$$3\times10^{15}\ \ fissions\left(\frac{0.20\ Gy}{0.561\ Gy}\right)\left(\frac{2.0\ m}{1.8\ m}\right)^{2}=1.3\times10^{15}\ \ fissions$$

A comparison of this estimate along with computed dose ratio estimates is shown in Table D-2. The computed dose ratio estimates indicate that the 0.20 Gy total in air would consist of 0.146 Gy from neutrons and 0.054 Gy from gamma rays, slightly different than the estimate based on tissue dose. In contrast to moderated systems, the radiation field at 2 metres from an unmoderated system is dominated by neutrons that have significantly different energy deposition properties in air than in tissue.

Table D-2: Unmoderated Assembly n/γ Dose Ratio Comparison

Estimated source	n/γ Dose Ratio	Yield for 0.20 Gy
Los Alamos Accidents (Tissue)	12.0	1.3×10^{15} fissions
Computed Tissue Dose	14.5	2.2×10^{15} fissions
Computed Air Dose	2.7	5.7×10^{15} fissions

D.2.2.3 Summary of n/γ Ratios for Moderated and Unmoderated Systems

The estimates presented above are for highly-enriched low-concentration solution systems, and highly enriched metal systems. Clearly, as the transition from low concentration to metal is made, the n/γ dose ratio must also change as does the total number of fissions required, although in the latter case the relationship between the two is weakly coupled. These items should be considered by an evaluator along with any other effects caused by differences, such as fissionable material type or enrichment, for those facilities where such accidents are deemed credible.

D.3 Methods

Determining the adequacy of detector coverage is inherently a complicated process. Several options are available to the evaluator, including but not limited to: *in situ* source testing; simple hand calculations; one-dimensional deterministic or Monte Carlo transport computations; and two- or three-dimensional deterministic or Monte Carlo transport computations.

D.3.1 *In situ* source testing

In lieu of, or as a supplement to, computations, a fixed neutron or gamma ray source may be used to estimate the adequacy of detector coverage for a facility. This would simply amount to placing a source at a hypothetical criticality accident location and observing the detectors' response. Factors that should be considered are:

1. Source strength. The source should be strong enough to provide an adequate signal, i.e., statistically significant, at the detector through the intervening shielding and distance.

2. Source scaling. A relationship between the source strength and the type of accident under investigation must be known. The scaling may be as simple as a constant for a source calibrated in Gy/h, or more complicated, as in the case of fission based sources.
3. Source spectrum. The source must either represent the spectral characteristics of credible accidents, or some type of estimated correction should be made to account for spectral differences. For example, the gamma rays from a monoenergetic cobalt-60 source attenuates differently than the gamma rays from an accident, despite the fact that both could produce the same dose rate at 2 metres.

The advantage of *in situ* source testing is that it is a physical measurement, relieving the evaluator of the need to estimate locations, thicknesses, and compositions of the intervening materials. For some situations, however, the source strength required could be so high as to be impractical.

D.3.2 Simple hand calculations

For cases in which little or no shielding exists, it may be possible to apply a simple hand calculation to estimate the range of a detector. Use of this type of calculation is best illustrated by example.

Example D-1: A Gamma Ray Rate Meter Detector
Given:

1. The system must respond to the minimum accident of concern due to a sustained fission reaction in moderated, unreflected fissile material delivering 0.0033 Gy/s.
2. The system must respond to the minimum accident of concern due to a rapid transient in an unreflected fissile system, moderated or unmoderated. A 1 ms minimum duration of the rapid transient may be assumed.
3. The detector is set to trip at a gamma ray dose rate of 0.0005 Gy/h.

Assumptions:

1. the n/γ dose ratio in air for a moderated system is 0.11 (see Table D-1), so that at 2 metres, 0.18 Gy is due to gamma rays and 0.02 Gy is due to neutrons
2. the n/γ dose ratio in air for an unmoderated system is 2.7 (see Table D-2), so that at 2 metres, 0.054 Gy is due to gamma rays, and 0.146 Gy is due to neutrons
3. the indicated detector response (needle movement) to a rapid transient is assumed to be at least 1/2500 of the actual peak dose rate
4. the gamma ray dose rate varies inversely as the square of the distance from the 2 metre point
5. an air transmission factor of 1/3 is assumed at large distances

Assumptions (4) and (5) are equivalent to decoupling the problem into two parts: transport through a vacuum, and a constant attenuation factor of 2/3 to account for absorption in the air.

Sustained reaction calculations

The dose rate as a function of distance is given by:

$$D_r(r) = D_{2m} \times \left(\frac{2}{r}\right)^2 \times t_{air}$$

where:

D_r	=	the dose rate at r metres
D_{2m}	=	the dose rate at 2 metres
t_{air}	=	the transmission factor for air

Rearranging and using the values cited above gives:

$$r = (2\ meters) \times \sqrt{\frac{(0.180\ Gy/\min)(60\ \min/h)}{(0.0005\ Gy/h) \times 3}} = 170\ metres$$

as the effective radius of coverage for a moderated system with a sustained reaction.

Rapid transient calculations

The indicated detector response as a function of distance from a rapid transient is given by:

$$D_r(r) = D_{2m} \times \left(\frac{2}{r}\right)^2 \times t_{air} \times \varepsilon$$

where:

D_r	=	the dose rate at r metres
D_{2m}	=	the dose rate at 2 metres
t_{air}	=	the transmission factor for air
ε	=	the assumed detector response to the fast transient

Rearranging and using the values cited above gives:

$$r = (2\ meters) \times \sqrt{\frac{(0.180\ Gy/\min)(3.6 \times 10^6\ ms/h)}{(0.0005\ Gy/h) \times 3 \times 2500}} = 831\ metres$$

as the effective radius of coverage for a moderated system, and:

$$r = (2\ meters) \times \sqrt{\frac{(0.054\ Gy/\min)(3.6 \times 10^6\ ms/h)}{(0.0005\ Gy/h) \times 3 \times 2500}} = 455\ metres$$

as the effective radius of coverage for an unmoderated system.

Clearly, the limiting case in the example above is the sustained reaction. Figure D-1 shows a plot of the gamma ray dose rate versus distance for this case as well as the unmoderated rapid transient, as determined by this method.

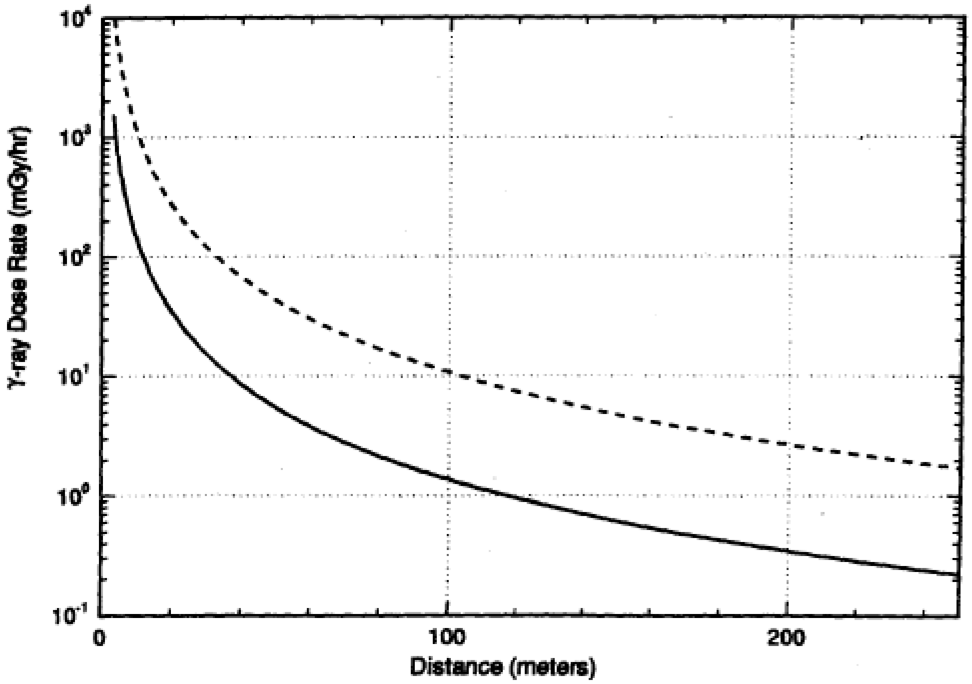

Figure D-1: Gamma Ray Dose Rate versus Distance, Based on a Total Dose of 0.20 Gy at 2 Metres
---------- Total dose delivered in 1 millisecond, unmoderated rapid transient criticality
………. Total dose delivered in 1 minute, moderated sustained criticality

The appeal of this method is its simplicity. However, it is noted that there are items that should be considered by an evaluator using this technique. First of all, the validity of the transmission factor of 1/3 for air should be defended, perhaps with the aid of experimental results or attenuation arguments. Secondly, technical justification for ignoring other intervening materials should be provided.

