Criticality Safety Principles for Health Physicists

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Introduction

The purpose of this session is to discuss the health physics aspects of nuclear criticality accidents. This unique class of accidents is of concern whenever an operation involves working with significant quantities of fissile materials. In the normal context, 'significant quantities' is taken to mean quantities in excess of 700 grams of uranium-235 or 450 grams of plutonium or uranium-233 (these are not the only fissionable elements, but they are the ones normally encountered). These values are typically the lower limits of material necessary to establish a critical system; under very special situations it may be possible to achieve such a state with less, but those situations are very rare in normal practices.

This session will begin with a basic discussion of the concept and principles of critical systems to understand what they are and how they work. This is only an introduction, as the study of criticality safety is a highly technical discipline of its own right, and would go beyond the needs of this audience. These concepts are important to understand, however, as they will provide the basis for a discussion of the radiological hazards resulting from these accidents.

Once the foundation is laid, we will discuss the radiological characteristics of a criticality accident, and potential consequences to victims. The past experiences with these accidents will be reviewed, with a focus on a few events as case studies. Concepts for planning for and responding to these accidents will be considered, along with identification of applicable standards.

In each step of this discussion we will include considerations of the tools available for evaluating and assessing the technical aspects of these accidents.

Basic principles of critical systems

What is a critical system, what causes a system to go critical, and how does one behave? In this section we will discuss the basic physics of a critical system.

Definition of a critical System

Quite simply and accurately, ***a critical system is an assembly of fissionable and potentially other material that produces a release of energy from a self-sustaining or divergent neutron chain reaction.***[A] Consequently, ***a criticality accident is an inadvertent assembly of a critical system.***

Note that this definition can be applied equally well to operations with reactors and to operations outside of reactors. Therefore, some accidents with reactors are classified as criticality accidents. Two notable examples would be the SL-1 reactor accident of 1961 and the Chernobyl accident of 1986. The Three Mile Island accident would not be one.

Determination of critical state

How is a critical state determined? In practice, this can be a quite complicated analysis to undertake, and beyond the scope of this course. However, in nuclear engineering the status of a potentially critical system is monitored by a single value known as ***k-effective***, or simply **k**. In technical terms, **k** is the eigenvalue of an equation balancing the rates of neutron production (from fissions and sources) and losses (from leakage and non-fission absorption) in a fissionable assembly.[B]

Before discussing the critical state of a fissionable system, we must digress for a moment to discuss the time-dependency of the fission process itself. For a large percentage of fission events, the neutrons are emitted essentially instantaneously with the fission of the nucleus, or at least within 10^{-9} seconds or less after the absorption (referred to as ***prompt neutrons***). However, a small fraction of the neutrons, generally on the order of about 0.65% for U-235 and only about 0.2% for Pu-239, are from unstable neutron-rich fission products that undergo neutron decay over longer periods of time, generally fractions of seconds to almost a minute after the fission takes place. These fission products are called ***delayed neutron precursors***, and are the whole basis on which a fission chain reaction can be controlled in a reactor. (Control systems could not respond to changes in neutron populations occurring in nanosecond timescales.) Typically these precursors are collected into six groups by decay rate, and each group is represented by two values, β_i, the fractional yield of the group, and λ_i, the decay rate in inverse seconds (i is the group number). The sum of the fractional yields is β. (Sometimes you will hear reference to β-effective, but we don't need to go into that here.)

For our purposes, all that is necessary to know is that **k** can be used to determine the critical state of the system in the following way:

k < 1.0	For a value of **k** less than 1.0, the neutron loss rate exceed the production rate, and the system cannot self-sustain the chain reaction. This state is referred to as being ***subcritical***.
k = 1.0	For a value of **k** equal to 1.0, the neutron production and loss rates are equal, and the system is self-sustaining. This state is referred to as being ***critical.***
1.0 < k < 1.0 +β	For a value of **k** greater than 1.0, but less than 1.0 + β, the neutron production rate exceeds the neutron loss rate, and the system will increase in power. This state is referred to as being ***supercritical.*** However, in this region these increases are gradual, and provide the degree of controllability necessary to operate reactors.
k ≥ 1.0 + β	When **k** is equal to 1.0 + β, the system is ***prompt critical***, that is, it is critical on the prompt neutrons alone. When **k** exceeds this threshold, then the system is considered to be ***super-prompt critical***, and the

power level will increase exponentially. At this point, active control systems are normally not responsive enough to allow control of the system.

Later we will discuss how systems in the latter two states behave when we talk about kinetics. For now, it is sufficient to note that we want to avoid situations where k is equal to or greater than 1.0.

As discussed, k is derived from equations that describe the neutron inventory in the system. It can normally be considered as the ratio of the number neutrons produced to the number lost from the system. While the calculation of k is normally quite complicated, the basic principle can be demonstrated by a simple product of six values, historically known as the ***six-factor formula*** (rather catchy title, isn't it?). The six-factor formula is:

$$k = \varepsilon p \eta f \left(1 - P_{fl}\right)\left(1 - P_{tl}\right). \tag{1}$$

This can also be written in another form (although we lose the 'six factors') as:

$$k = \upsilon \Sigma_f \Big/ \left(\Sigma_a + DB^2\right). \tag{2}$$

The terms in these equations are defined as:

ε = fast fission factor;
p = resonance escape probability;
η = number of neutrons produced per neutron absorbed in fuel;
f = thermal utilization = $\Sigma_a^f \Big/ \left(\Sigma_a^f + \Sigma_a^{nf}\right)$;
P_{fl} = fast neutron leakage probability;
P_{tl} = thermal neutron leakage probability;
υ = number of neutrons per fission event;
Σ = macroscopic cross-section (absorption, fission);
D = Diffusion coefficient; and
B^2 = geometric buckling.

So far, all we have is a couple equations with a bunch of Greek symbols with nonsensical descriptions. While these equations may look simple, it is the determination of the various terms that makes the process quite complicated. These terms are interrelated, and they are dependent on the neutron energy spectrum and the fission distribution within the system. The calculation of these terms usually requires an iterative numeric solution. However, for the purposes of this effort, we don't need to go into a detailed explanation and calculation of these terms. It is the physical characteristics of a critical system that are represented by these terms that we want to discuss here.

Physical properties influencing the critical state

The terms identified in the last section are mathematical representations of the various physical characteristics that influence the critical state of a fissionable system. Table 1 considers those properties, and which of the terms they influence:

Table 1: Physical Influences on a System's Critical State

Physical Property	***Influenced Factor***
material mass (fissionable and other)	Σ's, ε, f, η
fissionable material enrichment	p, f, η
density or concentration	Σ's, f, p
container volume	B^2
composition of non-fissionable materials (absorbers and moderators)	Σ_a, D, p, f
container design or surrounding conditions (presence of neutron reflectors)	P_{fl}, P_{tl}
system geometry	B^2
environmental conditions (temperature, primarily. Others ??)	Σ's, p, D

As can be seen from this table, there are at least 8 physical properties or characteristics that combine to determine the critical state of a fissionable system. A reactor is designed to optimize these properties to achieve a controlled state of criticality, but here we try to do just the opposite.

One other point to be made about the critical state of the system is that the equations presented above are *not* dependent on the amplitude of the neutron or fission flux in the system. This is important, because it implies that a critical condition can be achieved at any power level, as long as the various parameters are balanced against each other properly. For example, a reactor can be brought to critical at a very low power level under one set of conditions. Then the positions of control rods (absorbers) and the coolant temperature (moderator density and fuel temperature) are balanced against each other to raise and lower the power level as necessary. Note that this also implies that ***a system can achieve a critical or supercritical mass with no neutrons present, and then suffer a sudden increase in power when a stray neutron starts the chain reaction.*** This is an important concept for criticality safety, as it means that ***a system can undergo changes in its critical state, and achieve a super-prompt critical condition, completely undetected by personnel or normal radiation monitoring equipment.***

We have not yet discussed the various forms that a critical system can take. When one normally considers a fissionable system, one envisions a single unit, such as a single tank or a mass of solid material. However, arrays are another form of fissionable system that must also be considered. One can take a single unit, and design it to be subcritical. However, it is possible for conditions to be such that a collection of these single units, when put into a particular array, to go critical as a single unit. Therefore, activities such as the stacking of drums and arrays of solution containers must be analyzed in as much or more detail than the individual units.

Another form that must be considered is a reflected system. Reflectors are components exterior to the fissionable system that can scatter neutrons back into the system. For this to be an effective contribution, these components would generally be very close to the system. Any scattering medium can be a reflector, such as walls, shields, floors, ***or even human bodies.*** Even the thickness of a tank wall should be considered. For external reflectors, the absorption properties should be ignored, for they cannot induce the system to leak any more neutrons than normal. ***[From this, note that one cannot use externally positioned poisons to reduce the potential for a criticality event.]*** Reflectors are important because they ***reduce the quantity and geometrical size of a system necessary to go critical compared to an unreflected system.*** For the critical state of the system, it is vitally important that the criticality safety engineer consider possible reflectors in their determinations. For the health physicist, reflectors will also affect the characteristics of the radiation emitted by the system, which will be discussed further below.

The art and science of Criticality Safety is the discipline that evaluates the potential contributions of each of these properties, and determines how to control them to limit the potential for an inadvertent achievement of a critical state. Normally, criticality safety programs are based on a ***double contingency principle*** that recommends that designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. They also attempt to the extent possible to control two different physical properties, and to minimize reliance on administrative practices as a control mechanism. As we will see later when we discuss past accidents, the focus of the investigation is primarily to determine which of these characteristics changed to cause the event, and why.

Kinetics of supercritical systems

Nuclear kinetics is the study of the time-dependent behavior of a fissionable system. The actual sets of equations used to describe fissionable systems are both time- and space-dependent, and become very complex when describing a real system. Therefore, the process of determining the critical state of such as system normally uses the eigenvalue k to replace the time-dependency as a simplifying assumption. Consequently, one can only determine the critical state as to the three general conditions, subcritical, critical, and supercritical. Therefore, the opposite simplification is usually done for evaluating the time-dependency, by using the geometric buckling term B^2 as representative of the space-dependency, and then homogenizing the cross-sections to create a 'point-reactor' model. This is the origin of the ***point-reactor kinetics model***.[C]

To describe the point-reactor model, the following parameters are defined:

$$\ell_\infty \equiv \frac{1}{\nu\Sigma_a} \qquad = \textit{infinite medium neutron lifetime};$$

$$L \equiv \sqrt{\frac{D}{\Sigma_a}} \qquad = \textit{diffusion length};$$

$$k_\infty \equiv \nu\Sigma_f / \Sigma_a \qquad = \textit{infinite medium reproduction factor};$$

$k \equiv \frac{k_\infty}{1 + L^2 B^2}$ = *effective reproduction factor*;

$\ell_0 \equiv \frac{\ell_\infty}{1 + L^2 B^2}$ = *neutron lifetime*;

$\ell = \ell_0 / k$ = *neutron generation time;* and

$\rho = \frac{k - 1}{k}$ = *reactivity.*

Note that B^2 is the geometric buckling term from the eigenvalue solution of the diffusion equation.

Given these definitions, neutron flux as a function of time is:

$$\frac{dn(t)}{dt} = \frac{\rho - \beta}{\ell} n(t) + \sum_{i=1}^{g} \lambda_i C_i(t), \tag{3}$$

where β is the delayed neutron fraction, g is the number of delay groups, and $C_i(t)$ is the concentration of delayed neutron precursors for the ith, such that:

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\ell} n(t) - \lambda_i C_i(t). \tag{4}$$

One advantage of these equations is that they are insensitive to units. Therefore, the neutron density term, $n(t)$, can be replaced with a power term with no loss of meaning, as long as $C_i(t)$ is in the same units. Therefore, we can use these same equations for describing the system's power level as a function of time, which is very convenient.

