National Nuclear Security Administration

Criticality Safety

Qualification Standard

Reference Guide

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Acronyms	
AC	administrative control
ALARA	as low as is reasonably achievable
ANSI	American National Standards Institute
ANS	American Nuclear Society
BL	beaker leaching
CAAS	criticality accident alarm system
CAP	corrective action plan
CCR	criticality control review
CHF	constant heat flow
CI	criticality index
CO	certifying officer
CPIC	capital planning and investment control
CRD	contractor requirements document
CSDR	conceptual safety design report
CSE	criticality safety evaluation
CSO	cognizant secretarial officer
CSP	criticality safety program
CSVR	conceptual safety validation report
CTA	central technical authority
c-t-c	center-to-center
D&D	deactivation and decommissioning
DARTS	departmental audit report tracking system
db	decibel
DDC	direct digital control
DICARC	Departmental Internal Control and Audit Review Council
DoD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
D/P	differential pressure
DSA	documented safety analysis
EA	enterprise architecture
EO	Executive Order
EIS	environmental impact statement
ET	event tree (analysis)
FAQS	functional area qualification standard
FMEA	failure modes and effects analysis
FPD	Federal project director
FT	fault tree (analysis)
GAO	Government Accountability Office

Acronyms	
GM	Geiger-Mueller
HAZOP	hazard and operations study
HCO	headquarters certifying official
HENC	high efficiency neutron counter
HLW	high-level waste
H/P	high pressure
HPGe	high purity germanium
HQ	headquarters
HSS	Office of Health, Safety, and Security
HRGS	high resolution gamma-ray spectrometry
IAEA	International Atomic Energy Agency
ICIS	integrated crate interrogation system
IG	Inspector General
IP	implementation plan
ISM	integrated safety management
ISOCS	in situ object counting system
ISSM	integrated safeguards and security management
IT	information technology
IWAS	integrated waste assay system
KSA	knowledge , skill, and ability
LCO	limiting conditions for operation
LCS	limiting control settings
L/P	low pressure
LVDT	linear variable differential transformers
MAR	material at risk
MC&A	material control and accountability
MeV	million electron volt
MORT	management oversight risk tree
M&O	managing and operations
NaI	sodium iodide
NARA	National Archives and Records Administration
NCS	nuclear criticality safety
NDA	nondestructive assay
NMMSS	nuclear materials management and safeguards system
NNSA	National Nuclear Security Administration
NRC	Nuclear Regulatory Commission
NSCS	networked sensing and control system
NUREG	U.S. Nuclear Regulatory Commission regulation
OL	operating limit

Acronyms	
OMB	Office of Management and Budget
ORPS	occurrence reporting and processing system
ORR	operational readiness review
PAAA	Price-Anderson Amendments Act
PB	performance baseline
PDSA	preliminary documented safety analysis
PHAR	preliminary hazard analysis report
PI	proportional plus integral
PID	proportional plus integral plus derivative
PLC	programmable logic controller
POA	plan of action
PRA	probabilistic risk assessment
PSDR	preliminary safety design report
PSM	process safety management
PSO	program secretarial officer
QA	quality assurance
QAP	quality assurance program
RIS	reporting identification symbol
RA	readiness assessment
RR	readiness review
S&S	safeguards and security
SAA	Startup authorization authority
SAC	specific administrative control
SARP	safety analysis report for packaging
SDS	safety design strategy
SER	safety evaluation report
SGS	segmented gamma-ray scanner
SL	safety limit
SNF	spent nuclear fuel
SNM	special nuclear material
SNR	startup notification report
SR	surveillance requirements
SRL	Savannah River Laboratory
SRS	Savannah River Site
SSC	structure, system or component
STI	scientific and technical information
STICG	scientific and technical information coordinating group
THERP	technique for human error rate prediction
TIPR	technical independent project review