Note that the preceding example should not serve as the sole technical basis for the adequacy of detector placement for any facility. Additional technical justification of the technique and consideration of facility-specific conditions are warranted.

D.3.3 One-dimensional deterministic or Monte Carlo transport computations

In cases where simple hand calculations are insufficient, more detailed one-dimensional models can be constructed through the use of either deterministic or Monte Carlo computer codes.

The advantage of using a computer code is that spectral effects, absorption, scattering, and distance attenuation are all taken into account simultaneously. In addition, the method is not limited by the number of intervening materials, although it does require estimates of the amount and composition of those materials. Results can be generated that are either generic or detector-specific.

Generic results are a set of guidelines of allowed distances and shielding (types and amounts) between a hypothetical accident location and a detector's position. These criteria could then be applied to a facility floor plan to demonstrate coverage. For detector-specific results, a separate computation can be performed for each detector and each hypothetical accident location, with estimates of distance and amounts of intervening shielding being scenario-specific.

A one-dimensional example

In this example, a computer code with an appropriate neutron/gamma ray coupled cross section library containing air dose flux conversion factors was used to demonstrate how generic criteria might be generated. Figure D-2 shows a geometric representation of the computational model.

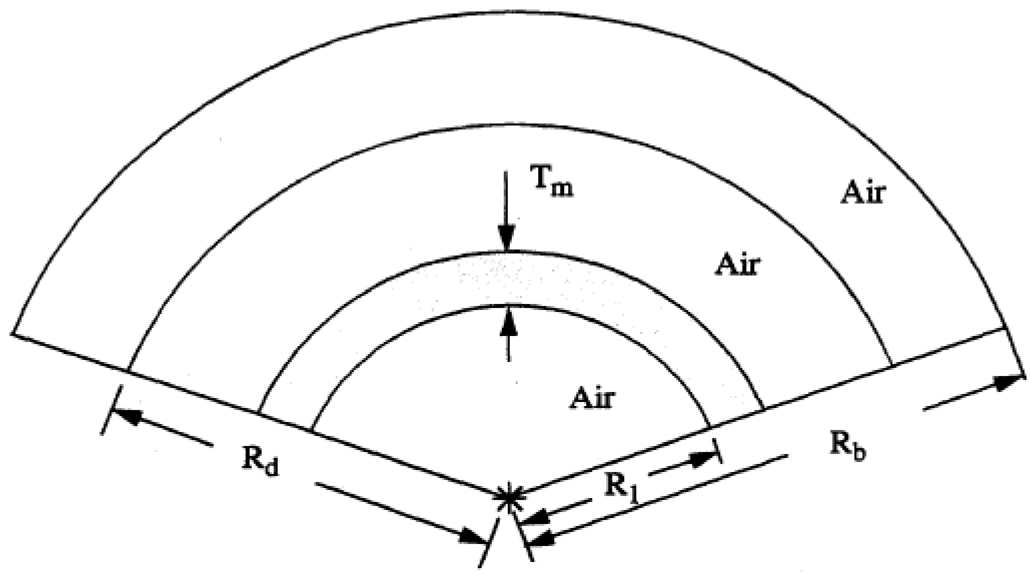

Figure D-2: Example of a One-Dimensional Computational Model

Where: * is the source location;

R_1 is the distance from the source to the beginning of any intervening material other than air in the model (R_1>2 metres);

R_d is the distance from the source to the detector;

R_b is the distance from the source to the boundary of the problem (R_b>R_d); and

T_m is the sum of the thicknesses of all the intervening material other than air

In this example:

1. The source term is equivalent in strength and spectrum to the neutron and gamma ray leakage from a 25 g/L Pu (95/5) solution criticality accident scaled to produce 0.20 Gy total at 2 metres with no other materials present.
2. The detector distance, R_d, is set at 50 metres.
3. The initial boundary for intervening materials, R_1, is set at 25 metres.
4. The problem boundary, R_b, is placed at 150 metres.

   (In this type of calculations, it is important to set R_b at a large distance beyond any location at which results are desired. Failure to do so may result in underprediction of the dose at the detector due to underestimation of “sky shine” effects).
5. The only type of intervening material considered was concrete.

Table D-3 shows a summary of the integrated computational results for this example.

Table D-3: Integrated Quantities for a 25 g/l Pu (95/5) Solution Criticality

	Integrated Quantities at 2 metres				
	air γ-dose (Gy)	**air n-dose (Gy)**	**Φ_n (n-cm/cm^3)**	**Φ_T (n-cm/cm^3)***	**Φ_γ (n-cm/cm^3)**
	0.180	0.02	5.5×10^{9}	6.4×10^{8}	3.9×10^{10}
Thickness of Concrete (cm)	**Integrated Quantities at 50 metres**				
	air γ-dose (Gy)	**air n-dose (Gy)**	**Φ_n (n-cm/cm^3)**	**Φ_T (n-cm/cm^3)***	**Φ_γ (n-cm/cm^3)**
0.0	2.7×10^{-4}	4.3×10^{-5}	1.8×10^{7}	2.6×10^{6}	9.3×10^{7}
10.0	1.5×10^{-4}	3.3×10^{-5}	1.5×10^{7}	3.1×10^{6}	5.2×10^{7}
20.0	8.0×10^{-5}	2.0×10^{-5}	8.2×10^{6}	2.4×10^{6}	2.5×10^{7}
30.0	4.3×10^{-5}	1.0×10^{-5}	4.0×10^{6}	1.4×10^{6}	1.2×10^{7}
60.0	6.8×10^{-6}	1.0×10^{-6}	3.9×10^{5}	1.6×10^{5}	1.7×10^{6}

* Φ_T is the thermal neutron fluence, E < 0.4 eV.

Usage of these results is specific to the detector type and dynamics of the accident. Two examples of application of the results follow.

Application 1: A thermal neutron fluence detector

Given:

1. The system must respond to the minimum accident of concern due to a sustained fission reaction in moderated, unreflected fissile material delivering 0.0033 Gy/s.
2. The system must respond to the minimum accident of concern due to a rapid transient in an unreflected fissile system, moderated or unmoderated. A 1 ms minimum duration of the rapid transient may be assumed.
3. The detector is set to trip if 16 counts are received within the cycle time of 1 s. For this detector, 16 counts correspond to a thermal neutron fluence of 500 n-cm/cm^3.

The number of counts received by the detector during its cycle time is given by:

$$C_D = \frac{\phi_T}{\tau_P} \times MIN[\tau_C, \tau_P] \times FTCC$$

where:

C_D	=	the number of counts
Φ_T	=	the thermal neutron fluence
τ_p	=	the pulse width duration
τ_c	=	the cycle time of the integrating detector
$MIN[\tau_c,\tau_p]$	=	use the smaller of these two time frames in determining the total counts

FTCC = the fluence-to-counts conversion factor

For 30 cm of concrete, the number of counts received at the detector from a sustained reaction would be:

$$\frac{1.4\times10^{6}\, n-cm/cm^{3}}{60\ s}\times 1\sec\times\frac{16\ counts}{500n-cm/cm^{3}}=747\ counts$$

and from a rapid transient of duration 1 millisecond:

$$\frac{1.4\times10^{6}\, n-cm/cm^{3}}{1\ ms}\times 1\ m\sec\times\frac{16\ counts}{500n-cm/cm^{3}}=44{,}800\ counts$$

Table D-4 shows the results for all thicknesses of concrete included in Table D-3 for this type of detector.

