaGrange Park, IL (1983).

28. *American National Standard Safety Guide for the Performance of Critical Experiments,* ANSI-N405-1975/ANS-1, American Nuclear Society, LaGrange Park, IL (1975).

29. *American National Standard for Safety in Conducting Subcritical Neutron-Multiplication Measurements in Situ,* ANSI/ANS-8.6-1983, American Nuclear Society, LaGrange Park, IL (1983).

30. W. R. Stratton, revised by D. R. Smith, "A Review of Criticality Accidents," Lawrence Livermore National Laboratory report DOE/NCT-04 (1989).

31. *American National Standard Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material,* ANSI/ANS-8.5-1986, American Nuclear Society, LaGrange Park, IL (1986).

32. G. E. Hansen and H. C. Paxton, "Reevaluated Critical Specifications of Some Los Alamos Fast-Neutron Systems," Los Alamos Scientific Laboratory report LA-4208 (1969).

33. J. T. Mihalczo, J. J. Lynn, J. R. Taylor, and G. E. Hansen, "Measurements With an Unreflected Uranium (93.2%) Metal Sphere," *Proceedings of a Topical Meeting on Physics and Methods in Criticality Safety, Nashville, TN, September 19-23, 1993*, American Nuclear Society, LaGrange Park, IL, 26–33 (1993).

34. G. R. Keepin, "Nuclear Fission and Nuclear Safeguards: Common Technologies and Challenges," *Proceedings of the Conference 50 Years with Nuclear Fission, Gaithersburg, MD, April 26-28, 1989*, National Academy of Sciences and National Institute of Standards and Technology, Washington, DC (1989).

35. R. L. Price and N. L. Pruvost, "Inventory Difference Information and Criticality Safety," *Proceedings of an International Topical Meeting on Safety Margins in Criticality Safety, San Francisco, CA, November 26-30, 1989*, American Nuclear Society report ISBN:89448-142-2, 165–167 (1989).

36. R. G. Taylor, "Monitoring for Uranium Accumulations in Diffusion Plant Equipment," Oak Ridge Gaseous Diffusion Plant report K-L-6316 (1973).

37. J. T. Mihalczo, E. D. Blakeman, G. E. Ragan, E. B. Johnson, and Y. Hachiya,"Dynamic Subcriticality Measurements Using the ^{252}Cf-Source-Driven Noise Analysis Method," *Nuclear Science and Engineering,* **104,** 314–338 (1990).

38. E. R. Martin, D. F. Jones, and L. G. Speir, "Passive Segmented Gamma Scan Operation Manual, " Los Alamos Scientific Laboratory report LA-5652-M (1974).

39. R. B. Walton, W. I. Whitted, and R. A. Forster, "Gamma-Ray Assay of Low-Enriched Uranium Waste," *Nuclear Technology,* **24,** 81–92 (1974).

40. C. J. Umbarger and L. R. Cowder, "Portable Radioactivity Monitor for Liquid Effluents, Surface Contaminations, and Bulk Solid Wastes," *Nuclear Instruments and Methods,* **121**, 491–498 (1974).

41. J. T. Mihalczo, "Randomly Pulsed Neutron Measurements for Safeguards Interrogation," Oak Ridge Y-12 Plant report Y-DR-79 (1972).

42. J. E. Foley and L. R. Cowder, "Assay of the Uranium Content of Rover Scrap with the Random Source Interrogation System," Los Alamos Scientific Laboratory report LA-5692-MS (1974).

43. R. Berg, R. Swennen, G. Birkhoff, L. Bondar, J. Ley, and B. Busca, "On the Determination of the Pu-240 in Solid Waste Containers by Spontaneous Fission Neutron Measurements. Application to Reprocessing Plant Waste," EUR 5158e, Joint Nuclear Research Centre, ISPRA Establishment, Italy (1974).

44. T. D. Reilly, E. R. Martin, J. L. Parker, L. G. Speir and R. D. Walton, "A Continuous In-Line Monitor for UF_6 Enrichment," *Nuclear Technology,* **23,** 318–327 (1974).