Acronyms	
TL	truckload
TRANSCOM	DOE transportation tracking and communications systems
TRU	transuranic
TSR	technical safety requirement
TSUNAMI	tools for sensitivity and uncertainty analysis methodology implementation
UCNI	unclassified controlled nuclear information
USQ	unreviewed safety question
USQD	unreviewed safety question determination
WIPP	Waste Isolation Pilot Project

PURPOSE

The purpose of this reference guide is to provide a document that contains the information required for a Department of Energy (DOE)/National Nuclear Security Administration (NNSA) technical employee to successfully complete the Criticality Safety Functional Area Qualification Standard (FAQS). Information essential to meeting the qualification requirements is provided; however, some competency statements require extensive knowledge or skill development. Reproducing all the required information for those statements in this document is not practical. In those instances, references are included to guide the candidate to additional resources. This reference guide should be used in conjunction with the reference guide document prepared by the Criticality Safety Support Group at the NNSA Service Center, which is available from the same website as this document.

SCOPE

This reference guide addresses the competency statements in the April 2009 edition of DOE-STD-1173-2009, *Criticality Safety Functional Area Qualification Standard*. The qualification standard contains 40 competency statements.

Please direct your questions or comments related to this document to the NNSA Learning and Career Development Department.

PREFACE

Competency statements and supporting knowledge and/or skill statements from the qualification standard are shown in contrasting bold type, while the corresponding information associated with each statement is provided below it.

A comprehensive list of acronyms and abbreviations is found at the beginning of this document. It is recommended that the candidate review the list prior to proceeding with the competencies, as the acronyms and abbreviations may not be further defined within the text unless special emphasis is required.

The competencies and supporting knowledge, skill, and ability (KSA) statements are taken directly from the FAQS. Most corrections to spelling, punctuation, and grammar have been made without remark. Only significant corrections to errors in the technical content of the discussion text source material are identified. Editorial changes that do not affect the technical content (e.g., grammatical or spelling corrections, and changes to style) appear without remark. When they are needed for clarification, explanations are enclosed in brackets.

Every effort has been made to provide the most current information and references available as of April 2011. However, the candidate is advised to verify the applicability of the information provided. It is recognized that some personnel may oversee facilities that utilize predecessor documents to those identified. In those cases, such documents should be included in local qualification standards via the Technical Qualification Program.

In the cases where information about an FAQS topic in a competency or KSA statement is not available in the newest edition of a standard (consensus or industry), an older version is referenced. These references are noted in the text and in the bibliography.

ACKNOWLEDGEMENTS

Some of the references and responses to the knowledge, skill, and ability statements that are related to specific criticality safety personnel responsibilities were provided by a team of subject matter experts at the NNSA Service Center.

TECHNICAL COMPETENCIES

1. **Criticality safety personnel must demonstrate a working level knowledge of the fission process.**

   a. **Define the following terms:**
      - **Excitation energy**
      - **Cross section**
      - **Fissile material**
      - **Fissionable material**
      - **Fertile material**

The following is taken from DOE-STD-1019/1-93.

Excitation Energy

The measure of how far the energy level of a nucleus is above its ground state is called the excitation energy (Eexc). For fission to occur, the excitation energy must be above a particular value for that nuclide. The critical energy (Ecrit) is the minimum excitation energy required for fission to occur.

Cross Section

The probability of a neutron interacting with a nucleus for a particular reaction is dependent upon not only the kind of nucleus involved, but also the energy of the neutron. Accordingly, the absorption of a thermal neutron in most materials is much more probable than the absorption of a fast neutron. Also, the probability of interaction will vary depending upon the type of reaction involved.

The probability of a particular reaction occurring between a neutron and a nucleus is called the microscopic cross section (σ) of the nucleus for the particular reaction. This cross section will vary with the energy of the neutron. The microscopic cross section may also be regarded as the effective area the nucleus presents to the neutron for the particular reaction. The larger the effective area, the greater the probability for reaction.