Table D-4: Thermal Neutron Fluence Detector Response

Thickness of Concrete (cm)	Counts Received During the Cycle Time at 50 metres	
	Sustained Reaction	1 ms Transient
0.0	1,387	83,200
10.0	1,653	99,200
20.0	1,280	76,800
30.0	747	44,800
60.0	85	5,120

Given the conditions of this example, it is now possible to generate position criteria to be applied to a facility floor plan; for example, “A detector must be within 50 metres of a credible location for the minimum accident of concern, with no more than 60 cm of intervening concrete, for the detector to respond to the accident by setting off the alarm.”

Note: This example is for illustrative purposes only. The data of Table D-3 and the sample application for a thermal neutron detector system should not be used as a technical basis for the adequacy of detector placement at any specific facility.

Application 2: A gamma ray dose rate meter

Given:

1. The system must respond to the minimum accident of concern due to a sustained fission reaction in moderated, unreflected fissile material delivering 0.0033 Gy/s.
2. The system must respond to the minimum accident of concern due to a rapid transient in an unreflected fissile system, moderated or unmoderated. A 1 ms minimum duration of the rapid transient may be assumed.
3. The detector is set to trip at a gamma ray dose rate of 0.0005 Gy/h.
4. The indicated detector response (needle movement) to a rapid transient is assumed to be at least 1/2500 of the actual peak dose rate.

The dose rate (D_r) at the detector is given by:

$$D_r = \frac{d_{50m}(\gamma)}{\tau_P} \times \varepsilon$$

where:

$d_{50m}(\gamma)$	=	the total gamma ray dose at 50 metres
τ_p	=	the pulse width duration
ε	=	the assumed detector response to a fast transient (note: for sustained fission reactions, $\varepsilon = 1.0$)

For the unshielded condition (concrete thickness of zero), the gamma ray dose rate at the detector from a sustained reaction would be:

$$D_r = \frac{2.7 \times 10^{-4}\ Gy}{60\ s} = 4.5 \times 10^{-6}\ Gy/s = 16.2\ mGy/h$$

For a rapid transient of duration 1 millisecond:

$$D_r = \frac{2.7 \times 10^{-4}\ Gy}{1\ ms} \times \frac{1}{2500} = 1.1 \times 10^{-4}\ Gy/s = 388.8\ mGy/h$$

Table D-5 shows the results for all thicknesses of concrete of Table D-3 for this type of detector.

Table D-5: Gamma Ray Dose Rate Detector Response

Thickness of Concrete (cm)	Gamma Ray Dose Rate at 50 metres (mGy/h)	
	Sustained Reaction	1 ms Transient
0.0	16.20	388.80
10.0	9.00	216.00
20.0	4.80	115.20
30.0	2.58	61.92
60.0	0.41	9.80

Given the conditions of this example, it is clear that a detector 50 metres from the location of the minimum accident of concern with 30 cm of intervening concrete would respond to the accident by setting off the alarm. With more than 30 cm of concrete, an alarm would not be guaranteed by the results shown, and more detailed computations would be required for thicknesses between 30 and 60 cm.

Note: This example is for illustrative purposes only. The data of Table D-3 and the sample application for a gamma ray detector should not be used as a technical basis for the adequacy of detector placement at any specific facility.

D.3.4 Two- and three-dimensional deterministic or Monte Carlo transport computations

More detailed facility- and accident-scenario-specific models can be constructed through the use of two- or three- dimensional computer codes, if judged to be necessary and cost-effective with respect to other options, e.g., adding more detectors. In addition to all of the advantages associated with the one-dimensional modeling, a three-dimensional model includes contributions from floor, ceiling, and possibly sky scatter, as well as a more detailed representation of streaming and scatter from room and corridor walls. Two-dimensional models are often adequate for situations where radial symmetry may be assumed.

In contrast to one-dimensional modeling, however, there are several aspects of two- or three-dimensional modeling that should be acknowledged by an evaluator prior to the use of this option. These include, but are not necessarily limited to, the following:

1. The modeling effort is more difficult and therefore more time consuming and error prone.
2. In the case of Monte Carlo calculations, it may be necessary to make heavy use of variance reduction techniques, such as source biasing and regional probability weighting, to obtain a sufficiently accurate and timely answer. Such biasing is problem-specific, and therefore, separate cases would have to be executed for each detector and accident scenario of concern. Furthermore, for results of this type (i.e., location-specific and energy-dependent, in which very few of the total sampled population may contribute), it is quite possible to obtain completely erroneous results despite low estimated standard deviations. Additional measures, such as multiple runs with different random number sequences, or more in-depth statistical analysis of the results (e.g., variance of the variance), should be performed to reduce this possibility.
3. In the case of deterministic calculations, the amount and detail of meshing required could be computationally prohibitive. Furthermore, any deterministic method relying on a discretized angular mesh (without some type of mitigation such as a first-collision source) is subject to ray effects that can lead to questionable or incorrect results.

APPENDIX E
Fuel Unit Handling, Storage, and Transportation—Criticality Safety Considerations

Fuel design parameters, storage array dimensions, fuel handling procedures, and moderation and reflection conditions are selected in performing nuclear criticality safety evaluations (NCSEs) to assure consideration of the most reactive credible conditions. Section 11, *Criticality Safety Criteria for the Handling, Transportation, Storage, and Long-term Waste Management of Reactor Fuel Outside Reactors*, section 11.3, *General Safety Criteria*, requires that consideration be given to normal and credible abnormal conditions and to related uncertainties, including design tolerances, associated with controlled parameters. Representative parameters and conditions are listed below.

E.1 Fuel rod parameters

Fuel rod parameters include:

1. fissile material content, form, density, nuclear properties, and distribution
2. burnable poison content, density, and distribution

   Caution: The reactivity of irradiated fuel containing burnable poisons may exceed that of unirradiated fuel.
3. fuel rod geometry including cladding material and thickness
4. other materials within the fuel rod that may affect reactivity

E.2 Fuel unit configuration

Fuel unit configuration considerations include:

1. number of fuel rods and their location within a fuel unit
2. dimensions of each fuel unit
3. other materials or rods that may be present

E.3 Array parameters

Array parameters include:

1. spacing of fuel units
2. fixed neutron absorbers between fuel units
3. materials of construction within the array (nuclear properties, quantities, location, and dimensions)
4. fuel handling during loading and unloading operations

E.4 Moderator conditions

Moderator conditions include:

1. credible conditions of moderation within and between fuel units; for example:
    - inclusion of plastic shims or other moderating material (fog, snow, mist, or personnel) for dry storage of fuel units
    - water density and temperature including consideration of void formation by boiling for storage of fuel units under water

E.5 Reflector and interaction conditions

Reflector and interaction conditions include:

1. reflector composition, configuration and location
2. interaction with other fissile material

APPENDIX F
Moderators and Moderating Materials

F.1 Typical moderating materials

Many materials routinely encountered in nuclear facilities can be neutron moderators. Some of these materials may be more effective moderators than water. The following list, while not complete, is intended to promote the consideration of the possible moderating properties of materials that could be encountered.

Alcohol	Hydrides
Ammonium or other hydrated radicals	Hydrocarbons and other organic materials
Antifreeze	Lubricants
Benelex	Oils
Beryllium	Paint
Biological materials	Paper and paper products
Butvar ®	Paraffin
Carbon (e.g., graphite, charcoal)	Partially halogenated organics
Cane fiber board (Cellotex ®)	People *
Cleaning agents	Plastic (containers, bags, sheets, etc.)
Concrete	Plexiglas, Lucite ®, etc.
Construction materials	Polyethylene
Deuterium compounds	Rags
Environmental or atmospheric moisture	Resins
Fire suppressants	Shielding materials
Fuel pellet binders and pore-formers	Solvents
Gasoline, kerosene	Sponges
Gloves	Stabilizers
Hands *	Water
Heavy water	Moist sand or soils
Hydraulic fluid	Wood and wood products

* The material content of the human body can provide significant moderating capability.