45. J. L. Forstner, "Nuclear Safety Control in the Chemical Processing Facilities of the Savannah River Plant," *Proceedings of a Symposium on Criticality Control of Fissile Materials, Stockholm, November 1-5, 1965*, International Atomic Energy Agency, Vienna, 627–639 (1965).

46. *American National Standard Criticality Accident Alarm System,* ANSI/ANS-8.3-1986, American Nuclear Society, LaGrange Park, IL (1986).

47. J. M. Juran, F. M. Gryna, Jr., and R. S. Bingham, Jr., editors, *Quality Control Handbook,* Third Edition, McGraw-Hill, New York, NY (1974).

48. *American National Standard Quality Assurance Program Requirements for Nuclear Facilities,* ASME NQA-1-1989 Edition, American Society of Mechanical Engineers, New York, NY (1989).

49. *American National Standard Quality Assurance Requirements for Nuclear Facility Applications,* ASME NQA-2-1989 Edition, American Society of Mechanical Engineers, New York, NY (1989).

50. J. S. Arendt, D. K. Lorenzo, and A. F. Lusby, "Evaluating Process Safety in the Chemical Industry, A Manager's Guide to Quantitative Risk Assessment," JBF Associates, Inc. (1989).

51. J. R. Wilson, "Applications of PRA to Criticality Safety at ICPP," *Proceedings of an International Topical Meeting on Safety Margins in Criticality Safety, San Francisco, CA, November 26-30, 1989*, American Nuclear Society report ISBN:89448-142-2, 186–191 (1989).

52. R. R. Jackson, "Science Applications International Corporation's Experience in Applying Probabilistic Safety Assessment Techniques to Nuclear Criticality Accident Analysis," *Proceedings of an International Topical Meeting on Safety Margins in Criticality Safety, San Francisco, CA, November 26-30, 1989*, American Nuclear Society report ISBN:89448-142-2, 179–182 (1989).

53. Nuclear Regulatory Commission Regulatory Guide 3.33, "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant," US Nuclear Regulatory Commission (1977).

54. W. Thomas and B. Gmal, "In-Depth Analysis of Accidental Criticality in a Reprocessing Plant," *Proceedings of an International Topical Meeting on Safety Margins in Criticality Safety, San Francisco, CA, November 26-30, 1989*, American Nuclear Society report ISBN:89448-142-2, 207–213 (1989).

55. T. P. McLaughlin, "Process Criticality Accident Likelihoods, Consequences and Emergency Planning," *Nuclear Energy,* **31**, No. 2, 143–147 (1992).

56. *Proceedings of the Fifth International Conference on Nuclear Criticality Safety, Albuquerque, NM, September 17-21, 1995* (1995).

57. R. L. Seale, "Consequences of Criticality Accidents," *Nuclear Criticality Safety,* US Atomic Energy Commission report TID-26286, 16–24 (1974).

58. A. D. Callihan and J. T. Thomas, "Accidental Radiation Excursion at the Oak Ridge Y-12 Plant – I," *Health Physics,* **1**, 363–372 (1959).

59. J. D. McLendon, "Accidental Radiation Excursion at the Oak Ridge Y-12 Plant – II," *Health Physics,* **2**, 21–29 (1959).

60. G. S. Hurst, R. H. Richie, and L. C. Emerson, "Accidental Radiation Excursion at the Oak Ridge Y-12 Plant – III," *Health Physics,* **2**, 121–133 (1959).

61. T. L. Shipman, "Acute Radiation Death Resulting from an Accidental Nuclear Critical Excursion," *Journal of Occupational Medicine,* Special Supplement, 147–149 (1961).

62. H. C. Paxton, R. D. Baker, W. J. Maraman, and Roy Reider, "Nuclear-Critical Accident at the Los Alamos Scientific Laboratory on December 30, 1958," Los Alamos Scientific Laboratory report LAMS-2293 (1959).