Because the microscopic cross section is an area, it is expressed in units of area, or square centimeters. A square centimeter is tremendously large in comparison to the effective area of a nucleus, and it has been suggested that a physicist once referred to the measure of a square centimeter as being "as big as a barn" when applied to nuclear processes. The name has persisted and microscopic cross sections are expressed in terms of barns, where 1 barn = 10^{-24} cm^2.

Whether a neutron will interact with a certain volume of material depends not only on the microscopic cross section of the individual nuclei but also on the number of nuclei within that volume. Therefore, it is necessary to define another kind of cross section known as the macroscopic cross section (Σ). The macroscopic cross section is the probability of a given reaction occurring per unit travel of the neutron, and is related to the microscopic cross section (σ) by the relationship: $\Sigma = N \sigma$

where:

Σ = macroscopic cross section (cm^{-1})
N = atom density of material ($atoms/cm^3$)
σ = microscopic cross-section (cm^2)

The difference between the microscopic and macroscopic cross sections is extremely important and is restated for clarity. The microscopic cross section (σ) represents the effective target area that a single nucleus presents to a bombarding particle. The units are given in barns or cm^2.

The macroscopic cross section (Σ) represents the effective target area that is presented by all of the nuclei contained in 1 cm^3 of the material. The units are given as 1/cm or cm^{-1}.

Fissile Material

A fissile material is composed of nuclides for which fission is possible with neutrons of any energy level. What is especially significant about these nuclides is their ability to be fissioned with zero kinetic energy neutrons (thermal neutrons). Thermal neutrons have very low kinetic energy levels (essentially zero) because they are roughly in equilibrium with the thermal motion of surrounding materials. Therefore, in order to be classified as fissile, a material must be capable of fissioning after absorbing a thermal neutron. Consequently, they impart essentially no kinetic energy to the reaction. Fission is possible in these materials with thermal neutrons, since the change in binding energy supplied by the neutron addition alone is high enough to exceed the critical energy. Some examples of fissile nuclides are uranium-235, uranium-233, and plutonium-239.