F.2 Potential sources of moderators

This appendix lists types of moderating materials commonly found in nuclear facilities. These materials could be introduced either by design or inadvertently from sources such as those examples listed below.

1. Service lines:
   - steam water
   - fire sprinkler lines
   - roof drains
   - floor drains
   - process/instrument air lines
2. Connections to fissile material operations:
   - instrument lines
   - processing lines
   - ventilation ducts
   - electrical conduit
   - vent lines
   - heating and cooling systems
3. Equipment
   - gloveboxes and fume hoods
   - hydraulic systems
   - heating and cooling lines
   - HEPA filters
   - buckets and containers
   - lubrication systems
   - criticality safety drains and overflows
4. Construction materials:
   - Room-Temperature Vulcanizing (RTV) silicone sealant
   - epoxy
5. Maintenance and modification activities
6. Decontamination materials:
   - cleaning agents
   - rags and paper towels
   - sponges
7. Environment:
   - atmospheric moisture
   - precipitation (such as rain and snow)
   - water films

8. Process chemicals or additives:
   - binders and pore-formers
   - feed streams
   - solvents
   - holdup of moderators from process operations
9. Accident and emergency response sources:
   - fire sprinkler lines
   - fire hoses
   - flooding
   - type A fire extinguisher
10. Human intervention:
    - fire fighting introduction of moderator
    - introduction of unapproved moderators
    - mop water
    - personnel presence

F.3 Moderator content measurements

The considerations in this appendix are intended to provide assurance of the integrity of the measurement and process controls.

1. Appropriate procedures include:
   - precautions needed during preparation and analysis of samples
   - operational maintenance requirements for the measurement equipment
   - configuration requirements for instrumentation
   - verification requirements
2. Sampling methodology provides representative samples for analysis. The integrity of each sample is maintained throughout the sampling and analysis process.
3. Consideration is given to analysis of at least two samples by independent analytical techniques. However, where reliance is based on a single analytical technique, the samples are analyzed by independent instrumentation.
4. Independent moderator measurements agree within a specified confidence level.
5. Appropriate control standards are used to verify that the attributes of each analytical technique are in conformance with applicable qualification plans.
6. Control standards are used to demonstrate acceptable results after system maintenance and are measured periodically prior to, and after, measurement of individual or groups of samples. Requirements are established for control of standards.
7. Analytical techniques are qualified by identifying the bias, uncertainty, and minimum and maximum moderator detection limits at a specified confidence level.
8. Continuous process monitoring techniques are used to demonstrate that process systems reliably produce material within the required moderator limits.

F.4 Examples of engineered barriers to control moderators

Engineered barriers can be used as a means to control the introduction of moderators. Typical barriers include:

- secondary roofs
- false ceilings (drop ceilings)
- secondary walls
- vapour barriers
- raised floors or structures
- normally closed apertures
- seals
- syphon breaks
- backflow prevention devices
- condensate traps
- double block and drain (bleed)
- double block and blank
- containers
- gloveboxes
- equipment (air dryers)
- instrumented and controlled systems (dew point indicators, neutron interrogation)

APPENDIX G
Example of a Partial Description of a Nuclear Criticality Safety Program for a Fuel Storage Facility

G.1 Template of contents of nuclear criticality safety program

This example is not a prescribed format and content guide for a nuclear criticality safety program. It presents one of a number of acceptable ways to satisfy the information requirements (items 1 and 2 in section 12.8.2). The main purposes of this example are:

- to identify a list of the sections of regulatory document RD-327, *Nuclear Criticality Safety* [1] that apply to a specific facility
- to highlight that a nuclear criticality safety program should contain exact text quoted directly from the applicable standards, guidelines, and CNSC requirements

In this example, the fuel storage facility is assumed to be a new facility; therefore, the example encompasses the full spectrum of activities necessary to establish a nuclear criticality safety program, including design, analysis, alarm systems, emergency response, training, management responsibilities, and administrative practices. For each activity, the example gives a partial list of relevant requirements. The complete list is to be created by taking into account the profile of the facility.

G.2 Identifying the requirements

This section presents one acceptable method of satisfying the information requirements (item 1 in section 12.8.2).

Sample text:

The facility is committed to the following subset of regulatory document RD-327, Nuclear Criticality Safety as appropriate to the needs of the facility:

1. *Section 2, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors;*
2. *Section 3, Criticality Accident Alarm System;*
3. *Section 6, Nuclear Criticality Safety in the Storage of Fissile Materials;*
4. *Section 11, Criticality Safety Criteria for the Handling, Transportation, Storage, and Long-term Waste Management of Reactor Fuel Outside Reactors;*
5. *Section 12, Administrative Practices for Nuclear Criticality Safety;*
6. *Section 13, Nuclear Criticality Safety Training;*
7. *Section 16, Nuclear Criticality Accident Emergency Planning and Response.*

This subset of regulatory document RD-327 captures all requirements that are relevant to the proposed operations at the facility. Some sections of regulatory document RD-327 have been excluded, for example, those related to the handling of powders or solutions.

The facility is committed to the following CNSC requirements:

- *Administrative margin of subcriticality is 50 mk in keff or, where appropriate, 20% of the critical mass;*
- *Semi-quantitative method will be used to demonstrate that the margin of subcriticality is not violated under normal and credible abnormal conditions (accidents or accident sequences) that have frequency of occurrence equal to or more than 10-6 per year; and*
- *The shielding and confinement system of the facility will be designed and operated such that the dose, resulting from exposure to direct radiation and to radionuclides released from the facility following a criticality accident, does not violate criteria established by international standards (Reference X1, Annex III, Section III-2) and national guidance (Reference X2) as a trigger for a temporary public evacuation.*

This subset of regulatory document RD-327 encompasses the full spectrum of activities necessary to establish a Nuclear Criticality Safety Program, including design, analysis, alarm systems, emergency response, training, management responsibilities, and administrative practices.

Reference X1: Food and Agriculture Organization of the United Nations, International Atomic Energy Agency, International Labour Organisation, OECD Nuclear Energy Agency, Pan American Health Organization, United Nations Office for the Co-Ordination of Humanitarian Affairs, World Health Organization, Preparedness and Response for a Nuclear or Radiological Emergency, Safety Requirements, Safety Standards Series No. GS-R-2, IAEA, Vienna, Austria, 2002.

Reference X2: Health Canada, Canadian Guidelines for Intervention during a Nuclear Emergency, Document H46-2/03-326E, Ottawa, Ontario, November 2003.

G.3 Sample procedure for criticality accident sequence assessment

This section provides an example of criticality accident sequence assessment (CASA). It employs a semi-quantitative risk index method for assessing accident sequences in terms of their likelihood of occurrence.

The risk index method framework enables the applicant to identify which accidents or accident sequences exceed the likelihood level identified in section 2.3.2.2 and, therefore, require designation of criticality safety controls (CSC) (i.e., engineered and/or administrative CSC) and supporting management measures. Descriptions of these accident sequences need to be reported in the Safety Analysis Report (SAR).

This section works through an example of how the Risk Index Method can be applied to a uranium powder blender. It describes one method of evaluating compliance with the likelihood level identified in section 2.3.2.2. The method is intended to permit quantitative information to be considered, if available. Since likelihoods are inherently quantitative, evaluation of a particular accident should be consistent with any facts available, which may include quantitative information concerning the availability and reliability of CSC involved.

This section presents one method of analysis of credible accident sequences for either the nuclear criticality safety evaluation (NCSE) or the SAR. The method of this section describes semi-quantitative criteria for evaluating frequency indexes of criticality safety controls. These criteria for assigning indexes, particularly the descriptive criteria provided in some tables of this section, are intended to be examples, not universal criteria.

It is preferable that each applicant develops such criteria, based on the particular types of CSC and management measure programs. The applicant should modify and improve such criteria as insights are gained during performance of the CASA.