63. W. C. Roesch *et al.*, "Dosimetry Investigation of the Recuplex Criticality Accident," *Health Physics,* **9**, 757–768 (1963).

64. T. G. Hughes, "Criticality Incident at Windscale," *Nuclear Engineering International,* **17**, No. 189, 95–97 (1972).

65. W. Sweet, "Kyshtym Visit Gives First Look at Soviet Plutonium Production Complex," *Physics Today,* November 1989, 87–89 (1989).

66. L. A. Buldakov, S. N. Demin, V. A. Kostyuchenko, N. A. Koshurnikova, L. Yu. Krestinina, M. M. Saurov, Z. B. Tokarskaya, V. L. Shvedov, and I. A. Ternovskij, "Medical Consequences of the Radiation Accident in the Southern Urals," *Proceedings of an International Symposium on Recovery Operations in the Event of a Nuclear Accident or Radiological Emergency, Vienna, November 6-10, 1989*, IAEA-SM-316/55-2, 419–431 (1989).

67. "Investigation of Incident in Ion Exchange Resin," HRC report 3719, Atlantic Richfield Hanford Company, Richland, WA (1976).

68. B. L. Broadhead and C. M. Hopper, "Updated Tool for Nuclear Criticality Accident Emergency Response," *Transactions of the American Nuclear Society,* **72**, 218–220 (1995).

69. J. T. Thomas, "Criticality of Large Systems of Subcritical U(93) Components," Oak Ridge National Laboratory report ORNL-CDC-1 (1967).

70. P. Lécorché and R. L. Seale, "A Review of the Experiments Performed to Determine the Radiological Consequences of a Criticality Accident," Oak Ridge Y-12 Plant report Y/CDC-12 (1973).

71. *American National Standard for Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors,* ANSI/ANS-8.12-1987, American Nuclear Society, LaGrange Park, IL (1987).

72. H. K. Clark, "Subcritical Limits for Plutonium Systems," *Nuclear Science and Engineering,* **79**, 65–84 (1981).

73. H. K. Clark, "Subcritical Limits for Uranium-235 Systems," *Nuclear Science and Engineering,* **81**, 351–378 (1982).

74. H. K. Clark, "Subcritical Limits for Uranium-233 Systems," *Nuclear Science and Engineering,* **81**, 379–395 (1982).

75. E. D. Clayton, H. K. Clark, Gordon Walker, and R. A. Libby, "Basis for Extending Limits in ANSI Standard for Mixed Oxides to Heterogeneous Systems," *Nuclear Technology,* **75**, 225–229 (1986).

76. J. W. Webster, "Calculated Neutron Multiplication Factors of Uniform Aqueous Solutions of ^{233}U and ^{235}U," Oak Ridge National Laboratory report ORNL-CDC-2 (1967).

77. S. R. Bierman, G. R. Smolen, and T. Matsumoto, "Experimental Criticality Data Comparing Organic and Water Moderation," *Transactions of the American Nuclear Society,* **54**, 195–196 (1987).

78. H. K. Clark, "Effect of Distribution of Fissile Material on Critical Mass," *Nuclear Science and Engineering,* **24**, 133–141 (1966).

79. W. H. Roach and D. R. Smith, "Estimates of Maximum Subcritical Dimensions of Single Fissile Metal Units," Oak Ridge National Laboratory report ORNL-CDC-3 (1967).

80. D. R. Smith and W. U. Geer, "Critical Mass of a Water-Reflected Plutonium Sphere," *Nuclear Applications and Technology,* **7**, No. 5, 405–408 (1969).

81. W. R. Stratton, "Criticality Data and Factors Affecting Criticality of Single Homogeneous Units," Los Alamos Scientific Laboratory report LA-3612 (1964).

82. C. G. Chezem and R. G. Steinke, "Low-Enrichment Uranium-Metal Exponential Experiments," *Nuclear Science and Engineering*, **31**, 549 (1967).