Fissionable Material

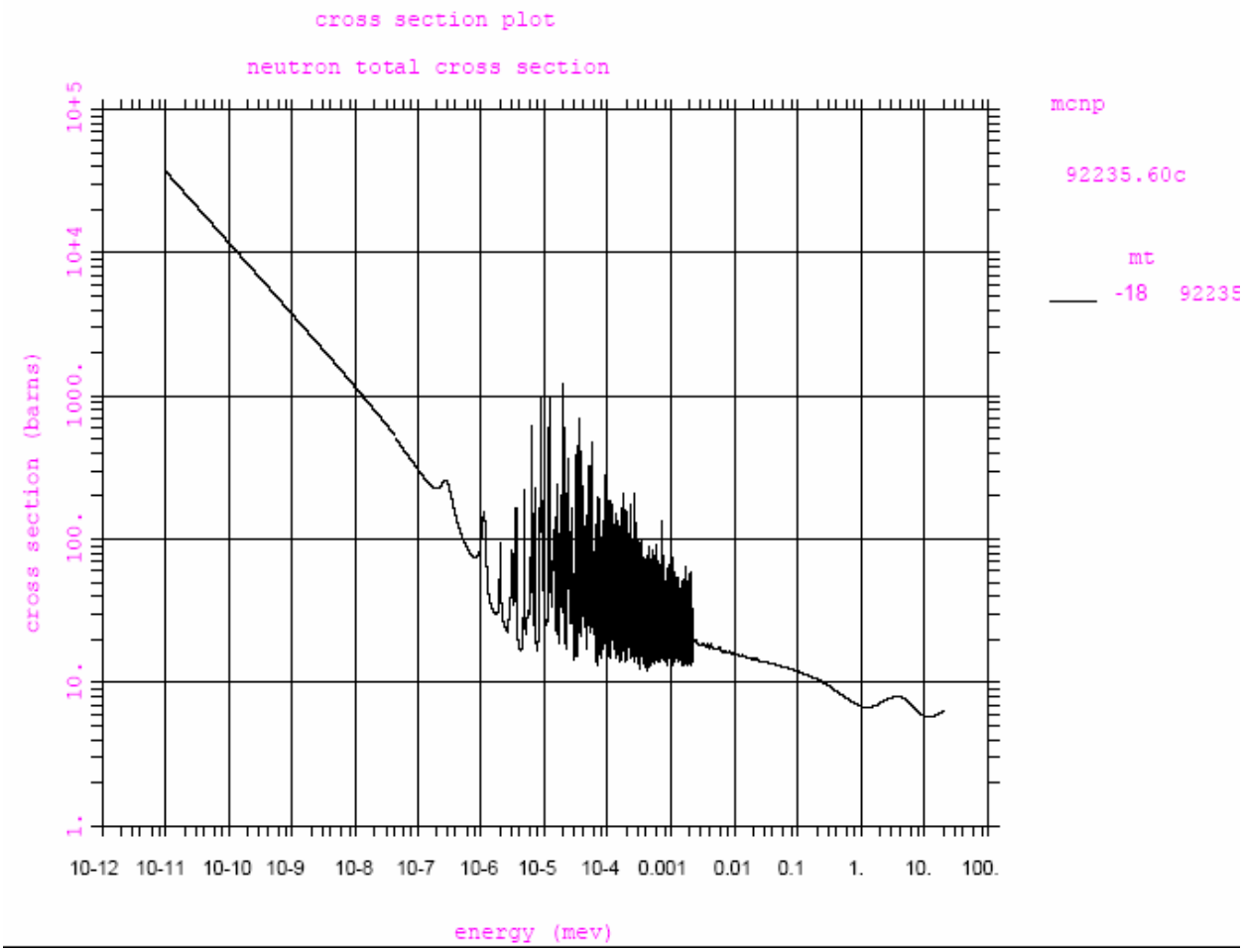
A fissionable material is composed of nuclides for which fission with neutrons is possible. All fissile nuclides fall into this category. However, also included are those nuclides that can be fissioned only with high energy neutrons. The change in binding energy that occurs as the result of neutron absorption results in a nuclear excitation energy level that is less than the required critical energy. Therefore, the additional excitation energy must be supplied by the kinetic energy of the incident neutron. The reason for this difference between fissile and fissionable materials is the so-called odd-even effect for nuclei. It has been observed that nuclei with even numbers of neutrons and/or protons are more stable than those with odd numbers. Therefore, adding a neutron to change a nucleus with an odd number of neutrons to a nucleus with an even number of neutrons produces an appreciably higher binding energy than adding a neutron to a nucleus already possessing an even number of neutrons. Thus, fissionable material is material for which fission caused by neutron absorption is possible provided the kinetic energy added with the binding energy is greater than the critical energy. Some examples of nuclides requiring high energy neutrons to cause fission are thorium-232, uranium-238, and plutonium-240.

Fertile Material

Fertile materials are materials that can undergo transmutation to become fissile materials. Two fertile nuclides, thorium-232 and uranium-238, undergo transmutation to produce uranium-233 and plutonium-239, respectively.