If the applicant evaluates accidents using a different method, the method should produce similar results in terms of the accident's likelihood. The method should be regarded as a screening method, not as a definitive method of proving the adequacy or inadequacy of the CSC for any particular accident. Because methods can rarely be universally valid, individual accidents for which this method does not appear applicable may be justified by an evaluation using other methods. The method does have the benefit that it evaluates, in a consistent manner, the characteristics of CSC used to limit accident sequences. This method permits identification of accident sequences with defects in the combination of CSC used. Such CSC can then be further evaluated or improved to establish adequacy. The procedure also ensures the consistent evaluation of similar CSC by different CASA teams. Sequences or CSC that have risk significance and are evaluated as marginally acceptable are good candidates for more detailed evaluation by the applicant and the reviewer.

The tabular accident summary resulting from the CASA should identify, for each sequence, what engineered or administrative CSC must fail to allow the likelihood that exceeds the levels identified in section 2.3.2.2. These requirements state that sequences of events leading to credible abnormal conditions shall be evaluated. The likelihood and possible consequences of such occurrences should be evaluated using reliable data and methodologies. The purpose of this section is to provide an example of an acceptable semi-quantitative method to perform such an evaluation.

The accident evaluation method described below does not preclude the need to comply with the double-contingency principle. Although exceptions are permitted with compensatory measures, double contingency protection should, in general, be applied. Double contingency protection is needed as there are usually insufficient firm data on the reliability of the CSC equipment and administrative CSC procedures used in criticality safety. If only one CSC were relied on to prevent a criticality, and it proved to be less reliable than expected, then the first time it failed, a criticality accident could result. For this reason, at least two independent CSC should be used. Inadequate CSC can then be determined by observing their failures without also suffering the consequences of a criticality accident. Even with double contingency protection, each CSC should be sufficiently unlikely to fail, so that if one of the two items that establish double contingency protection is actually ineffective, criticality is still extremely unlikely.

G.3.1 Assessing the effectiveness of the CSC

The risk of an accident sequence is reduced through the application of different numbers and types of CSC. By either reducing the likelihood of occurrence or by mitigating the consequences, CSC can reduce the overall resulting risk. The designation of CSC should generally be made to reduce the likelihood (i.e., prevent an accident), but the consequences may also be reduced by minimizing the potential hazards (i.e., mitigate the consequences) if practical. Based on hazards identification and accident sequence analyses for which the resulting unmitigated or uncontrolled risks are unacceptable, administrative and/or engineered CSC may be designated to reduce the likelihood of occurrence and/or mitigate severity of the consequences.

G.3.2 Risk score evaluation summary

As previously mentioned, an acceptable way for the applicant to present the results of the CASA is a tabular summary of the identified accident sequences. Table G-1 is an acceptable format for such a table. This table lists several example accident sequences for a powder blender at a typical facility.

Table G-1 summarizes two sets of information: (i) the accident sequences identified in the CASA; and (ii) a likelihood index, calculated for each sequence, to show compliance. The likelihood index calculation is summarized below.

Accident sequences result from initiating events, followed by failure of one or more CSC. Thus, in Table G-1, there are columns for the initiating event and for CSC. CSC may be mitigative or preventive. Mitigative CSCs are measures that reduce the consequences of an accident. The phrase “uncontrolled and/or unmitigated consequences” describes the results when the system of preventive CSC fails and mitigation also fails. Mitigated consequences result when the preventive CSC fail, but mitigative measures succeed. These are abbreviated in the table as “unmit.” and “mitig.”, respectively. Index numbers are assigned to initiating events, CSC failure events, and mitigation failure events, based on the reliability characteristics of these items.

With redundant CSC and in certain other cases, there are sequences in which an initiating event places the system in a vulnerable state. While the system is in this vulnerable state, a CSC must fail for the accident to result. Thus, the frequency of the accident depends on the frequency of the first event, the duration of vulnerability, and the frequency of the second CSC failure. For this reason, the duration of the vulnerable state should be considered, and a duration index assigned. The values of all index numbers for a sequence, depending on the number of events involved, are added to obtain a total likelihood index, T.

Note that, if all the failures in the accident sequence are independent, then summation of all index numbers is a valid approach for calculation of T. The following discussions and the example in the Table G-1 assume that the initiating event and all the CSC failures in the accident sequence are independent. However, if the independence is not demonstrated, then the dependent failures should be identified and accounted for. Examples of potential dependences are: common cause initiating events, intersystem dependences (such as functional dependences, shared-equipment dependences, physical dependences, and human-interaction dependences), and inter-component dependences.

The values of index numbers in accident sequences are assigned considering the criteria in Tables G-2 through G-4. Each table applies to a different type of event. Table G-1 applies to events that have frequencies of occurrence, such as initiating events and certain CSC failures. When failure probabilities are required for an event, Table G-3 provides the index values. Table G-4 provides index numbers for durations of failure. These are used in certain accident sequences where two CSCs must simultaneously be in a failed state. In this case, one of the two controlled parameters will fail first. It is then necessary to consider the duration that the system remains vulnerable to failure of the second. This period of vulnerability can be terminated in several ways. The first failure may be "fail-safe" or be continuously monitored, thus alerting the operator when it fails so that the system may be quickly placed in a safe state. Or, the CSC may be subject to periodic surveillance tests for hidden failures.

When hidden failures are possible, these surveillance intervals limit the duration that the system is in a vulnerable state. The reverse sequences, where the second CSC fails first, should be considered as a separate accident sequence. This is necessary because the failure frequency and the duration of outage of the first and the second CSC may differ. The values of these duration indexes are not judgmental; they are directly related to the time intervals used for surveillance and the time needed to render the system safe.

As shown in Table G-4, the duration of failure is accounted for in establishing the overall likelihood that an accident sequence will continue to the defined consequence. Thus, the time to discover and repair the failure is accounted for in establishing the risk of the postulated accident.

When the number is more negative, the failure is less likely; this applies to all of the index numbers. Accident sequences may consist of varying numbers of events, starting with an initiating event. The total likelihood index is the sum of the indexes for all the events in the sequence, including those for duration.

As shown in the first row of Table G-1, the failure duration index can make a large contribution to the total likelihood index. The reviewer should verify that there is adequate justification that the failure will be corrected in the time ascribed to the duration index. In general, duration indexes with values less than minus one (-1), corresponding to 36 days, should be based on intentional monitoring of the process. The duration of failure for an unmonitored process should be conservatively estimated.

Table G-1 provides two likelihood indexes for each accident sequence to permit evaluation of the risk significance of the CSC involved. To measure whether a CSC has high risk significance, the table provides an "uncontrolled index", determined by modeling the sequence with all CSC as failed (i.e., not contributing to a lower likelihood). In addition, a "controlled index" is also calculated, taking credit for the low likelihood and duration of CSC failures. When an accident sequence has an uncontrolled likelihood index exceeding (-2) but a controlled index of less than (-6), the CSCs involved have a high risk significance because they are relied on to achieve acceptable safety performance. In addition, use of two likelihood indexes allows demonstrating that no CSC has an exceedingly large contribution to risk. Thus, use of these indexes permits evaluation of the possible benefit of improving CSC and also whether a relaxation may be acceptable.

Table G-5 provides a more detailed description of the accident sequences used in the example of Table G-1. The reviewer needs the information in Table G-5 to understand the nature of the accident sequences listed in Table G-1. Table G-1 lacks room to explain any but the simplest failure events.

Table G-6 explains the CSC and external initiating events that appear in the accident sequences in Table G-1. The reviewer needs the information in Table G-6 to understand why the initiating events and CSC listed in Table G-1 have the low likelihood indexes assigned. Thus, Table G-6 should contain such information as (i) the margins to safety limits, (ii) the redundancy of a CSC, and (iii) the measures taken to ensure adequate reliability of a CSC. Table G-6 must also justify why external events, which are not obviously extremely unlikely, have the low likelihoods that are being relied on for safety. The applicant should provide separate tables to list the CSC for criticality, chemical, fire, radiological, and environmental accidents. If an applicant chooses to classify CSC by applying different levels or grades of quality assurance, then the applicant should also provide the appropriate quality assurance grade for the CSC.