83. M. Darrouzet, J. P. Chaudat, E. A. Fischer, G. Ingram, J. E. Sanders, W. Scholtyssek, "Studies of Unit K_∞ Lattices in Metallic Uranium Assemblies Zebra 8H, Sneak 8, Ermine, and Harmonie UK," *Proceedings of an International Symposium on Physics of Fast Reactors, Tokyo, October 16-19, 1973*, Committee for the International Symposium on Physics of Fast Reactors, A28 (1973).

84. C. E. Newlon, "The Effect of Uranium Density on the Safe ^{235}U Enrichment Criterion," Oak Ridge Gaseous Diffusion Plant report K-1550 (1962).

85. E. D. Clayton, H. K. Clark, D. W. Magnuson, J. H. Chalmers, Gordon Walker, N. Ketzlach, Ryohei Kiyose, C. L. Brown, D. R. Smith, and R. Artigas, "Basis for Subcritical Limits in Proposed Criticality Safety Standard for Mixed Oxides," *Nuclear Technology,* **35**, 97–111 (1977).

86. H. K. Clark, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water," DP-1014, Savannah River Laboratory (1966).

87. H. K. Clark, "Maximum Safe Limits for Slightly Enriched Uranium and Uranium Oxide," *Proceedings of a Symposium on Criticality Control of Fissile Materials, Stockholm, November 1-5, 1965*, International Atomic Energy Agency, Vienna, 35–49 (1965).

88. R. C. Lloyd, "Buckling Measurements of Fuel Elements in a Random Array, Water Moderated," *Nuclear Physics Research Quarterly Report, October-December, 1957,* Hanford Engineering Development Laboratory report HW-54591, 35–36 (1957).

89. R. C. Lloyd, "Buckling Measurements for Fuel Elements in a Random Array," *Nuclear Physics Research Quarterly Report, January-March, 1958,* Hanford Engineering Development Laboratory report HW-55879, 12–13 (1958).

90. J. T. Thomas, "Calculated Criticality of Water Moderated Oxides of Uranium-233, Thorium-232, and Carbon Mixtures," Oak Ridge Y-12 Plant report Y-DR-107 (1973).

91. R. C. Lloyd and E. D. Clayton, "Criticality Safety Data Applicable to Processing Liquid-Metal Fast Breeder Reactor Fuel," *Nuclear Science and Engineering,* **59**, 21–26 (1976).

92. J. K. Fox, L. W. Gilley, and D. Callihan, "Critical Mass Studies, Part IX, Aqueous U^{235} Solutions," Oak Ridge National Laboratory report ORNL-2367 (1958).

93. J. K. Fox and L. W. Gilley, "Critical Parameters for Poisoned Annular Cylinders Containing Aqueous Solutions of U^{235}," *Neutron Physics Division Annual Progress Report for Period Ending September 1, 1958,* Oak Ridge National Laboratory report ORNL-2609, 31–33 (1958).

94. C. Clouet d'Orval, E. Deilgat, M. Houelle, and P. Lécorché, "Experimental Research in France on Criticality Problems" (in French), *Proceedings of a Symposium on Criticality Control of Fissile Materials, Stockholm, November 1-5, 1965*, International Atomic Energy Agency, Vienna, 193–213 (1965).

95. J. E. Tanner and H. M. Forehand, "Critical Experiments for Large Scale Enriched Uranium Solution Handling," *Proceedings of a Topical Meeting on Criticality Safety in the Storage of Fissile Material, Jackson, WY, September 8-11, 1985*, American Nuclear Society report ISBN:89448-119-3, 65–79 (1985).

96. R. C. Lloyd and T. Koyama, "Criticality Experiments with Mixed Plutonium-Uranium Nitrate Solution at Plutonium Fractions of 0.2, 0.5, and 1.0 in Annular Cylindrical Geometry," *Transactions of the American Nuclear Society,* **56**, 318–319 (1988).

97. *American National Standard Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Material*, ANSI/ANS-8.9-1987, American Nuclear Society, LaGrange Park, IL (1987).