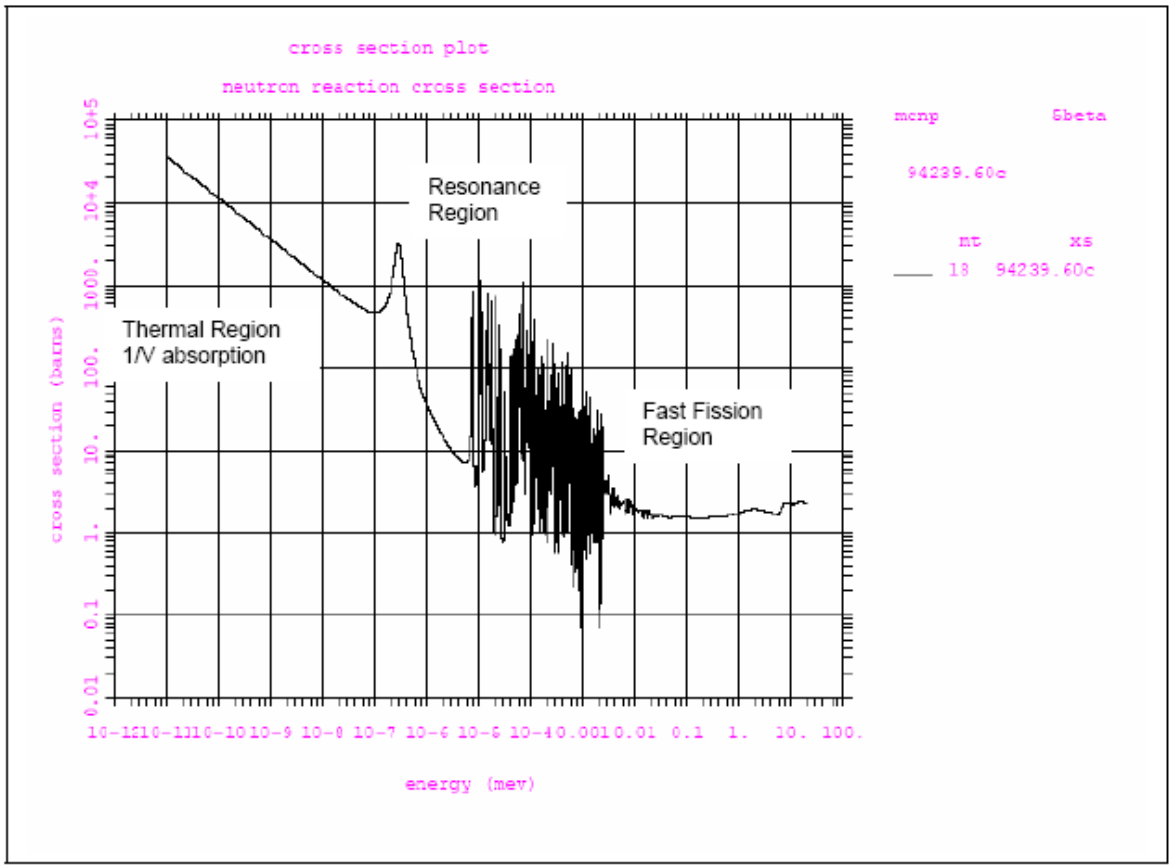
b. Sketch the fission cross section for both U-235 and Pu-239 as a function of neutron energy. Label each significant energy region and explain the implications of the shape of the curves for criticality safety.

This is a performance-based KSA. The Qualifying Official will evaluate its completion. However, the following information may prove useful.