G.3.3 Accident summary and likelihood index assignment for Table G-1

For each column in Table G-1, the following text provides the complete definition:

Accident Sequence / Identifier

This column identifies the accident sequence being analyzed. The CASA has all accident sequences for uniquely identified facility processes, referred to here as “nodes”. Symbols, names, or numbers of these nodes permit them to be uniquely identified. For example, the “blender hopper” node described in Table G-1 has the unique identifying symbol PPB2. PPB2-1 is the first accident sequence identified in that node. By reviewing sample accident sequences presented in the SAR and the selected accident sequences contained in the NCSE, the reviewer(s) can evaluate and confirm i) the adequacy of the CSC for preventing accidents, and ii) the bases for assigning the consequences and likelihoods in the table.

Initiating Event (a1) and Enabling Event (a2) (if applicable)

These columns list initiating events or CSC failures that are typically identified in the Process Hazard Analysis phase of the NCSE and that may lead to exceeding the levels identified in section 2.3.2.2.

Initiating events are of several distinct types: i) external events, such as hurricanes and earthquakes; ii) facility events external to the node being analyzed (e.g., fires, explosions, failures of other equipment, flooding from facility water sources); iii) deviations from normal operations of the process in the node; and iv) failures of CSC of the node. The tabulated initiating events should only consist of those that involve an actual or threatened failure of CSC or that cause a demand requiring CSC to function to prevent exceeding USL.

The frequency index number for initiating events is referred to in the table by the symbol “frqi.” Table G-2 provides criteria for assigning a value to frqi. Usually, there is insufficient room in a tabular presentation like Table G-1 to describe events accurately. Consequently, the applicant should provide supplementary narrative information to adequately describe each general type of accident sequence in Table G-1. Cross-referencing between this information and the table should be adequate (e.g., the unique symbolic accident sequence identifiers can be used). Table G-5 is an example of a list of supplementary accident sequence descriptions corresponding to Table G-1.

Preventive Safety Parameter 1 or CSC 1 Failure/Success (b)

This column addresses the failure or success of the safety parameter designated to prevent exceeding USL. Specific CSCs that may be needed to maintain the safety parameter should be included in this table. If separate parameters or CSCs are used to prevent different consequences, separate rows in the table should be defined corresponding to each type of consequence.

Table G-1 contains an example of a set of related sequences so separated. Accident sequences where two CSCs must simultaneously be in a failed state require assignment of three index numbers: i) the failure frequency of the first CSC, frq1; ii) the duration of this failure, dur1; and iii) the failure frequency of the second CSC, frq2. For such accident sequences, the initiating event is failure of the first CSC. In these cases, frq1 is assigned using Table G-2. The failure duration of the first CSC is assigned using Table G-4.

Other accident sequences may be more easily described as a failure of the CSC on demand after the occurrence of an initiating event. In these cases, the failure probability index number, prf1, is assigned using Table G-3.

Preventive Safety Parameter 2 or CSC 2 Failure/Success (c)

This column is provided in case a second preventive CSC is designated. The failure frequency or failure probability on demand is assigned in the same manner as for preventive CSC 1.

Preventive Safety Parameters or CSC Failure/Success (d1, d2…)

This column is provided in case other preventive CSC is designated. The failure frequency or failure probability on demand is assigned in the same manner as for preventive CSC 1.

Likelihood Index / Risk Score* T uncontrolled/ controlled (e)

This column lists the total likelihood index / risk score for an accident sequence. The total likelihood index, T, is the sum of the indices for those events that comprise an accident sequence which normally consists of the initiating event and failure of one or more CSC, including any failure duration indices. However, accident sequences may consist of varying numbers and types of undesired events. Methods for deciding what frequencies and failure durations need to be considered are described later in this appendix. Determination of the likelihood index for an accident sequence as the sum of the indices is valid if all the failures in the accident sequence are independent.

Consequence Evaluation Reference

This column permits identification of the consequence calculations that relate to this accident sequence. Multiple references may be required to refer to calculations of the different types of consequences (e.g., radiological, chemical, etc.).

Comments and Recommendations

This column records NCSE team recommendations. It is especially useful when the existing system of CSC is evaluated as being deficient. This may happen because a newly identified accident sequence is not addressed by existing CSC or because an unacceptable performance deficiency has been found in the existing CSC.

G.3.4 Determination of failure frequency index numbers in Table G-2

Table G-2 is used to assign frequency index numbers to plant initiating events and CSC failures as found in the columns of Table G-1. The term “failure” must be understood to mean not merely failure of the CSC but also a violation of the process safety. In the example in Table G-1, accident sequence PPB2 1A involves loss of mass control over uranium dioxide (UO_2) in a blender.

Table G-2 provides two columns with two sets of criteria for assigning an index value, one based on type of CSC, the other on observed failure frequencies. Since CSC of a given type have a wide range of failure frequencies, assignment of index values based on this table should be done with caution. Due consideration should be given to whether the CSC will actually achieve the corresponding failure frequency in the next column.

Based on operational experience, more refined criteria for judging failure frequencies may be developed by each applicant. In the column labelled “Based on Type of CSC”, references to redundancy allow for CSCs that may themselves have internal redundancy to achieve a necessary level of reliability.

Another objective basis for assignment of an index value is actual observations of failure events. These actual events may have occurred in the applicant’s facility or in a comparable process elsewhere. Justification for specific assignments should be noted in the Comments column of Table G-1.

Note that indices less than (more negative than) -1 should not be assigned to CSC unless the configuration management, auditing, and other required management measures are of high quality, because, without these measures, the CSC may be changed or inadequately maintained. The reviewer should be able to determine this from the tabular summary of CSCs provided in the application. This summary should identify the process parameters to be controlled and their safety limits and include a thorough description of the CSC and the applied management measures.

G.3.5 Determination of failure probability index numbers in Table G-3

Occasionally, information concerning the reliability of a CSC may be available as a probability on demand. That is, there may be a history of tests or incidents where the system in question is demanded to function. To quantify such accident sequences, the demand frequency, the initiating event, and the demand failure probability of the CSC must be known. Table G-3 provides an assignment of index numbers for such CSC in a way that is consistent with Table G-2. The probability of failure on demand may be the likelihood that it is in a failed state when demanded (availability) or that it fails to remain functional for a sufficient time to complete its function. Justification for specific assignments should be noted in the Comments column of Table G-1.

G.3.6 Determining management measures for CSC

Table G-6 is an acceptable way of listing CSCs in all the general types of accident sequences leading to exceeding the approved USL. The items listed should include all CSCs and all external events whose low likelihood of occurrence is relied upon to meet the likelihood level identified in section 2.3.2.2.

The reviewer(s) examine this list to determine whether adequate management measures have been applied to each CSC to ensure its continual availability and reliability. The types of management measures are maintenance, training, configuration management, audits and assessments, quality assurance, etc. Every CSC in uncontrolled accident sequences leading to exceeding the likelihood levels identified in section 2.3.2.2 should be assigned at least a minimal set of management measures. Specifically, to defend against common mode failure of all CSCs on a process, this minimal set of measures must include: i) adequate configuration management, ii) regular auditing for the continued effectiveness of the CSC, iii) adequate labelling, training, or written procedures to ensure that the operating staff is aware of the safety function, iv) adequate surveillance and corrective maintenance, and v) adequate preventive maintenance.

If lesser or graded management measures are applied to some CSCs, Tables G-1 and G-6 and the narratives associated with them must identify to which CSC these lesser measures are applied. In addition, information indicating that acceptable reliability can be achieved with these lesser measures must be presented. The specifics of how each management measure, such as the surveillance interval, type of maintenance, or type of testing, is applied to each CSC need not be provided; it is recognized that such specific measures must be applied differently to each CSC to achieve adequate reliability. The formality, documentation, and quality assurance requirements applied to these direct management measures that may be graded generically in a risk-informed manner must be documented.