98. D. Dickinson and C. L. Schuske, "An Empirical Model for Safe Pipe Intersections Containing Fissile Solution," *Nuclear Technology,* **10**, 179–187 (1971).

99. E. B. Johnson, "The Nuclear Criticality of Intersecting Cylinders of Aqueous Uranyl Fluoride Solutions," Oak Ridge Y-12 Plant report Y-DR-129 (1974).

100. N. F. Cross, G. E. Whitesides, and R. J. Hinton, "Monte Carlo Analysis of Experimentally Critical Pipe Intersections," *Transactions of the American Nuclear Society,* **17**, 268 (1973).

101. L. B. Engle, G. E. Hansen, and H. C. Paxton, "Reactivity Contributions of Various Materials in Topsy, Godiva, and Jezebel," *Nuclear Science and Engineering,* **8**, 543–569 (1960).

102. J. D. Orndoff, H. C. Paxton, and G. E. Hansen, "Critical Masses of Oralloy at Reduced Concentrations and Densities," Los Alamos Scientific Laboratory report LA-1251 (1951).

103. R. C. Lloyd, E. D. Clayton, and L. E. Hansen, "Criticality of Plutonium Nitrate Solutions Containing Soluble Gadolinium," *Nuclear Science and Engineering,* **48**, 300–304 (1972).

104. R. C. Lloyd and E. D. Clayton, "Effect of Boron and Gadolinium on the Criticality of Plutonium-Uranium Systems," *Transactions of the American Nuclear Society,* **23**, 234–237 (1976).

105. E. D. Clayton and C. L. Brown, "Criticality and Nuclear Safety of Slightly Enriched Uranium," *Chemical Engineering Progress Symposium Series,* **61**, No. 60, 33–43 (1965).

106. V. I. Neeley, J. A. Berberet, and R. H. Masterson, "k_∞ of Three Weight Percent ^{235}U Enriched UO_3 and $UO_2(NO_3)_2$ Hydrogeneous Systems," HW-66882, Hanford Engineering Development Laboratory (1961).

107. R. Gwin and D. W. Magnuson, "The Measurement of Eta and Other Nuclear Properties of ^{233}U and ^{235}U in Critical Aqueous Solutions," *Nuclear Science and Engineering,* **12**, 364–380 (1962).

108. W. E. Converse, R. C. Lloyd, E. D. Clayton, and W. A. Yuill, "Critical Experiments Using High-Enriched Uranyl Nitrate with Cadmium Absorber," *Transactions of the American Nuclear Society,* **32**, 328–330 (1979).

109. J. T. Thomas, "Reflectors, Infinite Cylinders, Intersecting Cylinders, and Nuclear Criticality," *Nuclear Science and Engineering,* **67**, 279–295 (1978).

110. R. C. Lloyd, C. R. Richey, E. D. Clayton, and D. R. Skeen, "Criticality Studies with Plutonium Solutions," *Nuclear Science and Engineering,* **25**, 165–173 (1966).

111. G. R. Handley, R. C. Robinson, and J. C. Cline, "Effects of Concrete Composition in Nuclear Criticality Safety Calculations," *Transactions of the American Nuclear Society,* **61**, 182–184 (1990).

112. H. F. Henry, J. R. Knight, and C. E. Newlon, "General Application of a Theory of Neutron Interaction," Oak Ridge Gaseous Diffusion Plant report K-1309 (1956).

113. H. C. Paxton, "History of Density-Analog Storage Criteria," *Proceedings of a Topical Meeting on Criticality Safety in the Storage of Fissile Material, Jackson, WY, September 8-11, 1985*, American Nuclear Society report ISBN:89448-119-3, 346–352 (1985).

114. J. T. Thomas, "Remarks on Surface Density and Density Analog Representation of Array Criticality," *Transactions of the American Nuclear Society,* **22**, 299–300 (1975).

115. J. T. Thomas, "Surface Density and Density Analogue Models for Criticality in Arrays of Fissile Materials," *Nuclear Science and Engineering,* **62**, 424–437 (1977).