Source: BNL, NNDC, Evaluated Nuclear Data File Retrieval & Plotting

Figure 1. Total cross section of U-235

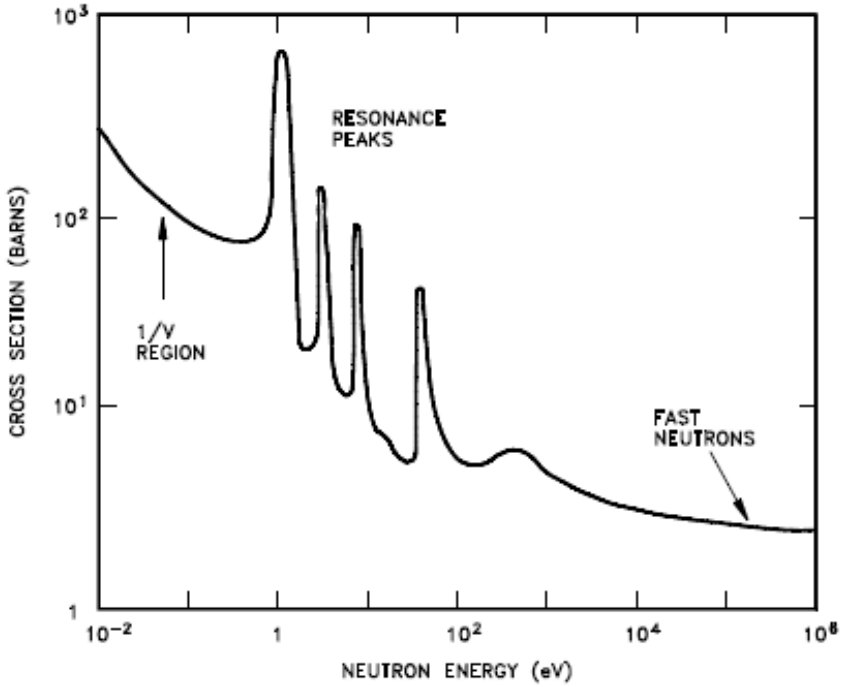


Source: BNL, NNDC, Evaluated Nuclear Data File Retrieval & Plotting

Figure 2. Total cross section of Pu-239

The following is taken from DOE-HDBK-1019/1-93.

For a considerable number of nuclides of moderately high (or high) mass numbers, an examination of the variation of the absorption cross section with the energy of the incident neutron reveals the existence of three regions on a curve of absorption cross section versus neutron energy. This cross section is illustrated in figure 3. First, the cross section decreases steadily with increasing neutron energy in a low energy region, which includes the thermal range (E < 1 eV). In this region the absorption cross section, which is often high, is inversely proportional to the velocity (v). This region is frequently referred to as the "1/v region," because the absorption cross section is proportional to 1/v, which is the reciprocal of neutron velocity. Following the 1/v region, there occurs the "resonance region" in which the cross sections rise sharply to high values called "resonance peaks" for neutrons of certain energies, and then fall again. These energies are called resonance energies and are a result of the affinity of the nucleus for neutrons whose energies closely match its discrete, quantum energy levels. That is, when the binding energy of a neutron plus the kinetic energy of the neutron are exactly equal to the amount required to raise a compound nucleus from its ground state to a quantum level, resonance absorption occurs.



Source: DOE-HDBK-1019/1-93

Figure 3. Typical neutron absorption cross section vs. neutron energy

c. Explain why only the heaviest nuclei are easily fissioned.

The DOE-HDBK-1019-93, *DOE Fundamentals Handbook, Instrumentation and Control,* answer is that the heaviest nuclei require only a small distortion from a spherical shape (small energy addition) for the relatively large coulomb forces forcing the two halves of the nucleus apart to overcome the attractive nuclear forces holding the two halves together. Consequently, the heaviest nuclei are easily fissionable compared to lighter nuclei. This liquid drop explanation is an oversimplification, and modern fission theory includes quantum theory. See LA-14098, Lynn, *Modern Fission Theory for Criticality.*

d. Explain why uranium-235 fissions with thermal neutrons and uranium-238 fissions only with fast neutrons.

The following was derived from information in DOE-HDBK-1019/1-93 by a team of subject matter experts at the NNSA Service Center.

Uranium-238 is even-even, which leads to a much more stable quantum state, and quantum tunneling is required to cross the fission barrier. The simpler explanation is that ^{238}U has a fission cross section that is six orders of magnitude less than Uranium-235.

The following is taken from DOE-HDBK-1019/1-93.

Uranium-235 fissions with thermal neutrons because the binding energy released by the absorption of a neutron is greater than the critical energy for fission. The binding energy released by uranium-238 absorbing a neutron is less than the critical energy, so additional kinetic energy must be possessed by the neutron for fission to be possible.

e. Characterize the fission products in terms of mass groupings and radioactivity.