The system reactivity, ρ, is the key to understanding this process. From its definition, one can see that it is a function of k, from the neutron balance equations above. A positive reactivity change would imply an increase in power level, and a decrease would be a reduction. A reactivity of zero would be a steady-state condition. However, this is not a constant value, and indeed is a function of time, even though it is not normally shown that way. The reason for this is that several of the parameters in the neutron balance equations, most notably the cross-sections, are functions of temperature, so consequently k will vary with temperature. As the system generates heat from the fission process, these parameters change. This temperature-induced change is referred to as feedback, and is accounted for by defining a ***temperature feedback coefficient,*** α, such that

$$\rho(t) = \rho_0 + \alpha(T(t) - T_o) \tag{5}$$

where $T(t)$ is the temperature at time t, and T_o is the initial temperature of the system.

The temperature can be calculated from Newton's Law of Cooling,

$$\frac{dT}{dt} = Kn(t) - \gamma\left(T(t) - T_c\right), \qquad (6)$$

where K is the reciprocal of the heat capacity and γ is a heat transfer coefficient. For the purposes of the rapid initial heating of the system, adiabatic conditions are assumed, and the heat transfer coefficient can be set to zero. However, we will later see that it does come back into play if the system does not self-disassemble after the initial event.

The temperature feedback coefficient is actually a sum of various components, and can be either positive or negative, depending on the system's design, and which parameters dominate the change. Obviously, a positive feedback term would cause power to continue to increase, and a negative term would cause power to decrease as temperature increased. However, for most simple systems like those normally encountered for criticality safety concerns, the feedback is typically negative. In these cases, this feedback is normally due to material densities decreasing from thermal expansion, causing macroscopic cross-sections to go down.

The formulas above allow us to completely predict the power history, energy production, fission yield, and final temperature of the system. The main problem with all of this is that many of the parameters, especially the temperature feedback coefficient, are difficult to predict for an inadvertent system, since there are many variable conditions that could be postulated as leading to the event. As a result, the kinetics equations are typically used to qualitatively understand the likely behavior, and to back-calculate a system's behavior following an event.

Now that we have a basic understanding of the equations, we can discuss the qualitative aspects of what they tell us.

The supercritical regime can be broken up into two parts, by drawing a line at the point where $\rho = \beta$. This point is referred to as ***prompt critical***, since the system is critical on prompt neutrons alone. Systems with reactivities above this point are referred to as ***super-prompt critical.***

This line also represents a significant point with respect to the kinetics of the system, as can be seen from equation 3. Below the line, the delayed neutron groups dominate the power changes, and time response is rather slow, with positive periods[1] ranging from many seconds down to tens of milliseconds. There are no "power pulses" in this range.

It is possible to have a criticality accident occur in this range, but it is rare to see one in practice. The reason is that in the absence of a neutron source, the system will not actually start up until a stray neutron comes along to start the chain reaction. In this time period, whatever is happening

[1] The system's period is the amount of time it takes for the power to change by a factor of e^1. Negative periods indicate decreasing power, positive are increasing power.

to cause the reactivity increase will continue. Also, even if the system started operating in this range, the temperature increase will be very slow, so there will be essentially no feedback to negate the reactivity addition. Therefore, the addition will continue until the system exceeds super-prompt critical. It should be noted that plutonium systems will have a higher neutron background than uranium systems due to spontaneous fissions and (α, n) reactions. Consequently, the kinetics of plutonium systems can be somewhat different than uranium systems, at least for the initial transient.

It is when the system's reactivity exceeds prompt critical that things really get interesting. At this point the period of the system becomes fast enough to 'outrun' the delayed neutrons, and they no longer contribute significantly to the pulse dynamics. From equation 3 it can be seen that if one ignores the delayed neutron groups, the summation term drops out and the equation reduces to a simple exponential function. Therefore, the power increases exponentially with a time constant determined by the neutron generation time in the system. This time constant is highly dependent on the composition of the system, but generally ranges from 10^{-5} to 10^{-8} seconds. Obviously, the power will increase very rapidly.

As the system power increases, it deposits a significant amount of energy into the system faster than any cooling capability can respond to. Therefore, the system rapidly expands, and the temperature feedback ultimately causes the power to peak and begin to fall. This is the power pulse that criticality accidents are famous for. The integral of the power is the total energy deposited, and this can be significant, since peak powers could in some cases exceed 100,000 MW before turning over as feedback comes in. Note that after the pulse is over, there is still a problem: now the delayed neutrons that we previously ignored are coming into play, and the system is still supercritical, although less than prompt critical. Therefore, there is still significant power generated in the 'tail' region, unless in some manner the pulse physically disrupts the system. As an example of what the initial pulse and power tail from a criticality accident may look like, figure 1 displays the power levels and energy release for the SPR III fast burst reactor as a function of time. This reactor is an unmoderated, highly enriched metal system designed for super-prompt critical pulse operations. The fuel temperature increase for a pulse such as this would be about 300^{o}C.

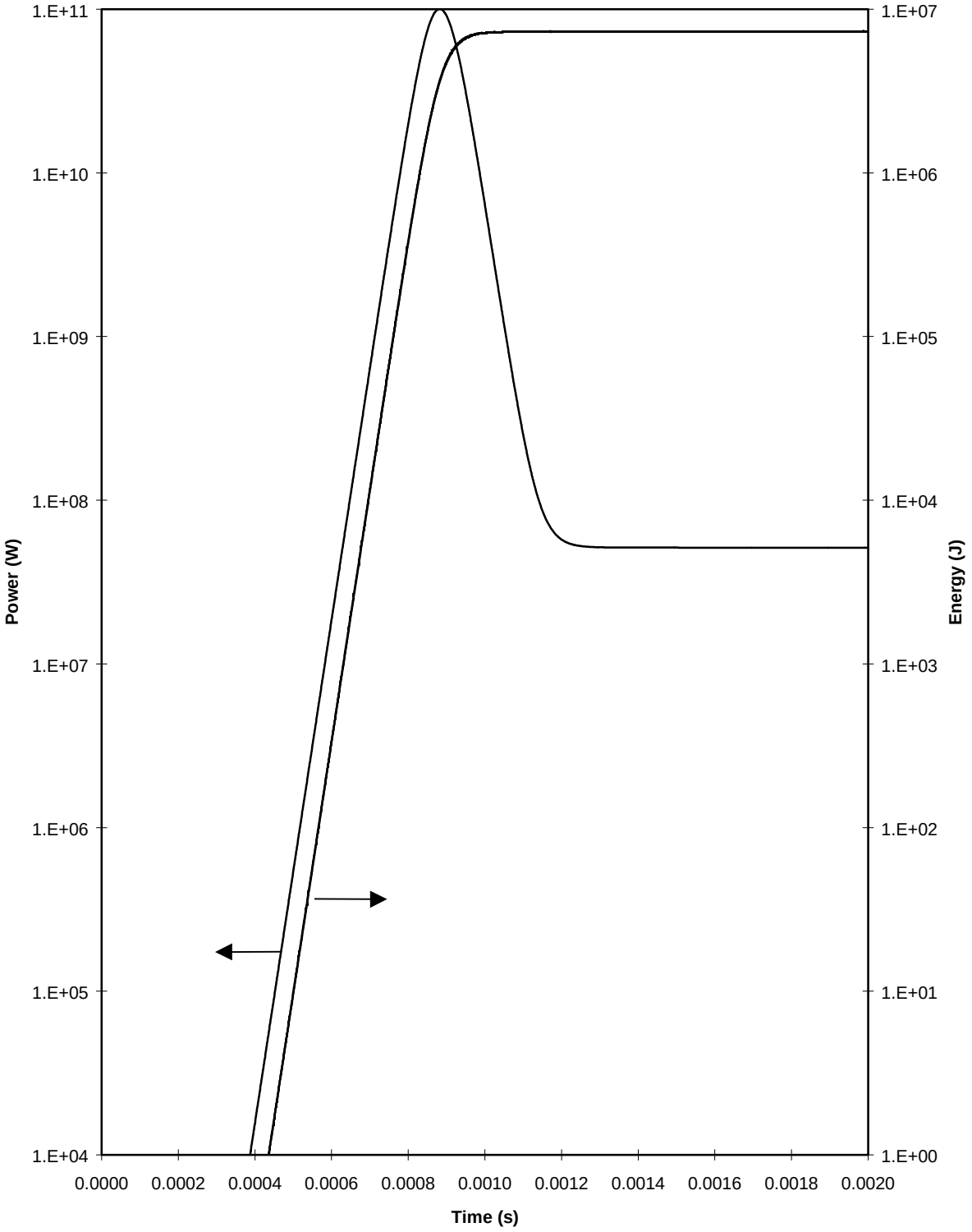

Figure 1: A typical super-prompt critical pulse on the SPR III fast burst reactor, inserted reactivity is 1.10 x Beta. The resulting fuel temperature increase would be about 300 C.

The potential for continued operation is a significant point to remember. In some cases with solutions the energy deposited in the pulse may be sufficient to cause a steam bubble to be generated, splashing out the solution. If the container is open, enough material may be ejected from the top to bring the system subcritical, shutting it down. In other cases, the geometry may change having the same effect. (For example, some previous accidents were caused by transient geometrical conditions, such as starting stirrers that formed a vortex. When the vortex collapsed, so did the geometry that supported the criticality condition.) Normally with solutions, radiolytic gas bubbles will form, causing volumetric changes that tend to shut the system down. As the bubbles migrate from the system, it may return to its initial supercritical geometry, thus initiating another pulse.

It is feasible in a metal system for fuel to melt and slump as a shutdown mechanism, but usually heat transfer to surrounding media will preclude this from happening. However, in lieu of these effects, the system will continue to generate fissions. As heat is slowly transferred out of the system, it could go super-prompt critical again, and generate another pulse.

For either type of system, these oscillatory cycles may well continue, resulting in a gradual increase in power until it stabilizes at some power level commensurate with heat transfer from the system. Therefore, such a system could continue indefinitely, or until enough solution has boiled away to take it subcritical. This effect was clearly observed in both the recent Japanese accident and in the second Russian criticality accident of 1997. In both cases, direct action was necessary to physically disrupt the system before the chain reaction could be terminated.

Obviously, the fission yield of a criticality accident is highly dependent on the configuration of the system and the circumstances of the situation. However, it is important to note that one can establish a range for the fission yield based on both past experiences and first principles. While I will not go into details, ***the normal range for the yield from a criticality accident is between 10^{15} and 10^{20} fissions, with 10^{17} fissions as a nominal mean value.***

Tools for analyzing the critical state of fissionable systems

There are some simple techniques for estimating the critical state of a system, but this would not be recommended for anything beyond an exercise. Remember that the difference between being just subcritical and super-prompt critical is a positive change of only 0.7% for a uranium system, and even less when plutonium is involved. Therefore, accuracy is vitally important.

There are several computer codes that can do criticality calculations. The most popular are the Monte Carlo-based codes like MCNP from Los Alamos or KENO from ORNL. However, all of these codes are quite complicated to use with steep learning curves, so I would suggest getting a criticality safety engineer involved if you have a real situation to evaluate. [In this discussion I will use results from MCNP to demonstrate various aspects of criticality events. Remember that each situation is quite unique, and the results shown here are for informational purposes only, and should not be accepted as general conditions.]

As an alternative to computer codes, there are a variety of references that can be used to determine criticality limits for a wide range of materials and conditions. The most widely known are the ANSI/ANS 8.xx series. This series discusses various aspects of criticality safety, including design of criticality safe containers, criticality alarm systems, critical masses for various geometries and forms, and operation and management of criticality safety programs.