The following paragraphs describe the application of management measures to CSCs based on the risk importance of the item in an accident sequence, as defined by the uncontrolled likelihood index shown in Table G-1.

For a particular accident sequence that would have high uncontrolled likelihood index, CSCs should reduce the risk from initially high risk (an uncontrolled index of -2 or more from Table G-1) to an acceptable risk (controlled index of -6 or less).

Some accidents could have a relatively high uncontrolled likelihood. Further, for accident sequences resulting in nuclear criticality, double contingency should be achieved, thus requiring at least one more CSC and an initiating event of low probability. The uncertainty in determining low failure likelihood requires compensatory measures in the form of increased assurances (high-level criteria) that the CSC is indeed kept at low failure likelihood.

G.3.7 Risk-informed review of CSC

Column (e) in Table G-1 gives the likelihood indices / risk scores for each accident sequence that was identified in the CASA. There are two indices, uncontrolled and controlled. The uncontrolled index is a measure of risk without credit for the CSC. If the uncontrolled index is a value of -2 or more, while the controlled index is an acceptable value (-6 or less), the set of CSC involved are significant in achieving acceptable risk. That is, these CSCs have high risk significance. The uncontrolled likelihood index will be used by the reviewer(s) to identify all risk-significant systems of CSCs. These systems of CSCs will be reviewed more closely.

Table G-1: Sample Accident Sequence Summary and Likelihood Index Assignment

Process: uranium dioxide (UO_2) powder preparation (PP); Unit Process: additive blending; Node: blender hopper node (PPB2)

Accident Sequence / Identifier	Initiating Event (a1)	Enabling Events (if applicable) (a2)	Preventive Safety Parameter 1 or CSC 1 Failure/Success (b)	Preventive Safety Parameter 2 or CSC 2 Failure/Success (c)	Preventive Safety Parameters or CSC Failure/Success (d1, d2…)	Likelihood Index / Risk Score* T uncontrolled/ controlled (e)	Consequence Evaluation Reference	Comments and Recommendations
PPB2-1A (Criticality from blender leak of UO_2)	See CSC 1 (Note 1)		PPB2-C1: Mass Control Failure: Blender leaks UO_2 onto floor, critical mass exceeded Frq1 = -1 Dur1 = -4	PPB2-C2: Moderation Failure: Suffic. Water for criticality introduced while UO_2 on floor: Frq2 = -2	N/A	Unc T = -1 Con T = -7		CSC 2 fails while CSC 1 is in failed state. T = -1-4-2 = -7
PPB2-1B (Rad. release from blender leak of UO_2)	Blender leaks UO_2 Frqi = -1		PPB2-C1: Mass Control Success: leaked UO_2 below critical mass	PPB2-C2: Moderation Success: no Moderator	N/A	Unc T = -1 Con T = -4	Rad 36	Rad consequences, no criticality unmitigated sequence: CSC 1 and mitigation fail. T = -1-3 = -4 Mitig: CSC 1 fails, mitig CSC does not fail. T = -1
PPB2-1C	See CSC 1 (Note 1)		PPB2-C2: Moderation Failure: Suffic. water for Criticality on floor under UO_2 blender Frq1 = -2 Dur1 = -3	PPB2-C1: Mass Control Failure: Blender leaks UO_2 on floor while water present Frq2 = -2	N/A	Unc T = -2 Con T = -6		Criticality by reverse sequence of PPB2-1A. Moderation fails first. Note different likelihood. T = -6
PPB2-2	Fire in Blender Room Frqi = -2		Fire Suppression Failure: Fails on demand: Prf1 = -2	N/A	N/A	Unc T = -2 Con T = -4	Rad 37	Event sequence is just initiating event plus one CSC failure on demand

* Likelihood index / risk score, T, is a sum calculated as follows:

Uncontrolled index: T = a1 or T = a1 + a2

Controlled index (includes all indexes): T = a1 + a2 + b + c + d

Note: For these sequences, the initiating event is failure of one of the CSCs, hence the frequency is assigned under that CSC.

Table G-2: Failure Frequency Index Numbers

Frequency Index Number	Based on Evidence	Based on Type of CSC **	Comments
-6 *	External event with freq. < 10^{-6}/yr		If initiating event, no CSC needed.
-4 *	No failures in 30 years for hundreds of similar CSC in industry	Exceptionally robust passive engineered CSC (PEC), or an inherently safe process, or two independent active engineered CSC (AECs), PECs, or enhanced admin. CSC	Rarely can be justified by evidence. Further, most types of single CSC have been observed to fail.
-3 *	No failures in 30 years for tens of similar CSC in industry	A single CSC with redundant parts, each a PEC or AEC	
-2 *	No failure of this type in this facility in 30 years	A single PEC	
-1	A few failures may occur during facility lifetime	A single AEC, an enhanced admin. CSC, an admin. CSC with large margin, or a redundant admin. CSC	
0	Failures occur every 1 to 3 years	A single administrative CSC	
1	Several occurrences per year	Frequent event, inadequate CSC	Not for CSC, just initiating events
2	Occurs every week or more often	Very frequent event, inadequate CSC	Not for CSC, just initiating events

* Indices less than (more negative than) -1 should not be assigned to CSC unless the configuration management, auditing, and other management measures are of high quality, because, without these measures, the CSC may be changed or not maintained.

** The index value assigned to a CSC of a given type in column 3 may be one value higher or lower than the value given in column 1. Criteria justifying assignment of the lower (more negative) value should be given in the narrative describing NCSE methods. Exceptions require individual justification.

Table G-3: Failure Probability Index Numbers

Probability Index Number	Probability of Failure on Demand	Based on Type of CSC	Comments
-6 *	10^{-6}		If initiating event, no CSC needed.
-4 or -5 *	$10^{-4} – 10^{-5}$	Exceptionally robust passive engineered CSC (PEC), or an inherently safe process, or two redundant CSC more robust than simple admin. CSC (AEC, PEC, or enhanced admin.)	Can rarely be justified by evidence. Most types of single CSC have been observed to fail.
-3 or -4 *	$10^{-3} – 10^{-4}$	A single passive engineered CSC (PEC) or an active engineered CSC (AEC) with high availability	
-2 or -3 *	$10^{-2} – 10^{-3}$	A single active engineered CSC, or an enhanced admin. CSC or an admin. CSC for routine planned operations	
-1 or -2	$10^{-1} – 10^{-2}$	An admin. CSC that must be performed in response to a rare unplanned demand	

* Indexes less than (more negative than) -1 should not be assigned to CSC unless the configuration management, auditing, and other management measures are of high quality, because, without these measures, the CSC may be changed or not maintained.

Table G-4: Failure Duration Index Numbers

Duration Index Number	Average Failure Duration	Duration in Years	Comments
1	More than 3 years	10	
0	1 year	1	
-1	1 month	0.1	Formal monitoring to justify indices less than -1
-2	A few days	0.01	
-3	8 hours	0.001	
-4	1 hour	10^{-4}	
-5	5 minutes	10^{-5}	

Table G-5: Accident Sequence Descriptions
Process: uranium dioxide (UO_2) powder preparation (PP); Unit: additive blending; Node: blender hopper node (PPB2)

Accident (see Table G-1)	Description
PPB2-1A Blender UO_2 leak criticality	The initial failure is a blender leak of UO_2 that results in a mass sufficient for criticality on the floor (this event is not a small leak). Before the UO_2 can be removed, moderator sufficient to cause criticality is introduced. Duration of critical mass UO_2 on floor estimated to be 1 hour.
PPB2-1B Blender UO_2 leak, rad. Release	The initial failure is a blender leak of UO_2 that results in a mass insufficient for criticality on the floor or a mass sufficient for criticality but moderation failure does not occur. Consequences are radiological, not a criticality. A ventilated enclosure should mitigate the radiological release of UO_2. If the ventilated enclosure fails during cleanup or is not working, unmitigated consequences occur.
PPB2-1C	The events of PPB2-1A occur in reverse sequence—the initial failure is introduction of water onto the floor under the blender. Duration of this flooded condition is 8 hours. During this time, the blender leaks a critical mass of UO_2 onto the floor. Criticality occurs.
PPB2-2	Initiating event is a fire in the blender room. Fire is not extinguished quickly, and UO_2 is released from process equipment. Offsite dose estimated to exceed 1 mSv (100 mrem).