The following was derived from information in by a team of subject matter experts at the NNSA Service Center.

The fission products of ^{235}U form a Mae West curve with peaks at mass numbers of about 90-100 and 135-145. The fission products are neutron rich, and mostly beta decay to more stable isotopes. The decay to stability is also accompanied by gamma emissions. See DOE-HDBK-1019/1-93, page 57, figure 21.

f. Define the following terms:

- **Sub-critical**
- **Critical**
- **Super-critical**
- **Reproduction factor**
- **Prompt neutron fraction**
- **Delayed neutron fraction**

The following is taken from DOE-HDBK-1019/2-93.

Sub-Critical

If the neutron production is less than the absorption and leakage, the reactor is called subcritical. In a subcritical reactor, k_{eff} is less than one, and the flux decreases each generation.

Critical

The condition where the neutron chain reaction is self-sustaining and the neutron population is neither increasing nor decreasing is referred to as the critical condition and can be expressed by the simple equation $k_{eff} = 1$.

Super-Critical

If the neutron production is greater than the absorption and leakage, the reactor is called supercritical. In a supercritical reactor, k_{eff} is greater than one, and the neutron flux increases each generation.

Reproduction Factor

Most of the neutrons absorbed in the fuel cause fission, but some do not. The reproduction factor (η) is defined as the ratio of the number of fast neutrons produced by thermal fission to the number of thermal neutrons absorbed in the fuel. The reproduction factor is shown below.

$$\eta = \frac{\text{number of fast neutrons produced by thermal fission}}{\text{number of thermal neutrons absorbed in the fuel}}$$

The reproduction factor can also be stated as a ratio of rates as shown below.

$$\eta = \frac{\text{rate of production of fast neutrons by thermal fission}}{\text{rate of absorption of thermal neutrons by the fuel}}$$

The rate of production of fast neutrons by thermal fission can be determined by the product of the fission reaction rate ($\Sigma_f^u \phi^u$) and the average number of neutrons produced per fission (υ). The average number of neutrons released in thermal fission of ^{235}U is 2.42. The rate of absorption of thermal neutrons by the fuel is $\Sigma_a^u \phi^u$. Substituting these terms into the equation above results in the following equation.

$$\eta = \frac{\Sigma_f^U \phi^U \nu}{\Sigma_a^U \phi^U}$$

Prompt Neutron Fraction

The prompt neutron fraction is the fraction of all fission neutrons that are produced essentially instantaneously from fission.

Delayed Neutron Fraction

The delayed neutron fraction is the fraction of all fission neutrons that are born as delayed neutrons for a particular type of fuel (i.e., ^{235}U and ^{239}Pu).

g. Discuss isotopes other than U-235 and Pu-239 that are fissionable.

The following was provided by a team of subject matter experts at the NNSA Service Center.

^{233}U is fissile, and has a minimum critical mass between ^{239}Pu and ^{235}U. ^{232}Thorium and ^{238}U are both fissionable. Roughly, any actinide that has a finite fission cross section is fissionable. Isotopes with odd numbers of neutrons are likely to be fissile. Isotopes with even numbers of neutrons typically have a fission threshold. Isotopes above the actinide series and actinides above element 98 are likely to be fissile or fissionable, but do not exist in quantities sufficient to be of criticality concern.

2. Criticality safety personnel must demonstrate a working level knowledge of the various types of radiation interaction with matter.

The information for this competency statement is taken from DOE-HDBK-1019/1-93

a. Describe the interactions of the following with matter:

- **Alpha particle**
- **Beta particle**
- **Positron**
- **Neutron**

Alpha Particle (α)

α has a high linear energy transfer factor due to both size and mass. α frequently adds enough mass/energy to a nucleus to cause a neutron emission (α-n). α most often causes only electron ionization. α eventually absorbs two electrons to become a helium atom.