To some extent the same comments apply to the neutron kinetics analysis, although the equations are much more straightforward and easy to use. The problem here is that some of the parameters such as the temperature feedback coefficient are difficult to predict, and the models are very sensitive to that term. However, there is not as great of a need to perform kinetics calculations in practice. Generally, most criticality analyses are conducted by normalizing to a value of total fissions for the excursion. This is often broken into the total for the initial pulse, and the total for continued operations. Therefore, it is only necessary to consider the detailed kinetics when conducting a post-mortem of an actual event.

Radiation hazards from critical systems

Now that we have some understanding as to the physics and dynamics of a criticality event, we can discuss the personnel hazards involved.

One point to make at the beginning, and one that we will keep returning to, comes from the discussions of the last section on kinetics. ***Criticality accidents are characterized by a very short, very intense pulse of radiation, which may be following by additional pulses leading to a continuous steady-state power level.*** This is a key concept to remember. ***For personnel in the vicinity of such an occurrence, there is no way to avoid the initial exposure from the pulse***. The initial pulse is generally over in much less than a second, and cannot be anticipated. All they can do is to evacuate (or be evacuated from) the area as quickly as possible to avoid further exposure from the system. Entry into the area after the event should be done very cautiously, if at all, until the status of the system can be determined.

Under normal conditions, the primary radiation hazards from a criticality accident are the direct emissions of neutrons and photons from the system. These emissions can be extremely intense during the excursion, and can extend to large distances from the system, depending on the amount of intervening shielding. Following the excursion there will continue to be gamma radiation from the decaying fission products and activated materials in the surrounding area, but typically at much lower intensities. Depending on the situation, there is a potential for significant localized contamination, and some possibility of environmental releases of radioactive materials.

This section will discuss the characteristics of each of these hazards, and consider the situation specific influences on them.

Before going further, it should be noted that to demonstrate the various types of critical systems and their characteristics, we would consider six hypothetical systems. These systems may not represent real situations, but they do illustrate the principles discussed in this report. All systems are designed to be super-prompt critical. These systems are:

1. A sphere of U_3O_8, enriched to 97% ^{235}U, and packed to about 50% theoretical density (8.3 gm/cm^3). This sphere is sitting in the corner of a room with walls of 10 cm thick concrete. (This will be referred to as "3-sided reflection".)

2. A sphere of uranium metal, enriched to 97% ^{235}U, at a density of 17 gm/cm^3. This sphere is sitting in the corner of a room with walls of 10 cm thick concrete.

3. A sphere of U_3O_8, enriched to 97% ^{235}U, and packed to about 50% theoretical density (8.3 gm/cm^3). This sphere is sitting in the corner of a room with walls of 10 cm thick steel.

4. An unreflected sphere of uranium metal, enriched to 97% ^{235}U, at a density of 17 gm/cm^3. (Imagine a ball floating in mid-air.)

5. A spherical aluminum tank with a diameter of 48", filled with $Pu(NO_3)_4$ solution to a height of 40". The plutonium is the "reactor-grade" isotopic mix.

6. Eight plutonium metal cylinders, 3 kg each, in a 2 x 2 x 2 array in a water-flooded storage container.

These systems will be used throughout this discussion for comparison purposes. Obviously, one should not consider these as generic systems, nor use them to represent real configurations. In fact, the point of much of this is to demonstrate the uniqueness of each situation. As I continue to stress, each system should be analyzed individually.

Neutron flux characteristics

In this section we will discuss the characteristics of the neutrons being emitted from a criticality event. There are three primary factors that should be considered in understanding the characteristics of the neutron field. These are the neutron leakage probabilities for the system (the same probabilities as discussed in the six-factor formula above), the presence of neutron reflectors near the system, and the energy spectra of the emitted neutrons. Besides these primary factors, one should also consider the potential impact that any intervening material, such as walls or shields, may have on either the total flux or the energy spectrum.

Neutron Leakage Probabilities

The neutron leakage probability is the probability that a neutron will escape the boundary of the fissionable system and not return. In the six-factor formula, this was broken into two parts for the fast and thermal components. Since the energy spectrum of the escaped neutron flux can vary from the spectrum in the system, we will not concern ourselves with these separate components, but only consider the overall leakage. The energy spectrum at the receptor location will be discussed below.

The percentage of neutrons that leak from a system is highly dependent on the specifics of the system. The size of a system can affect the leakage significantly. Larger systems such as solution tanks will generally have lower leakage, since the volume-to-surface area ratio is high and the fission density is fairly low. On the other hand, highly enriched metal systems are usually quite small, with a high fission density, resulting in higher leakage rates.

The leakage probability for a fissile system is difficult to estimate from first principles, as it is highly dependent on the physical composition of the system. The composition will also have a strong impact on the neutron spectrum within the assembly. Therefore, we will not go into determining either of these further in this document, but only demonstrate some values calculated for the example systems shown. This is not a complication for this report, as the spectrum in the assembly will not necessarily be representative of the spectrum at the receptor locations, and the leakage probability is the main parameter of the system that will be needed for later use. Spectral effects will be discussed further below.

Table 2 shows the results of leakage probabilities for some of the critical systems, as calculated by MCNP:

Table 2: Examples of Neutron Leakage Probabilities

System Description	**Neutron Leakage** (Out – In = Net)
Bare Uranium Metal (97% U-235)	56% - 0.02% = 56%
Concrete Reflected (3-sided) Uranium Metal (97% U-235)	65.6% - 10.1% = 55%
$Pu(NO_3)_4$ Solution in 48" Dia. Tank (97.4% Pu-239)	10.7% - 0.05% = 10.65%
3kg Pu-239 Pieces in 2x2x2 Flooded Storage Array (Leakage to outside of array)	42.8% - 7.1% = 35.6%

If there is no basis for estimating the neutron leakage, then one can make a simple assumption for a first-order approximation. We know that there are roughly 2.7 neutrons created per fission for uranium and it takes slightly more than one to keep the chain reaction going. If we assume that non-fission absorption is fairly low (if it were not, it would poison the system), then one can easily envision a leakage probability of around 50% for a small metal system. This is consistent with the results of table 2. For larger and more complex systems, such as those with non-fissionable components, then one would expect a lower leakage probability. Reflectors would tend to increase the apparent leakage of a system, as it will increase the fission density above the unreflected condition. Arrays go critical based upon the interaction of leakage neutrons between the individual units, which would imply the potential for high leakage probabilities, especially for accessible areas within the array.

Presence of Reflectors

For materials within a fissionable system, one is concerned with both the absorption and the scattering properties of the material. For materials in the proximity of the system, the scattering properties become the main concern. While reflectors will scatter neutrons back into the system, they will also scatter neutrons into the surrounding area, and may act to enhance the neutron flux

in occupied areas. Also, the scattering by reflectors can reduce the energy of the neutron spectrum of the radiation field. For example, the concrete reflected uranium system in table 2 is based on a subcritical quantity of material being put in a corner of a room. In this case the floor and the two intersecting walls reflect neutrons back into the uranium (note the 10% in-leakage in table 2), ***and also reflect neutrons back towards a person standing in the room.*** In the particular case from table 2, ***the neutron flux at the receptor location, one meter from the centerline of the system was 3 times higher than the flux at the same point for the unreflected system, for the same number of fission neutrons.***

Also, the effect of external reflectors is to reduce the amount of fissionable material necessary to go critical. Therefore, reflected systems are generally smaller, with higher leakage probabilities, than their unreflected counterparts.

Neutron Spectral Effects

As discussed, there are various factors that can affect the spectrum of the neutrons at a receptor location external to the fissionable system. Self-absorption within the system can result in a higher average energy, as demonstrated in figure 2. This example is a hypothetical unreflected metal system, composed of a 19.8 cm diameter sphere of 97% enrichment uranium metal. Since it is unreflected, it would be expected to have essentially a fission spectrum, as confirmed by the MCNP calculations. Note that the spectrum at the receptor location, one meter from the centerline of the sphere, clearly displays a shift towards the higher energy neutrons, as can also be seen by the average energy values. Since there are no other features present in this calculation, self-absorption induced hardening can be the only explanation.

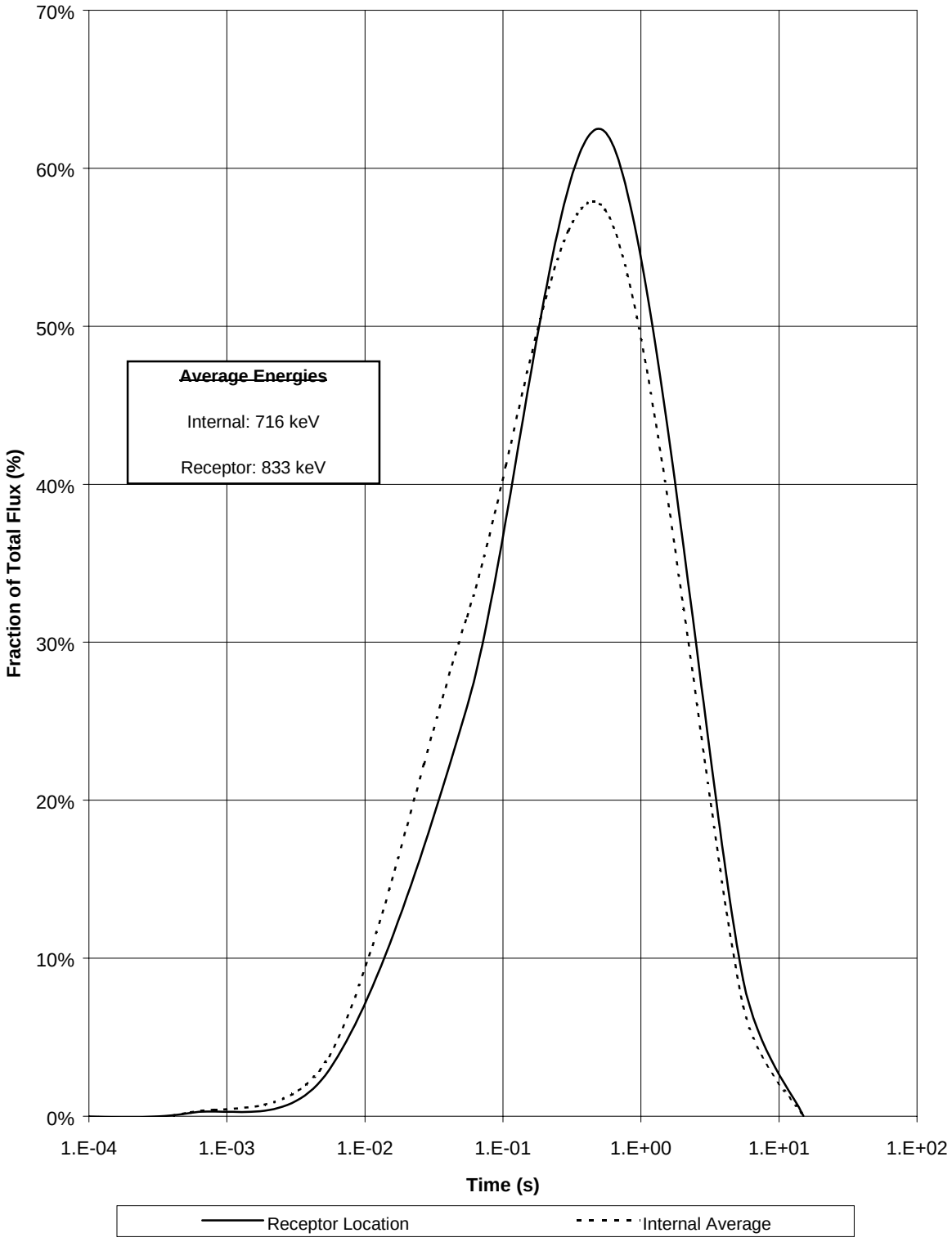

Figure 2: The internal and external neutron spectra for a bare, unreflected assembly of 97% enriched uranium metal.