Table G-6: Descriptive List of Criticality Safety Controls
Process: uranium dioxide (UO_2) powder preparation (PP); Unit: additive blending; Node: blender hopper node (PPB2)

CSC Identifier	Safety Parameter and Limits	CSC Description	Maximum Value of Other Parameters	Reliability Management Measures	Quality Assurance Grade
PPB2-C1	Mass outside hopper: zero	Mass outside hopper: Hopper and outlet design prevent UO_2 leaks, double gasket at outlet	Full water reflection, enrichment 5%	Surveillance for leaked UO_2 each shift	A
PPB2-C2	Moderation: in $UO_2 < 1.5$ wt. % External water in area: zero	Moderation in UO_2: Two sample measurements by two persons before transfer to hopper External water: Posting excluding water, double piping in room, floor drains, roof integrity	Full water reflection, enrichment 5%	Drain, roof, and piping under safety-grade maintenance	A

Note: In addition to engineered CSC, Table G-6 should include descriptions of external initiating events of which the low likelihood is relied on to achieve acceptable risk, especially those which are assigned frequency indices lower than (-4). The descriptions of these initiating events should contain information supporting the frequency index value selected by the applicant.

Abbreviations

ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
CASA	criticality accident sequence assessment
CSA	Canadian Standards Association
CSC	criticality safety control
IAEA	International Atomic Energy Agency
NCS	nuclear criticality safety
NCSE	nuclear criticality safety evaluation
NCS staff	nuclear criticality safety staff
QM	quality management
USL	upper subcritical limit

Glossary

The following terms have definitions that are applicable to this specific document. Other specialized terms are defined in [49, 50, 51, 52].

accidents or accident sequences
Events or event sequences, including *external events*, that lead to violation of subcriticality margin (i.e., to exceeding the USL). This definition is of a restricted nature for the purposes of this document.

active engineered nuclear criticality safety control
A physical device that uses active sensors, electrical components, or moving parts to maintain safe process conditions without any required human action.

administrative (nuclear) criticality safety control
Either an *enhanced administrative control* or a *simple administrative control*, as defined herein.

agitation
The physical movement of the glass rings relative to one another that may cause breakage or gravitational settling.

area (or areas) of applicability
The limiting ranges of material compositions, geometric arrangements, neutron energy spectra, and other relevant parameters (such as heterogeneity, leakage, interaction, absorption, etc.) within which the bias of a calculational method is established.

areal density
The product of the thickness of a uniform slab and the density of fissionable material within the slab; hence, it is the mass of fissionable material per unit area of slab. For nonuniform slurries, the areal density limits are valid for a horizontal slab subject to gravitational settling, provided the restrictions for uniform slurries are met throughout.

array
Any fixed configuration of fissile or fissionable material units maintained by mechanical devices.

assembly
See *fissile assembly*.

benchmark experiment
A well-characterized experiment at the critical state that may be used to establish the reliability of calculational methods.

bias
A measure of the systematic differences between calculational method results and experimental data.

calculational method
The calculational procedures—mathematical equations, approximations, assumptions, associated numerical parameters (e.g., cross sections)—that yield the calculated results.

cell
See *storage cell*.

CNSC nuclear criticality safety requirements
Regulatory requirements and derived acceptance criteria listed in operating licence conditions or other legally enforceable documents. This definition is of a restricted nature for the purposes of this document.

control Raschig rings (controlled sample)
Raschig rings that are periodically removed from service for scheduled measurements, and then are returned to service after these short test periods.

controlled parameter
A parameter that is kept within specified limits, and, when varied, influences the margin of subcriticality.

credible abnormal conditions
Accidents or accident sequences that have frequency of occurrence equal to or more than one in a million years.

criticality accident
The release of energy as a result of accidental production of a self-sustaining or divergent neutron chain reaction.

criticality safety control (CSC)
Structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at a facility or to mitigate their potential consequences. This does not limit the licensee from identifying additional structures, systems, equipment, components, or activities of personnel (i.e., beyond those in the minimum set necessary for compliance with the performance requirements) as items relied on for safety. All safety controls, as defined in this document (*active engineered control*, *passive engineered control*, *simple administrative control*, and *enhanced administrative control*) are CSC. Also called *nuclear criticality safety control (CSC)*.

criticality safety staff
See *nuclear criticality safety staff (NCS staff)*.

double contingency principle
A characteristic or attribute of a process that has incorporated sufficient safety factors so that at least two unlikely, independent, and concurrent changes in process conditions are required before a nuclear criticality accident is possible.

drill
Supervised instruction intended to test, develop, maintain, and practice the skills required in a particular emergency response activity. A drill may be a component of an exercise.

effective multiplication factor (k_{eff})
Physically, the ratio of the total number of neutrons produced during a time interval (excluding neutrons produced by sources whose strengths are not a function of fission rate) to the total number of neutrons lost by absorption and leakage during the same interval. Mathematically (computationally), the eigenvalue number that, when divided into the actual mean number of neutrons emitted per fission in an assembly of materials, would make the calculated result for the nuclear chain reaction of that assembly critical.

emergency coordinator
A person authorized to direct the overall emergency response.

emergency response
Actions taken from the time of identification of a suspected, imminent, or actual criticality accident to stabilization of the event. These actions include the assumption that an accident has occurred, response to the emergency, and actions to begin subsequent recovery operations.

engineered (nuclear) criticality safety control
Either an *active engineered control* or a *passive engineered control*.

enhanced administrative control
A procedurally required or prohibited human action, combined with a physical device that alerts the operator that the action is needed to maintain safe process conditions, or otherwise adds substantial assurance of the required human performance.

excessive radiation dose
Any dose to personnel corresponding to an absorbed dose from neutrons and gamma rays equal to or greater than 0.12 Gy (12 rad) in free air.

exercise
An activity that tests one or more portions of the integrated capability of emergency response plans, equipment, and organizations.

external event
An event for which the likelihood cannot be altered by changes to the regulated facility or its operation. This would include all *natural phenomena events*, plus airplane crashes, explosions, toxic releases, fires, etc., occurring near or on the nuclear site.

fissile assembly
A system consisting of fissile material and other components that significantly influence the reactivity.

fissile material
A material, other than natural uranium, that is capable of sustaining a thermal neutron chain reaction.

fissile nuclide
A nuclide capable of undergoing fission by interaction with slow neutrons provided the effective thermal neutron production cross section, $\upsilon\sigma_f$, exceeds the effective thermal neutron absorption cross section σ_a. Most actinide nuclides containing an even number of neutrons are non-fissile, but there may be exceptions, such as ^{232}U and ^{236}Pu (which have even numbers of neutrons and approximately equal thermal capture and fission cross sections), which perhaps can be made critical with slow neutrons. Conversely, whereas most nuclides with an odd number of neutrons are fissile, ^{237}U (which is an odd number of neutrons nuclide with a very small thermal fission cross section) cannot be made critical with thermal neutrons.

fissionable
Capable of undergoing fission.

fixed moderator
A moderator with an established geometric relationship to the locations occupied by the fixed neutron absorber and fissionable material.

fixed neutron absorber
Neutron absorbers in solids with an established geometric relationship to the locations occupied by fissionable material.

fuel rod
A long slender column of material containing fissile nuclides, normally encapsulated by metallic tubing.

fuel unit
The fundamental item to be handled, stored, or transported. It may be an assembly of fuel rods, canned spent fuel, or consolidated fuel rods.

glass volume fraction
The fraction of the interior volume of a Raschig ring-filled vessel that is occupied by the glass in the rings.

immediate evacuation zone
The area surrounding a potential criticality accident location that must be evacuated without hesitation if a criticality accident alarm signal is activated.

***in situ* experiment**
Neutron multiplication or other nuclear reactivity-determining measurement on a subcritical fissile assembly where protection of personnel against the consequences of a criticality accident is not provided.

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