116. M. C. Evans and J. R. Bowe, "Applications of the Limiting Surface Density Method to Transport and Storage of Special Nuclear Materials," *Proceedings of a Topical Meeting on Criticality Safety in the Storage of Fissile Material, Jackson, WY, September 8-11, 1985*, American Nuclear Society report ISBN:89448-119-3, 307–321 (1985).

117. D. R. Smith, "Criteria and Evaluation for the Storage of Fissile Materials in a Large and Varied Reactor Research Development Programme," *Proceedings of a Symposium on Criticality Control of Fissile Materials, Stockholm, November 1-5, 1965*, International Atomic Energy Agency, Vienna, 667–677 (1965).

118. H. C. Paxton, "Density-Analog Techniques," *Proceedings of the Livermore Array Symposium, Livermore, CA, September 23-25, 1968*, Lawrence Radiation Laboratory report LRL-CONF-680909, 6–11 (1968).

119. *American National Standard Guide for Nuclear Criticality Safety in the Storage of Fissile Materials,* ANSI-N16.5-1975/ANS-8.7, American Nuclear Society, LaGrange Park, IL (1975).

120. H. K. Clark, "A Simple Practical Method for Calculating Interaction," *Proceedings of a Symposium on Criticality Control of Fissile Materials, Stockholm, November 1-5, 1965*, International Atomic Energy Agency, Vienna, 87–102 (1965).

121. H. K. Clark, "Interaction of Fissionable Units," *Nuclear Science and Engineering,* **15**, 20–28 (1963).

122. H. K. Clark, "Application of a Simple, Practical Method for Computing Interaction to Arrays Found Experimentally to be Critical," *Nuclear Science and Engineering,* **20**, 307–313 (1964).

123. A. F. Thomas and R. A. Scriven, "Neutron Interaction in Fissile Assemblies," *Progress in Nuclear Energy, Series IV, Vol. 3 – Technology, Engineering and Safety,* Pergamon Press, London, 253–291 (1960).

124. J. T. Thomas, "Critical Three-Dimensional Arrays of U(93.2)-Metal Cylinders," *Nuclear Science and Engineering,* **52**, 350–359 (1973).

125. O. C. Kolar, H. F. Finn, and N. L. Pruvost, "Livermore Plutonium Array Program: Experiments and Calculations," *Nuclear Technology,* **29**, 57–72 (1976).

126. J. T. Thomas, "Experimental and Calculated System Criticality," *Proceedings of a Symposium on Criticality Control of Fissile Materials, Stockholm, November 1-5, 1965*, International Atomic Energy Agency, Vienna, 149–175 (1965).

127. J. T. Thomas, "The Criticality of Cubic Arrays of Fissile Materials," Oak Ridge Y-12 Plant report Y-CDC-10 (1971).

128. D. Yearwood, E. D. Clayton, and B. L. Koponen, "Anomalous Effects of Moderation in Transportation and Storage Arrays – Revisited," *Proceedings of a Topical Meeting on Physics and Methods in Criticality Safety, Nashville, TN, September 19-23, 1993*, American Nuclear Society, LaGrange Park, IL, 99–101 (1993).

129. J. T. Thomas, "Experimental Measurements with Arrays of U(97.7)F_6 Neutron-Coupled Through Concrete," *Transactions of the American Nuclear Society,* **19**, 199–200 (1974).

130. G. D. Ellis and G. R. Handley, "Enriched Uranium Storage in Steel Tubes Embedded in Concrete," *Proceedings of a Topical Meeting on Criticality Safety in the Storage of Fissile Material, Jackson, WY, September 8-11, 1985*, American Nuclear Society report ISBN:89448-119-3, 186–198 (1985).

131. J. T. Thomas, "A Criticality Indicator System for Storage of Fissile Materials," Oak Ridge National Laboratory report UCCND-CSD-INF-48 (1975).