Figure 3 is a plot of the spectra at the receptor location for several of the hypothetical systems listed in table 2. One can clearly see that for various types of systems, the spectra can vary by large amounts. It can also be seen that even for systems of identical internal composition, there can be significant differences in the spectra based on the presence and composition of reflectors.

As can be seen from figure 3, the neutron spectrum can change dramatically with only subtle changes in the configuration. Note for example, the differences between the unreflected and concrete-reflected uranium metal spheres, and between the concrete-reflected and steel-reflected U_3O_8 systems. The average neutron energies are listed in table 3 below.

Table 3: Average Neutron Energy at Receptor Location

System Description	Average Energy (keV)
Concrete Reflected 97% U_3O_8 Sphere	**0.55×10^{-3}**
Concrete Reflected 97% Uranium Metal Sphere	**0.43×10^{-3}**
Steel Reflected 97% U_3O_8 Sphere	**44.0**
Unreflected 97% Uranium Metal Sphere	**833.**
$Pu(NO_3)_4$ Solution in 48" Diameter Tank	**0.18×10^{-3}**
Pu Metal in 2x2x2 Array, Flooded	**0.41×10^{-3}**

Note that Table 3 demonstrates a range of energies that spans six orders of magnitude. As will be seen later, this will translate into a range of two orders of magnitude in the dose conversion factors for determining the neutron dose to a victim. ***Consequently, this demonstrates the importance of being able to determine the neutron spectrum for the exact situation under study. It is difficult to estimate the neutron spectrum without detailed information regarding the exact condition of the system and its surroundings.***

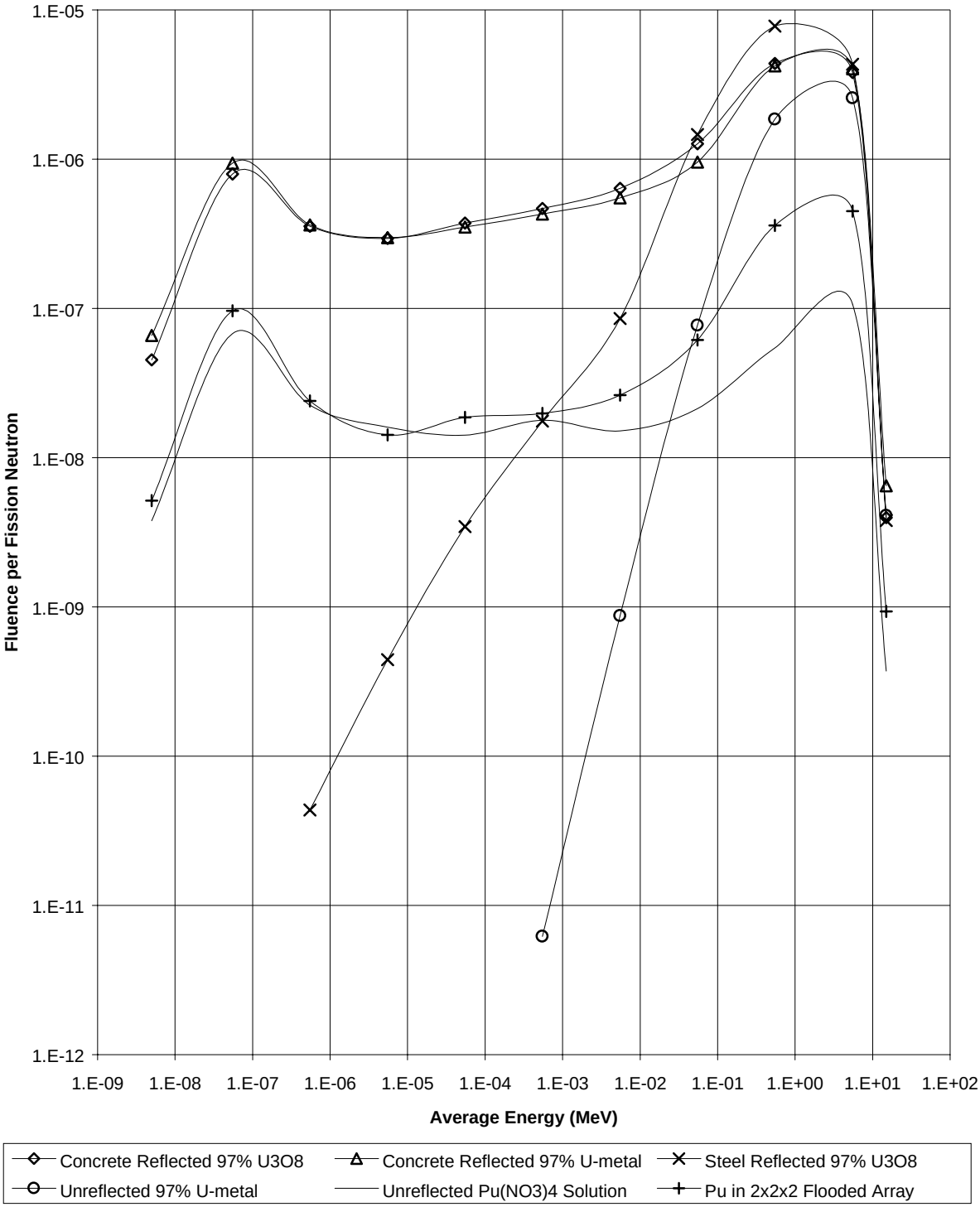

Figure 3: Neutron spectra at an external receptor location for various combinations of fissionable systems and reflectors.

Gamma flux characteristics

There can be several sources of gamma radiation during and after a criticality event, again depending on the exact configuration of the system and its surroundings. Contributors to the gamma flux at the receptor location are ***fission gamma rays, fission-product-decay gamma rays, capture gamma rays, inelastic-scatter gamma rays, reaction-product gamma rays, activation-product gamma rays, annihilation radiation,*** and ***bremsstrahlung***[D]. While this list pretty much covers the entire range of possible sources, we can reduce the discussion somewhat by dividing them into groups based on the timeframes in which they are significant. For the purpose of this discussion, let's define the following groups:

- ***Prompt gamma rays*** are those which are produced during the fission process. This would include both fission gamma rays, and those produced during neutron interactions with matter, such as capture, reaction-product, and neutron scatter gamma rays. Once the fission process is terminated, these would not be present.

- ***Early decay gamma rays*** are those emitted during the decay of short-lived fission and activation products that contribute to the gamma radiation field immediately after the event. This group will contribute to the acute doses to the victims before they can be evacuated from the area. While it is difficult to define a universally applicable timeframe, one convenient value would be the first 45 seconds, as this has been evaluated in the literature (ref. D).

- ***Late decay gamma rays*** are those emitted from the decay of longer-lived fission and activation products. This group will contribute to the residual radiation field after an event, resulting in doses to rescue and recovery personnel.

Due to the processes involved, the gamma radiation field associated with any criticality event will be highly dependent on the exact physical arrangement involved. This is apparent from the various mechanisms for creating the photons described above. Therefore we cannot create a detailed estimate of the dose that one would receive, but we can discuss and estimate major contributors to the doses from the event itself.

Prompt Gamma Rays

Gamma rays emitted simultaneously with the fission process have been extensively studied for ^{235}U, and much less so for other fissionable materials (at least at the time ref. D was published). Therefore, we will assume ^{239}Pu to yield similar results. These emissions have been found to produce a fairly continuous spectrum of energies from 10 keV up to about 10 MeV. For those gamma rays emitted during the first 50 nanoseconds after fission, there is an average of 8.1 photons per fission, carrying off 7.25 MeV per fission event. In addition, another 0.98 MeV of energy is emitted during the rest of the first second after fission. Short-lived fission products contribute another 0.3 MeV/fission of gamma radiation in the first second (ref. D). The number of photons emitted per fission in the last two cases is not given in the reference, but for our purposes we can assume that each is represented by one photon carrying the total energy per fission.

Capture and neutron scattering gamma rays can also be significant contributors in this initial phase, but are harder to estimate. Materials in the immediate area of the event emit them, and therefore one must know the specifics of the physical layout of the event. Photons from capture events can be very energetic, ranging from 2.2 MeV up to about 11 MeV, with neutron scattering photons being typically lower in energy. Both types of emissions depend on the initial neutron energy, but in opposite directions. Capture gamma rays will be higher for lower energy neutrons, where capture cross-sections are higher, but neutron scattering photons will increase with higher neutron energies.

Reaction-product gamma rays are produced in events similar to inelastic neutron scatters, but a particle other than a neutron is emitted from the absorbing nucleus. For example, the $^{10}B(n,\alpha)^{7}Li$ reaction emits a gamma ray of 0.5 MeV. However, since only relatively few isotopes react in this way, we will ignore them for now.

Early Decay Gamma Rays

According to reference D, about 6.65 MeV per ^{235}U fission is emitted by fission products as delayed gamma rays. Over 75% of this is released within the first 1000 seconds. This has been broken down further, as shown in table 4:

Table 4: Gamma Emissions Between 0.2 and 45 Seconds after Fission (ref. D)

Isotope	Photons/fission	MeV/fission	MeV/photon
^{233}U	**2.02**	**1.97**	**0.975**
^{235}U	**3.31**	**3.18**	**0.961**
^{238}U	**5.50**	**5.08**	**0.924**
^{239}Pu	**3.26**	**2.86**	**0.877**

As with capture gamma rays, the quantity of activation gamma rays emitted in the first 45 seconds will be strongly dependent on the specifics of the physical layout. However, one must be sure to include consideration for the $^{16}O(n,p)^{16}N$ reaction in air and water, which emits 6.1 and 7.1 MeV photons with a half-life of 7.13 seconds.

Late Decay Gamma Rays

Late decay gamma rays will be composed of contributions from both longer-lived fission products and activation products. The long-term fission product contribution can be estimated directly from the total number of fissions, but the quantity and type of activation products will be specific to the situation. Therefore, we will not discuss these further in this document, other than to note that they are important to consider when evaluating rescue and recovery operations.

Environmental concerns

The primary environmental concerns for a criticality accident are the intense, energetic neutron and gamma radiation fields accompanying the event itself. Depending on the presence or absence of shielding, these can extend for large distances from the source. While doses exceeding thresholds for acute injury and fatality will typically be limited to the immediate

vicinity of the event, significant doses can be received by collocated personnel, potentially exceeding annual regulatory limits. In addition, depending on the situation, members of the public could also be exposed.

Environmental releases of radioactive material can also occur, and should be considered for completeness. Depending on the exact situation, these may or may not have significant impact on the situation. Both DOE and the NRC have discussions of techniques for evaluating these releases in their accident analysis handbooks. DOE's version is *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, DOE-HDBK-3010-94, volume I. It is available online at http://tis.eh.doe.gov/techstds/standard/appframe.html (be forewarned, this is the full document, and is over 3.9 megabytes in size).

In either case, the approach is the same. First, the airborne source term is determined from

$$\textbf{Airborne Source Term} = \Sigma(MAR_i \times DR_i \times ARF_i \times RF_i)$$

where

MAR_i= **Material-at-Risk,** the inventory of fissionable material or radionuclide of species *i* available in the system or generated in the excursion;

DR_i= **Damage Ratio**, the fraction of the MAR_i actually impacted by the accident-related conditions;

ARF_i= **Airborne Release Fraction**, the coefficient used to estimate the amount of material suspended in air as an aerosol and thus available for transport due to the mechanical stresses of the event; and

RF_i= **Respirable Fraction**, the fraction of the airborne radionuclides that can be transported and inhaled into the human respiratory system (typically assumes 10-μm AED and less).

After the airborne source term has been determined, then it is dispersed to the environment using the normal atmospheric dispersion models generally used for radioactive material releases.

The DOE handbook assumes that RF is always 1.0 for criticality accidents. Values for MAR are expected to be evaluated from either a computer code such as ORIGEN2 or using tables provided in the handbook or NRC Reg. Guides 3.33, 3.34, or 3.35, depending on the fissionable material. [It should also be noted that there is a potential for very short-lived gases and halogens to be released into the local area around the accident. These isotopes would not normally be considered in estimating environmental consequences since they would decay before transported any significant distances, however they could contribute significantly to personnel doses in the immediate area. Be careful, because these isotopes may not be included in some tabulations such as those in the DOE handbook.] The other values used in the equation are summarized below:

Table 5: Summary of DRs and ARFs for Airborne Source Terms

Scenario	MAR Type	DR	ARF
Solution Systems	**Noble Gases generated by event**	**1.0**	**1.0**
	Radioiodines generated by event	**1.0**	**0.25**
	Non-volatiles in quantity evaporated	**1.0**	**0.0005**
	Radioruthenium in reprocessing solutions	**1.0**	**0.001**
Fully Moderated/Reflected Solids (metal pieces)	**Noble gases generated by event**	**0.1**	**0.5**
	Radioiodines generated by event	**0.1**	**0.05**
Fully Moderated/Reflected Solids (fines, powders)	**Noble gases generated by event**	**1.0**	**0.5**
	Radioiodines generated by event	**1.0**	**0.05**
Bare, Dry Solids (metal pieces)	**Noble gases generated by event**	**0.1**	**0.5**
	Radioiodines generated by event	**0.1**	**0.05**
Large Storage Arrays (Upper Bound Only)	**Noble gases generated by event**	**1.0**	**0.5**
	Radioiodines generated by event	**1.0**	**0.05**

Past experiences have also shown that criticality accidents are often accompanied by significant localized contamination. This is most likely resulting from a combination of the material releases discussed above in this section and activation of dust and loose material in the vicinity of the event. While this is a hazard for the personnel involved in rescue and recovery operations, it can also be a source of environmental releases. While not a criticality accident, it was observed at the plutonium glovebox fires of the 1950's and 1960's at Rocky Flats that a primary environmental release was due to the spread of contamination during firefighting and recovery operations. This can be evaluated using typical resuspension models. For example, DOE-HDBK-3010-94 recommends a bounding value of 4×10^{-5} /hour as an airborne release rate, with a respirable fraction of 1.0, for a freshly deposited, homogeneous bed of powder exposed to ambient conditions.

When dealing with the environmental releases from a criticality accident, it must be recognized that a large fraction of the noble gases and halogens produced have short halflives. Within the realm of a normally operating system, these radioisotopes would rarely be present since they would decay before release. However, in the case of many criticality accident scenarios, one cannot assume that a ventilation system is designed and optimized for dealing with such releases. Therefore, there are two basic approaches to evaluating doses from these shortlived isotopes.

The first approach would be to assume that the criticality accident directly releases the material into the ambient airstream, resulting in immediate dispersion to the environment. Radioactive decay could be accounted for during the transit time to the target of interest if desired. This assumption would tend to overestimate the resulting doses, and would therefore be conservative.

However, it would be desirable and sometimes necessary to consider the more realistic case where the system disperses the material into a room at one characteristic rate, and the material is removed from the room at a different rate characterized by the flow of the ventilation system.

This results in a time delay between production and release to the environment. While this approach is in general a much more complex system than the first, there is a method for approximating the fraction of material that would decay during the time delay versus the fraction that would be released outside the room. From Appendix B of ICRP 37, the concentration of radioactive material in a room as a function of time (the "ventilation equation") is

$$C(t) = \frac{A}{Q + \lambda\mu}\left(1 - e^{-((Q+\lambda\mu)/\mu)t}\right) + C_o e^{-((Q+\lambda\mu)/\mu)t}$$

where

$C(t)$	is the concentration in the volume at time t (activity/volume);
A	is the activity entering the volume per unit time (activity/second);
Q	is the ventilation flow rate (volume/second);
λ	is the radioactive decay constant (sec^{-1}); and
μ	is the room volume.

If the contaminant is stable, that is, does not decay radioactively, then the equation reduces to

$$C(t) = \frac{A}{Q}\left(1 - e^{-(Q/\mu)t}\right) + C_o e^{-(Q/\mu)t}.$$

Also, if one assumes that the ventilation flow rate is expressed in the room exchange rate (room changes per unit time) rather than based on unit volumes, then μ becomes 1, and $C(t)$ becomes the total quantity of activity in the room at time t. (This is convenient in that it reduces the amount of information necessary in order to evaluate a postulated scenario.)

Now, assume that the concentration in the room reaches equilibrium between production and removal (in effect, letting t approach infinity) in both equations, and take the ratio of the two. The resulting expression,

$$Ratio = \frac{Q}{Q + \lambda} = \frac{1}{1 + \lambda/Q}$$

is the fraction of material that is removed by the ventilation system after decay. This result yields a simple relationship that allows the estimation of the fraction of a shortlived radioisotope that would escape from the room into the environment. While this ratio does not account for the time-dependency of the release, it should provide a reasonable approximation for the purpose of this document.

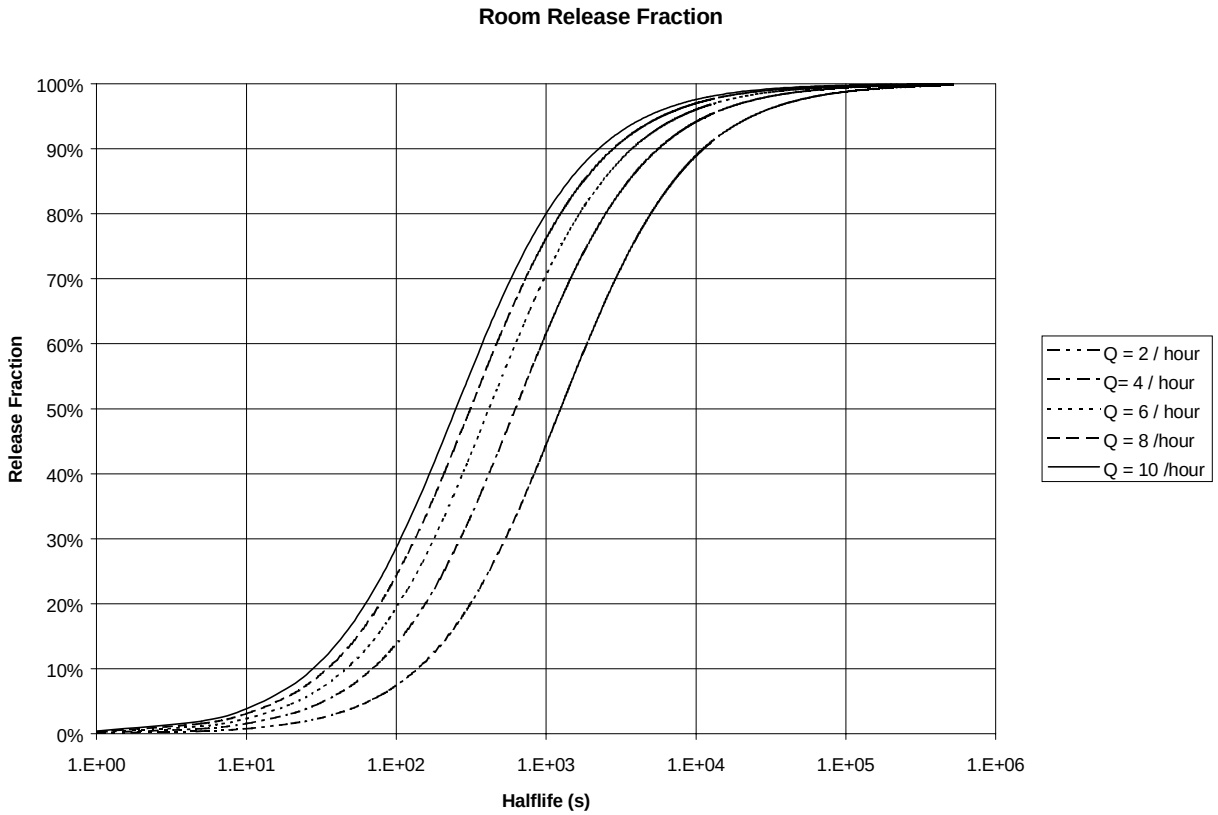

Figure 4: Room release fraction as function of radioisotope halflife.

The result of this estimation is shown in figure 4 for several possible room exchange rates. Note that for halflives of less than 1 second, essentially all decay occurs in the room, and for those of greater than about 1 day (86,400 s), essentially all escape without decay. This relationship can be used directly to modify the airborne release fractions discussed above to provide a more realistic estimate of the radiation doses to individuals outside of the building where the criticality accident occurs.

Tools for determining radiation hazards

Besides the computer codes already discussed for determining neutron radiation fields, there are several tools available for evaluating the hazards present from a criticality accident. In the hands of a knowledgeable person, MCNP may also be able to provide adequate information regarding the gamma radiation fields (sorry, no tips from me - I am not that much of an expert with MCNP).

Also of use for neutrons is IAEA Technical Report 180, Compendium of Neutron Spectra in Criticality Accident Dosimetry, 1978, which contains measured and calculated neutron spectra for a wide variety of fissionable systems. This document also included spectra from reflected sources and through various shields.

For environmental impacts, the methodology of the accident analysis handbooks will only provide an estimate of the airborne source terms. Additional tools are necessary for estimating the dispersion of this material into the environment. While simple dispersion calculations can be performed with the basic Pasquill-Gifford dispersion models[E], computer codes such as HOTSPOT from LLNL, or MACCS2 can perform more detailed evaluations of environmental and population impacts from such an accident.

Operational experiences with critical systems

There have been about 60 criticality accidents since 1945. In the Free World there have been 9 nonreactor process accidents, including the recent one in Japan, and 33 accidents involving either reactors or critical experiments. There have also been at least 14 criticality accidents in the Former Soviet Union, although details on those are somewhat sketchy.

By my count (not official by any means) this has resulted in about 48 fatalities and more than 69 significant non-fatal exposures. This breaks down as follows:

Non-Soviet:

- 3 fatalities in process accidents (LASL 1958, Wood River Junction 1964, JCO 1999)
- 3 fatalities in critical experiments (LASL 1945, LASL 1946, Russia 1997)
- 4 fatalities in reactor accidents (SL-1 1961 (3), Brazil, 1980)
- 69 non-fatal significant exposures ranging from 3 to ~500 rem

Former Soviet Union (pre-1997):

- 6 fatalities in 12 accidents (3 in one solution-based critical experiment accident)
- 1 individual blinded
- 1 amputation of both legs
- 17 cases of “radiation sickness”
- ~30 – 33 acute fatalities at Chernobyl
- ??? non-fatal significant exposures at Chernobyl

These accidents have been discussed extensively in the literature[F], and this document will not try to repeat the descriptions here. However, in Knief’s book (see ref. E, first bullet) there is an analysis of the frequent elements and factors for criticality accidents in U.S. processing plants that does bear repeating here:

Table 6: Frequent Elements and Factors in Criticality Accidents in U.S. Processing Plants (Knief 1996)

Causal Element or Contributing Factor	Number of Criticality Accidents Involved
Critical configuration of liquids	**7**
Bulk transfer to unsafe vessel	**6**
Unintended transfer	**3**
Ignorance of concentration in intended transfer	**3**
Valve problems	**5**
Motive force due to high-pressure air	**2**
Poor operational communication	**≥ 2**
Lack of current knowledge of system configuration	**≥ 2**
Development of dangerous routine practices	**≥ 2**
Errors of commission by operators	**≥ 2**
Errors by supervisors and managers	**≥ 2**
Existence of "abnormal" conditions	**≥ 2**

Although a few years old, the data in table 6 still provides valuable insight into the types of factors that have led to criticality accidents in process systems. One can easily insert the recent Russian solution accident of 1997 and the latest accident in Japan. Both would fit into the categories of "bulk transfer to unsafe vessel", "ignorance of concentration in intended transfer", "lack of current knowledge of system configuration". The Japanese accident would also fit "development of dangerous routine practices", and both of the "error…" categories.