132. J. T. Thomas, "Criticality of ^{233}U Aqueous Nitrate Solutions in Reflected and Unreflected Arrays," *Transactions of the American Nuclear Society,* **10**, 538–539 (1967).

133. R. C. Lloyd, E. D. Clayton, and J. H. Chalmers, "Criticality of Arrays of ^{233}U Solution," *Nuclear Applications,* **4**, 136–141 (1968).

134. R. E. Rothe, "A Survey of Fissile Solution Storage Methods: Champion the Poisoned Tube Tank," *Proceedings of a Topical Meeting on Criticality Safety in the Storage of Fissile Material, Jackson, WY, September 8-11, 1985*, American Nuclear Society report ISBN:89448-119-3, 42–64 (1985).

135. J. T. Thomas, J. K. Fox, and E. B. Johnson, "Critical Mass Studies, Part XIII. Borosilicate Glass Raschig Rings in Aqueous Uranyl Nitrate Solutions," Oak Ridge National Laboratory report ORNL-TM-499 (1963).

136. J. T. Thomas, "Critical Experiments with Aqueous Solutions of $^{233}UO_2(NO_3)_2$," *Neutron Physics Division Annual Progress Report for Period Ending May 31, 1968,* Oak Ridge National Laboratory report ORNL-4280, 53–55 (1968).

137. R. C. Lloyd, S. R. Bierman, and E. D. Clayton, "Criticality of Plutonium Nitrate Solutions Containing Borated Raschig Rings," *Nuclear Science and Engineering,* **50**, 127–134 (1973).

138. R. C. Lloyd and E. D. Clayton, "Criticality of Pu-U Nitrate Solution Containing Glass Raschig Rings," *Transactions of the American Nuclear Society,* **21**, 236–237 (1975).

139. J. K. Fox and L. W. Gilley, "Critical Parameters of Aqueous Solutions of U^{235}," *Neutron Physics Division Annual Progress Report for Period Ending September 1, 1957,* Oak Ridge National Laboratory report ORNL-2389, 71–83 (1959).

140. D. Callihan, "Experiments for Criticality Control," *Proceedings of a Symposium on Criticality Control in Chemical and Metallurgical Plants, Karlsruhe, Germany,* 589–614 (1961).

141. O. D. Thompson and D. D. Yearwood, "Plutonium Solution Storage in Arrays of Six-inch Pipes," *Proceedings of a Topical Meeting on Criticality Safety in the Storage of Fissile Material, Jackson, WY, September 8-11, 1985*, American Nuclear Society report ISBN:89448-119-3, 92–98 (1985).

142. Safety Series No. 6, *Regulations for the Safe Transport of Radioactive Materials, 1985 Edition,* International Atomic Energy Agency, Vienna (1985).

143. Safety Series No. 37, *Advisory Material for the Application of the IAEA Transport Regulations, Second Edition*, International Atomic Energy Agency, Vienna (1982).

144. 49 CFR 173.4, *Regulations, Packaging and Transportation, Radioactive Materials*, Department of Transportation (1991).

145. 10 CFR 71, *Packaging and Transportation of Radioactive Material*, Nuclear Regulatory Commission (1991).

146. DOE Order 5480.3 (1985), *Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes* (1985).

147. W. C. Jordan, N. F. Landers, and L. M. Petrie, "Validation of KENO V.a, Comparison with Critical Experiments," Oak Ridge National Laboratory report ORNL/CSD/TM-238 (1986).

148. J. P. Nichols, C. L. Schuske, and D. W. Magnuson, "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material," Oak Ridge Y-12 Plant report Y-CDC-8 (1971).

149. E. O. Nurmi, "Use of Borosilicate-Glass Raschig Rings in a Hostile Environment," *Transactions of the American Nuclear Society*, **19**, 181 (1974).

150. R. F. Christy and J. A. Wheeler, "Chain Reaction of Pure Fissionable Materials in Solution," US Government report CP-400 (1943).

Los Alamos
NATIONAL LABORATORY

Los Alamos, New Mexico 87545