Note that a simple review of this table would suggest essentially all of these accidents resulted from administrative and conduct of operations type errors. Even though the "bulk transfer to unsafe vessel" category would suggest potential design problems, most of these accidents involved actions that took the configuration outside of the established safe design of the vessels, although some did involve the use of unapproved vessels.

If one were to assemble a similar table for the reactor and critical assembly accidents, it could be expected to yield a similar result. Therefore, it is important to remember that ***the most important factor in achieving a safe operation with fissile materials is the formality of the operation and the degree of rigor with which those operations are conducted. The operating staff must be fully knowledgeable of the design envelope within which they operate, and be fully aware of the risks involved. They must always respect the hazard.*** There are multiple examples where the mindset of the employees involved strongly influenced their behavior, leading to the development of dangerous practices.

While I have not tried to compile them, there are also many examples of failures of criticality controls that could feasibly have led to accidents also. These should always be considered as precursor events and treated very seriously. Too often (at least in DOE) it appears that these situations are dealt with casually, because the mindset seems to conclude that there was never an unsafe condition.

It is often observed that criticality safety controls are dominated by administrative rather than physical controls. While it is recommended that systems be designed to physically control at least two independent parameters (see table 1), it is often difficult to do in practice. Therefore, systems often depend on multiple controls of single parameters, and frequently administrative controls are used in combination with physical controls. This should always be done with caution, as the experience base indicates that administrative controls are often the weakest link in the operation.

One other point to be made regarding the past experience base. The most common characteristic of these accidents is that each is quite unique and unanticipated, and obviously not predictable. On the opposite hand, it is occasionally observed that safety analyses of fissile material processes assume that the criticality safety controls result in dropping the accident frequency down to the "incredible" or "extremely unlikely" probability bins (i.e., a frequency of less than 10^{-6} per year). From this discussion, one can see that this should always be done with great caution, since it will affect the mindset of the operating staff.

Radiation doses from criticality accidents

Up to this point, this discussion has attempted to provide a fundamental understanding of the physics, behavior, and characteristics of critical systems. Given that basis, we can now discuss the human consequences from such an event. We have seen that critical systems have basic commonalities in the behavior of the system and the types of radiation fields. However, it is also obvious that the specifics of each situation will have a strong impact on the relative significance of each of these factors in determining the consequences from an event. Therefore it is obvious that each situation must be evaluated separately, but the tools used are generally common to all situations.

In this section we will discuss the basic techniques for evaluating the consequences from a criticality accident. It should be noted that, once the characteristics of the radiation fields are established, the techniques for estimating radiation doses are the same as would be applied for any other health physics application.

Estimation of doses for postulated accidents

The first thing to remember is that for the directly involved victims of a criticality accident, radiation doses will likely exceed thresholds for the onset of acute injury or death. [While we have not discussed the biological effects of high doses of radiation, they are well documented in the literature, and should be familiar to practicing health physicists.[G]] These deterministic effects are based on the absorbed dose to the individual, and not the dose equivalent, which is intended to estimate stochastic effects like cancer induction. Therefore, it is customary and appropriate that doses be evaluated in terms of absorbed dose in **rad** or **gray**. For collocated workers and other incidental exposures, it will still be appropriate to determine the dose equivalent, but we will not concern ourselves with that here, as it will typically only involve the inclusion of the quality factor (QF) into the dose calculations.

As with any radiation field, the important characteristics to know for dose assessment are the type, intensity, and energy spectrum. Once these are determined, then the appropriate dose conversion factors can be applied. In the case of criticality accidents, this is not always straightforward without detailed information, as has been stressed previously. However, one can make some assumptions that simplify matters for first order estimates. While this may not be recommended in general practice, these assumptions help to gain an appreciation for the relative contributions from the different components of an exposure to a criticality event. The basic assumptions that we will apply in this section are:

1. The radiation fields (both neutron and photons) will be assumed to follow the inverse-square law with distance from the critical system. The effects of room scattering and spectral changes will be ignored for photons.

2. Scattering of neutrons outside of the system will only be considered if the scenario suggests that there is a partial reflector near the system, such as a wall, that will scatter neutrons towards the receptor. In this case, the reflected neutrons will be treated as a simple multiple of the source neutrons.

3. There are no secondary sources of neutrons, such as photoneutron production, that will increase the number of neutrons from the system. Consequently, the total number of neutrons generated is a direct function of the number of fissions.

4. The energy spectrum of the neutrons at the receptor will be the same as the spectrum of the neutrons leaking from the system. No corrections for in-transit spectral changes will be made.

Neutron Dose

The accurate calculation of the neutron dose depends on the degree of knowledge available regarding the neutron spectrum. This is due to the fact that the dose conversion factors for neutrons are dependent on energy and the neutron spectrum can range over eight orders of magnitude, depending on the situation. Therefore, the conversion factors must be 'folded' together with the spectrum to arrive at an accurate dose estimate. While one can in principle assume an average energy for the neutrons and use a single-point estimate, it will be seen that these are of somewhat limited value.

For neutron conversion factors, I elected to create them from the table of quality factors for neutrons in DOE's 10CFR835 (I have not compared these to 10CFR20, but would expect them to be the same, since they are both based on ICRP 38). Table 7 contains the relevant information for creating these factors.

Table 7: Neutron Absorbed Dose Conversion Factors

Energy (MeV) (Note 1)	Quality Factor (Note 1)	$\frac{n/cm^2\text{-}s}{100\ mrem/40\ hrs}$ (Note 1)	$\frac{rem}{n/cm^2}$ (Note 2)	$\frac{rad}{n/cm^2}$ (Note 3)
1×10^{-8}	2	680	1.02×10^{-9}	5.1×10^{-10}
1×10^{-7}	2	680	1.02×10^{-9}	5.1×10^{-10}
1×10^{-6}	2	560	1.24×10^{-9}	6.2×10^{-10}
1×10^{-5}	2	560	1.24×10^{-9}	6.2×10^{-10}
1×10^{-4}	2	580	1.2×10^{-9}	6.0×10^{-10}
1×10^{-3}	2	680	1.02×10^{-9}	5.1×10^{-10}
1×10^{-2}	2.5	700	9.92×10^{-10}	3.97×10^{-10}
1×10^{-1}	7.5	115	6.04×10^{-9}	8.05×10^{-10}
1.0	11	19	3.65×10^{-8}	3.32×10^{-9}
10.0	6.5	17	4.09×10^{-8}	6.28×10^{-9}
20.0	8	18	6.31×10^{-8}	7.89×10^{-9}

(1) Value taken directly from 10 CFR 835.2.
(2) Value rem/(n/cm^2) obtained by dividing column 3 into 6.944×10^{-7} rem/s.
(3) Value rad/(n/cm^2) obtained by dividing column 4 by the quality factor.

[It should be noted that the energies selected in table 7 were chosen to correspond with energy ranges that I used in calculating neutron energy spectra with MCNP. One could certainly use a different approach, depending on the form of the conversion factor necessary for the application.]

Using these conversion factors and the neutron spectrum from a modeled scenario, one can directly estimate the neutron dose at some receptor location. Following the examples that have been previously discussed and graphed in figure 3, one obtains the results shown in table 8.

Table 8: Neutron Absorbed Dose per Fission Neutron at Receptor Location

System Description	Average Energy (keV)	Absorbed Dose rad/(fission neutron)
Concrete Reflected 97% U_3O_8 Sphere	0.55×10^{-3}	4.1×10^{-14}
Concrete Reflected 97% Uranium Metal Sphere	0.43×10^{-3}	4.2×10^{-14}
Steel Reflected 97% U_3O_8 Sphere	44.0	5.43×10^{-14}
Unreflected 97% Uranium Metal Sphere	833.	2.24×10^{-14}
$Pu(NO_3)_4$ Solution in 48" Diameter Tank	0.18×10^{-3}	9.54×10^{-16}
Pu Metal in 2x2x2 Array, Flooded	0.41×10^{-3}	4.18×10^{-15}

From table 8 one notices the broad spread between the different values of absorbed dose, varying by as much as a factor of 56 between the highest and lowest values. This demonstrates the accumulation of the effects we have discussed previously. Doses will tend to go down as neutron energy goes down, but partial reflectors such as walls would increase the flux at the receptor. If we use row 4 as a reference point (no reflectors and essentially a fission spectrum), then one can make the following observations:

- Rows 1 and 2 were higher due to the effects of the 'wall' reflector, but this was reduced somewhat by the significant moderation of the neutrons.

- Row 3 was higher due to the 'wall' also, but with the steel reflectors there was less moderation, so the average energy was higher than rows 1 and 2, resulting in a higher dose.

- Row 5 is lower due to the lower leakage probability of the larger volume system, and the low average energy due to moderation.

- Row 6 is not as low as row 5 due to somewhat higher leakage from the compact, fully reflected array configuration.

One would be tempted to use the average neutron energy to calculate a dose rather than spectrum-weighting the conversion factors as done above. This could be based on a simple formula

$$D_{neutrons} = \frac{\upsilon F \rho_l}{4\pi r^2} xDCF(E_{ave}), \quad (7)$$

where

υ	= number of neutrons per fission;
F	= number of fissions;
p_l	= total leakage probability;
$DCF(E_{ave})$	= Dose Conversion Factor for average neutron energy; and
r	= Distance to receptor in centimeters.

For first order estimates, one could use the leakage probabilities and average energies determined in this document, as long as they were appropriate for the system under study.

Note though that this should be resisted unless there is no other information available, since as seen from figure 3, many of these spectra tend to be bimodal. The average energy is not necessarily representative of the full spectrum. In order to demonstrate this effect, table 9 contains the resulting values.

Table 9: Point Estimates of Neutron Dose per Fission Neutron

System Description	Average Energy (keV)	Absorbed Dose rad/(fission neutron)
Concrete Reflected 97% U_3O_8 Sphere	0.55×10^{-3}	6.8×10^{-15}
Concrete Reflected 97% Uranium Metal Sphere	0.43×10^{-3}	6.7×10^{-15}
Steel Reflected 97% U_3O_8 Sphere	44.0	8.2×10^{-15}
Unreflected 97% Uranium Metal Sphere	833.	1.4×10^{-14}
$Pu(NO_3)_4$ Solution in 48" Diameter Tank	0.18×10^{-3}	1.7×10^{-16}
Pu Metal in 2x2x2 Array, Flooded	0.41×10^{-3}	5.94×10^{-16}

These results demonstrate dose estimates that are up to 7 times lower than the spectrum-weighted results of table 8. Note that row 4, the bare metal sphere, is only a factor of 1.6 lower, as one would expect due to the minimal spread in the spectrum since there was no moderator or reflector. All other cases ranged from 5.6 to 7.0 times lower.

Finally, we can arrive at an estimate of absorbed dose from neutrons for a likely criticality accident scenario. The doses listed in tables 8 and 9 above are normalized to 1 fission neutron. If one assumes an event resulting in 1×10^{17} fissions, and using 2.5 neutrons/fission for ^{235}U and 3.0 neutrons/fission for ^{239}Pu, then one can estimate the neutron doses as shown in table 10.

Table 10: Neutron Dose at 1 meter from 10^{17} Fissions

System Description	Absorbed Dose (rad)
Concrete Reflected 97% U_3O_8 Sphere	10,250
Concrete Reflected 97% Uranium Metal Sphere	10,500
Steel Reflected 97% U_3O_8 Sphere	13,575
Unreflected 97% Uranium Metal Sphere	5600
$Pu(NO_3)_4$ Solution in 48" Diameter Tank	286
Pu Metal in 2x2x2 Array, Flooded	1254

As an alternative approach, one might consider using ***An Updated Nuclear Criticality Slide Rule***, NUREG/CR-6504[H]. This document provides a 'cookbook' style process for evaluating criticality safety accident situations and provides estimates of doses for a variety of basic configurations of fissionable systems. Of particular interest are the appendices of volume 1 (the technical basis document) that contain several procedures for initial response considerations should an event occur. Volume 2 contains a number of sliding graphs for estimating doses from the evaluated systems. Note that this document only deals with uranium systems of various enrichments, and no plutonium systems. Unfortunately (for me, at least) NUREG/CR-6504 looks at different systems than those considered in this document. However, one is similar enough to allow comparison. For a 93.2% enriched sphere of uranium metal, the slide rule yields a dose of about 2400 rad neutrons at 1 meter. This is comparable to the 97% metal system above. The difference of the factor of 2 could be due to the higher enrichment of our system, which would result in a smaller physical system with a higher leakage probability.

Gamma Dose

The estimation of the gamma dose rate from an accident is complicated due to basically the same reason the neutron dose is difficult to determine – one should have detailed information about the physical configuration of the system, and the layout and characteristics of the location. In this discussion, we have ignored photons generated by capture, neutron scattering, and reaction-product interactions both within the fissionable system and the surrounding area. This is not due to the fact that they are negligible, but rather due to the fact that they cannot be determined for a generic situation. They should certainly be considered to the extent possible for an actual evaluation.

Given this limitation, we can still estimate the gamma doses from the prompt gamma rays generated during the fission process and the early decay gamma rays from the fission products. We will ignore the late gamma rays from fission product decay (those after the first 45 seconds), but not because they are indeterminable as the others discussed above. But rather, in most past accidents the personnel involved have generally evacuated or been removed from the room very quickly after the first pulse. Therefore, we will assume that they are only exposed to the system for the first 45 seconds, and that the later radiation levels would only be of concern for rescue/recovery operations.

Recall from our previous discussion on the photon radiation fields the following results:

- Prompt gamma rays: 7.25 Mev per 8.1 photons (average energy 0.895 MeV) within the first 50 nanoseconds; 0.98 MeV per 1 photon additional up to first second; and 0.3 MeV per 1 photon from very short-lived fission products in first second.

- Early decay gamma rays: For ^{235}U, 3.18 MeV per 3.31 photons (average energy 0.961 MeV) in first 45 seconds. For ^{239}Pu, 2.86 MeV per 3.26 photons (average energy 0.877 MeV) in first 45 seconds.

For photon flux-to-dose conversion factors, I generally use a table from ANS-6.1[l], although there are probably newer tables that should be considered. Pulling only the values necessary for the appropriate energies and converting to proper units:

Table 11: Appropriate Photon Flux-to-Dose Conversion Factors

Photon Energy (MeV)	rad per photon/cm^2
0.3	**2.11×10^{-10}**
0.877	**4.99×10^{-10}**
0.895	**5.06×10^{-10}**
0.96	**5.33×10^{-10}**
0.98	**5.41×10^{-10}**

For photons we can use an equation similar to equation 7. As with neutrons, there will be some scattering and spectral softening that occurs with photons. However, since we do not have sufficient information (even less than what is available for the neutron case), we will ignore these issues and apply a simple inverse-square law approximation.

$$D_{photons} = \sum_{all\ E_{ave's}} \frac{\delta(E_{ave})F}{4\pi r^2} xDCF_p(E_{ave}), \quad (8)$$

where

F = number of fissions;
$\delta(E_{ave})$ = number of photons of average energy E_{ave} per fission;
$DCF_p(E_{ave})$ = Dose Conversion Factor for photons of average energy E_{ave}; and
r = Distance to receptor in centimeters.

Applying this method to the previous scenarios yields the following estimates of gamma doses.

Table 12: Gamma Dose at 1 meter from 10^{17} Fissions

System Description	Absorbed Dose (rad)
Concrete Reflected 97% U_3O_8 Sphere	**5263**
Concrete Reflected 97% Uranium Metal Sphere	**5263**
Steel Reflected 97% U_3O_8 Sphere	**5263**
Unreflected 97% Uranium Metal Sphere	**5263**
$Pu(NO_3)_4$ Solution in 48" Diameter Tank	**5153**
Pu Metal in 2x2x2 Array, Flooded	**5153**

These numbers are clearly higher than one might expect, by comparison with both NUREG/CR-6504 and with doses determined from actual events when personnel were within this range of the system. Comparison with the *Slide Rule* suggest that the bare metal sphere gamma dose above could be 17 – 20 time high, and the solution system dose could be a factor of 5 high. Therefore, let's consider what may have led to this situation:

- From table 12, note that gamma doses are the same for all systems of the same fissionable material. This is a limitation of the simplifications that we took in arriving at the estimates. In reality, the doses would vary based on system configuration, mainly due to self-absorption within the system. The effectiveness of self-absorption would be based on the density and volume of the fissionable system. This would certainly be highest for the four uranium case studies (rows 1 – 4). The uranium metal density is about 17 gm/cm^3, and the U_3O_8 density is about half of that. The solution systems are much lower.

- This estimate assumed a point-energy value for the gamma rays. However, in reality these photons will cover a broad range of energies, and a spectrum-weighted conversion factor should be used, similar to that applied to the neutrons. For example, the major contributors to table 12 are the prompt fission gamma rays.

Using a more detailed approach based on the spectral data provided in reference D, a spectrum-weighted conversion factor for the prompt fission gamma rays was calculated to be 4.48×10^{-10} rad/(photon/cm^2). This only lowered the prompt dose contribution by about 12%, resulting in an overall decrease of less than about 9%. Therefore, one can conclude that self-absorption is probably the main reason for the discrepancy. This conclusion would be supported by the comparison with the *Slide Rule* discussed above, the difference was higher for the denser systems than those less dense.

Let us consider the effect of self-absorption a little further to try and improve these estimates. The self-absorption correction factor for large spherical gamma ray sources can be estimated by[J]

$$I/I_o = \frac{3}{4(\mu a)^2}\left[\mu a + e^{-2\mu a} - \frac{1}{2\mu a}\left(1 - e^{-2\mu a}\right)\right] \qquad (9)$$

where μ is the linear attenuation coefficient and a is the radius of the sphere.

In a spreadsheet, this equation was used to recalculate the spectrum-weighted dose conversion factor discussed above to include self-absorption correction for the U_3O_8 and the uranium metal systems. Corrections were also applied to the other gamma ray contributors based on the average energy of the photon. These correction factors are significant reductions from the original estimates. Table 13 demonstrates the magnitude of the corrections.

Table 13: Corrected Gamma Dose at 1 meter from 10^{17} Fissions

System Description	Original Absorbed Dose (rad)	Corrected Absorbed Dose (rad)
Concrete Reflected 97% U_3O_8 Sphere	**5263**	**323**
Concrete Reflected 97% Uranium Metal Sphere	**5263**	**323**
Steel Reflected 97% U_3O_8 Sphere	**5263**	**323**
Unreflected 97% Uranium Metal Sphere	**5263**	**253**

The values listed in table 13 are very similar to values expected from comparison with the other sources discussed above. The relative contributions from the two main gamma ray groups are shown in table 14, demonstrating that the predominant contributor are the prompt fission gamma rays.

Table 14: Relative Contributions of the Gamma Ray Groups

Gamma Ray Group	U_3O_8 System	U-metal System
Prompt Fission Gamma Rays	**69.6%**	**69.4%**
Early Fission Product Decay	**30.4%**	**30.6%**

The slight differences between the two systems are most likely due to the effects of the spectral weighting of the lower energy prompt fission gamma rays in the denser system (U-metal) versus the higher energy early fission product decay gamma rays.

Determination of doses from actual accidents

In the event of an actual criticality event, it is very important to attempt to estimate the dose of the victims as quickly as possible to aid in their medical treatment. However, this is extremely difficult to do accurately, because of the need for detailed information regarding the radiation fields involved. For the neutron doses, the energy spectrum must be determined, and this will take some time, as the techniques are rather complex. However, some traditional techniques are available to help with the initial estimates, as discussed in reference G, and some new techniques are becoming available.

Initial Screening of Potential Victims

When an event occurs, it is important to screen all personnel that could have been in the general area, as some, especially those with lower doses, may not display any symptoms immediately. This is traditionally done by using a simple direct survey with a Geiger-type survey instrument to monitor for neutron activation within the body of the personnel. The detector should be calibrated to indicate a response of 3200 cpm in a 1 mR/hr gamma radiation field. Hold the detector against the abdominal area and record the response. If there is a response, segregate the individual for further monitoring. The neutron dose can be estimated by the equation

$$D = 1.1^{K/M} \tag{10}$$

where K is the count rate in cpm, M is the body weight of the individual in kilograms, and D is the first collision neutron dose in rad. Immediately after an accident there is a 50% margin of error due to ^{38}Cl, but after 4 hours the error margin reduces to 1%. Typical sensitivity is estimated to be about 1 rad in a standard man. Be cautioned that external contamination on the body or clothing may confound the results, and should be checked for.

In general, contamination is a secondary concern when acute injuries are involved. Therefore, always make sure that the victim's medical concerns are dealt with first. Contamination control can be re-established after rescue operations are concluded, if necessary.

Once the potential victims have been identified, then further work can be conducted to improve the neutron dose estimates. This could include measuring the quantities of ^{24}Na in the blood, ^{32}P in the hair, and activation of metals in jewelry, belt buckles, and other personal effects on the individual's body at the time of the exposure. The formula for converting ^{24}Na activity in the blood is

$$D = 1.6x10^{5} Ce^{0.0462t} \tag{11}$$

where ***D*** is the neutron dose in rad, ***C*** is the concentration of ^{24}Na in the blood in μCi/cc at time ***t***, and ***t*** is the time in hours between the accident and the analysis.

If the victim was wearing a dosimeter, then obviously that should be processed as soon as possible for the gamma dose. If not, then evaluation of the gamma dose may be more complicated. Some recent work[K] has developed techniques for rapidly determining doses from the kinetics of lymphocyte depletion in the victim's blood within the first 8 hours of the exposure. However, this work was based on accidents involving whole body gamma irradiation only, and it is not clear whether the same technique will work for neutrons or whether it will allow a method for separating the two components.

Evaluating the Neutron Spectrum

Once initial screening, triage, and recovery have been completed, then efforts can be started to attempt to determine the neutron spectrum. Hopefully, the facility will have Nuclear Accident Dosimetry (NAD) in place. These are dosimetry packages containing a variety of activation foils and filters for unfolding the neutron spectrum from the measured activation. This will provide good information regarding the general fields. However, it is important to try to determine the exact field the victims were in, since it can vary widely with location. Ideally, the individual would be wearing a personal nuclear accident dosimeter (PNAD), but this may not be the case. Therefore, one should consider trying to analyze objects collected from the individuals for activation products that may help unfold the neutron spectrum.

Unfolding a neutron spectrum requires a significant understanding of the cross-sections for the various materials and reactions involved. By selecting the reactions properly, one can build up the spectrum by comparing the various responses to each other. This will require some expertise, and so a specialist in activation analysis should be tasked to conduct the effort. I will not try to provide any guidance on this, as it is beyond the scope of this work.

Caring for the Victims

Once the victims have been turned over to medical personnel, the health physicist should provide any support necessary, as would be done with any other radiation-related injury. It is beyond the scope of this document to discuss the medical treatment further, although reference G provides a good basis for further study if desired.

Emergency Preparedness

This section will discuss in a general way the preparations that a facility should consider before conducting activities with significant quantities of fissionable material. First and foremost would obviously be to have qualified criticality safety personnel conduct an extensive evaluation of the operation for potential criticality safety concerns. This document will not address this, and please do not attempt to use this document as a substitute for such an analysis. It is only intended to provide a basic review of the considerations involved in criticality safety, and to provide the health physicist with an understanding of the radiological hazards involved when dealing with a criticality accident. Nevertheless, there are some basic considerations that should be addressed.

It would be expected that any facility dealing with significant quantities of fissionable material would have an emergency response program in place. However, there are aspects of criticality safety that should be included in these programs.

- Criticality alarm systems are usually installed in these facilities to detect such an event. ***Bear in mind that the alarm will not protect individuals from the initial pulse of radiation, but will only warn that one has occurred. Do not allow the personnel to come to rely on these systems as a protective measure!*** These alarms act as an aid in rapidly activating the emergency response, and should initiate an evacuation of the facility. Criticality alarms are usually specially designed systems in order to detect the rapid transients associated with an inadvertent criticality event which might be lost in the time constants of normal radiation monitors. To minimize false alarms and increase reliability, these systems will normally involve multiple detectors with logic systems that require signals from multiple detectors before signaling an alarm. Refer to ANS/ANS-8.3, ***Criticality Alarm Systems*** for the criteria for these systems.

- Alarms from criticality alarm systems are usually intended to be unique from other facility alarms, and should automatically trigger an evacuation of personnel in affected areas. Evacuation routes should be clearly marked, and should be designed to minimize the possibility of directing personnel through any high radiation areas that might be generated by the event. In other words, ***lead the personnel in a direction away from the location of the event as quickly as possible.*** This may well require limiting fissionable material operations to particular areas so that evacuation routes can be effectively planned ahead of time. The personnel should be trained to be aware of the potential for continuing operation or recurring pulses from such an event, so that they realize the importance of evacuating the facility. Since the time between pulses is generally on the order of minutes, evacuations should be conducted in an orderly manner. Panic and "break-neck" speed is not helpful, and may lead to more serious injuries. Generally, acute doses are only incurred within the first few meters of the system.

- The emergency plan should include procedures for the rapid screening of personnel to identify potential victims, and equipment should be calibrated and pre-staged to facilitate this process. Since these events are very rare, the plan should include regular verification of the operability and availability of this equipment.

- Nuclear Accident Dosimetry (NAD) should be positioned in and around the facility in locations that will provide a representative sampling of the neutron radiation fields. They should also allow easy access and removal by emergency response personnel without overly exposing them to any residual radiation hazards. Procedures should also be prepared and equipment and personnel available to quickly evaluate the results of this dosimetry. This same consideration should also be applied to personnel NADs. ANSI N13.1 provides criteria for a nuclear accident dosimetry program. Also, IAEA Technical Report 211, ***Dosimetry for Criticality Accidents: A Manual***, 1982, provides a great deal of information regarding methods for the dosimetry of such accidents using physical and biological media.

- As has been emphasized previously, a super-prompt critical system can oscillate in power, producing multiple pulses and may achieve a steady-state power level. Therefore, response and recovery personnel should always take great caution before approaching such a system until its status is determined. ***After recovery of any victims, the state of the system is the most important question that must be answered by the emergency personnel.***

All aspects of the emergency program are expected to be drilled against annually. Refer to ANSI/ANS-8.19, ***Administrative Practices for Nuclear Criticality Safety*** for overall criteria for the management of a criticality safety program.

Conclusions

Criticality accidents are very rare, but have the potential to cause significant injury and possible death to involved personnel. They are complicated systems to fully evaluate in general, and require detailed information of the system and the facility in order to adequately predict results. Unfortunately, many of the necessary details will not be available until after the event occurs, which requires the criticality safety expert and the health physicist to depend on assumptions and estimations in order to prepare for such an event.

This document has demonstrated techniques that should allow the health physics practitioner to better understand the potential hazards involved, and to make basic estimates of the order-of-magnitude of the potential consequences. The influences of various factors have been discussed with reference to how they may affect the radiation fields generated, and techniques are demonstrated for how to evaluate those influences. While there are much more advanced tools and techniques that should be relied upon when dealing with a real operation, the simple techniques discussed here are designed to help the practitioners better understand the fundamental aspects of such systems, and the concepts that would be applied to their study.

Problems in Criticality Safety

1. Which of the following values of ***k***-effective does not represent a self-sustaining chain reaction?
   A. 1.000
   B. 1.0065
   C. 1.0005
   D. 0.9998

2. Above what reactivity level is a fissionable system considered to be uncontrollable by active control systems?
   A. Subcritical
   B. super-prompt critical
   C. super critical
   D. critical

3. Which of the following properties does not have an effect on the critical state of a fissionable system?
   A. criticality alarm system
   B. enrichment
   C. container geometry
   D. moderators

4. Which of the following best describes the initial hazard from a criticality accident?
   A. A steady-state emission of neutrons and gamma rays.
   B. An intense pulse of neutrons and gamma rays.
   C. A sustained release of radioactive material.
   D. A bright blue flash of light.

5. Which of the following does not provide protection from a criticality accident?
   A. Administrative procedures to control enrichment.
   B. Criticality safety analysis of a system.
   C. Criticality alarm systems.
   D. Criticality-safe geometry.

6. A criticality accident has occurred at your facility. Which technique would you use to identify potentially exposed personnel?
   A. Send all personnel dosimeters in for processing and evaluate the results.
   B. Interview personnel as to their location when the alarms sounded.
   C. Screen personnel with a Geiger counter reading of their abdominal area.
   D. Pull the facility's nuclear accident dosimeters and send in for processing.

7. What is the primary concern in responding to a criticality accident after recovery of potential victims?
   A. Assessing contamination levels in the area.
   B. Securing ventilation to reduce environmental releases.
   C. Establish a mechanism for cooling the system.
   D. Determining the potential for re-achieving a critical state.

8. What is normally the primary contributor of gamma rays for a criticality accident?
   A. Prompt fission gamma rays.
   B. Early fission product decay gamma rays.
   C. Activation-product decay gamma rays from neutron capture in surrounding materials.
   D. Skyshine.

9. What is the main purpose of a criticality alarm system?
   A. To warn personnel of an inadvertent accumulation of fissile material.
   B. To warn personnel of an impending super-prompt critical pulse.
   C. To alert personnel that a criticality event has occurred.
   D. To monitor general radiation levels in a room.

10. Recently a criticality accident occurred at a fuel processing facility in Tokai-mura, Japan. The accident involved a solution of uranyl nitrate, enriched to 18.8% ^{235}U, in a precipitation tank. The tank was cylindrical with a diameter of 45 cm, and had a 5 cm thick cooling-water jacket on the outside of the tank. The estimated yield of the excursion was 2.5×10^{18} fissions, with roughly half of the yield from the initial pulse, and the rest from continuing steady-state operation of the system for about 18 hours.

    Assume the following:
    - The neutron leakage probability from the system was 20%;
    - assume an average neutron energy of 0.2 eV;
    - self-absorption of photons in the system can be ignored;
    - ignore all attenuation of photons and neutrons in the surrounding air;
    - meteorological conditions resulted in a X/Q of 9×10^{-4} at 100 meters from the release;
    - the room ventilation rate is 8 changes/hour;
    - ignore all intervening shielding;
    - use other data and equations as necessary from this module; and
    - if other assumptions are necessary, please identify and justify.

For an individual standing at a distance of 100 meters from the system, but in direct line-of-sight and directly downwind:

a) Estimate the neutron and gamma deep dose equivalents due to the direct radiation from the initial pulse;

b) Estimate the neutron and gamma deep dose equivalent rates due to the direct radiation from the steady-state operation of the system;

c) Estimate the total neutron and gamma dose equivalents due to the direct radiation, for the duration of the event;

d) Estimate the total deep dose equivalent to the same individual from the passage of a plume of radioactive gases and halogens from the system, using the table below, for the duration of the event;

e) Discuss any impact on the results obtained from the assumptions listed above; and

f) Discuss what implications these results would have on the design of a building within which a similar process will take place (note – building design, not process design).

Fission Product Noble Gas and Halogen Data (see notes)

Isotope	**Fission Yield (Ci/fission)**	**Half-life**	**Immersion Dose Conversion Factor (mrem/yr)/(μCi/m^3)**
Kr-85m	1.5 e-17	4.4 hr	817
Kr- 87	9.9 e-17	76 m	4470
Kr-88	6.5 e-17	2.8 hr	11300
Kr-89	4.2 e-15	3.18 m	10200
I-131	8.7 e-19	8.05 d	1910
I-132	1.1 e-16	2.26 hr	11900
I-134	4.5 e-16	52 m	13800
I-135	4.7 e-17	6.68 hr	8340
Xe-135	3.6 e-17	9.14 hr	1250
Xe-135m	2.2 e-16	15.6 m	2150
Xe-137	4.9 e-15	3.9 m	955
Xe-138	1.3 e-15	17.5 m	6270

Notes:

1. Fission yield data from ***DOE Release Fractions Handbook***, DOE-HDBK-3010-94.
2. Dose Conversion Factors from ***External Dose-Rate Conversion Factors for Calculation of Dose to the Public***, DOE/EH-0070, 1988.
3. Note that this listing is not intended to be comprehensive, but was selected to include the most significant contributors in order to demonstrate the magnitude of the concern without overcomplicating the problem.

References and Notes

[A] ANSI/ANS-8.1-1983, American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside of Reactors; Reaffirmed, Nov. 30, 1988.

[B] Most any good nuclear engineering textbook will discuss the various forms of equations used for determining the neutron balance, the two most typical are the Boltzmann transport equation and the neutron diffusion equation.

[C] Most of the kinetics material is from Hetrick, D. L., Dynamics of Nuclear Reactors; The University of Chicago Press, 1971. Other nuclear engineering texts also contain derivations and variations on the same theme.

[D] The treatment of gamma radiation characteristics is based on Schaeffer, N.M. (editor), Reactor Shielding for Nuclear Engineers, U.S. Atomic Energy Commission Office of Information Services; TID-25951, 1973. Similar information can be found in other nuclear engineering texts.

[E] Cember, H.; Introduction to Health Physics; McGraw-Hill, Inc.; 1992.

[F] Discussions of previous criticality accidents can be found in:

- Knief, R.; Nuclear Criticality Safety: Theory and Practice; American Nuclear Society, 1996.
- Vargo, G.J.; ***A Brief History of Nuclear Criticality Accidents in Russia – 1953-1997***; April 1999; PNNL-12199.
- Stratton, W. R.; ***A Review of Criticality Accidents***; Los Alamos National Laboratory, LA-3611; 1967. (There may be an updated version available)
- Horan, J. R. and Gammill, W. P.; ***The Health Physics Aspects of the SL-1 Accident***; **Health Physics Journal**; Vol. 9, pp. 177-186; 1963.
- Union Carbide Nuclear Company; ***Accidental Radiation Excursion at the Y-12 Plant***; Report # Y-1234; Aug. 4, 1958.

And various other documents.

[G] Mettler, F. A., C. A. Kelsey, and R. C. Ricks; Medical Management of Radiation Accidents; CRC Press; 1990.

[H] Broadhead, B.L., C. M. Hopper, R. L. Childs, and J. S. Tang; ***An Updated Nuclear Criticality Slide Rule***, (Vol.1 is the technical basis, Vol. 2 is the functional slide rule), NUREG/CR-6504 (aka ORNL/TM-13322); prepared by ORNL for the NRC, 1997.

[I] ***Report of ANS-6.1 Committee on Shielding Standards***, June 1974.

[J] Courtney, J. C.; ***A Handbook of Radiation Shielding Data***; ANS/SD-76/14; sponsored by the Nuclear Science Center, Louisiana State University, and the Shielding and Dosimetry Division, American Nuclear Society; July 1976.

[K] Goans, R. E., E. C. Holloway, M. E. Berger, and R. C. Ricks; ***Early Dose Assessment Following Sever Radiation Accidents***; Health Physics Journal; Vol. 72, Number 4; April 1997.