The alpha particle consists of two neutrons and two protons, so it is essentially the same as the nucleus of a helium atom. Because it has no electrons, the alpha particle has a charge of +2. This positive charge causes the alpha particle to strip electrons from the orbits of atoms in its vicinity. As the alpha particle passes through material, it removes electrons from the orbits of atoms it passes near. Energy is required to remove electrons and the energy of the alpha particle is reduced by each reaction. Eventually the particle will expend its kinetic energy, gain 2 electrons in orbit, and become a helium atom. Because of its strong positive charge and large mass, the alpha particle deposits a large amount of energy in a short distance of travel. This rapid, large deposition of energy limits the penetration of alpha particles.

Beta Particle (β-)

The beta particle has a smaller linear energy transfer than α, and thus penetrates farther from the release point. Beta particles are eventually absorbed as an atomic electron.

Beta particles cause ionization by displacing electrons from atom orbits. The ionization occurs from collisions with orbiting electrons. Each collision removes kinetic energy from the beta particle, causing it to slow down. Eventually the beta particle will be slowed enough to allow it to be captured as an orbiting electron in an atom. Although more penetrating than the alpha, the β- is relatively easy to stop and has a low power of penetration.

Positron (β+)

Except for the positive charge, β+ are identical to β- and interact with matter in a similar manner. Positrons are very short-lived, however, and quickly are annihilated by interaction with a negatively charged electron, producing two gammas with a combined energy equal to the rest mass of the positive and negative electrons (about 1 million electron volts [MeV]).

Neutron

The neutron has negligible interaction with the electron cloud. The interactions with the nuclei can leave the target nucleus with increased kinetic energy (elastic), increased kinetic energy and an excited state (inelastic), or be absorbed, causing a change in kinetic energy and an excited state. The excited states are alleviated by photon decay or for inelastic and photon decay or fission for absorption. Because of its lack of charge, the neutron is difficult to stop and has a high penetrating power.

b. Describe the following ways that gamma radiation interacts with matter:
- **Compton scattering**
- **Photoelectric effect**
- **Pair production**

Compton Scattering

In Compton scattering a gamma interacts with an orbital or free electron; however, in this case, the photon loses only a fraction of its energy. The actual energy loss depends on the scattering angle of the gamma. The gamma continues on at lower energy, and the energy difference is absorbed by the electron.. This reaction becomes important for gamma energies of about 0.1 MeV and higher.

Photoelectric Effect

In photoelectric effect, a low energy gamma strikes an atom and the total energy of the gamma is expended in ejecting an electron from orbit. The result is ionization of the atom and expulsion of a high energy electron. This reaction is most predominant with low energy gammas interacting in materials with high atomic weight and rarely occurs with gammas having an energy above 1 MeV. Annihilation of the gamma results. Any gamma energy in excess of the binding energy of the electron is carried off by the electron in the form of kinetic energy.

Pair Production

In pair production, when a high energy gamma passes close enough to a heavy nucleus, the gamma completely disappears, and an electron and a positron are formed. For this reaction to take place, the original gamma must have at least 1.02 MeV energy. Any energy greater than 1.02 MeV becomes kinetic energy shared between the electron and positron. The probability of pair production increases significantly for higher energy gammas.

3. Criticality safety personnel must demonstrate a working level knowledge of criticality control and safety parameters.

a. Discuss the effects and applications of the following factors relevant to criticality safety of operations:
- **Mass**
- **Absorption**
- **Geometry**
- **Interaction**
- **Concentration**
- **Moderation**
- **Enrichment**
- **Reflection**
- **Volume**

The material for KSAs a and b was derived from nuclear criticality safety engineer training (NCSET) modules on the DOE/Nuclear Criticality Safety Program website, and from *Nuclear Criticality Safety, Theory and Practice* by R. A. Knief, by a team of subject matter experts at the NNSA Service Center.

Mass

As an extensive property, mass is the most commonly used measure of how much fissile material is available to contribute to a potentially critical system. Mass is frequently one of the easiest items to control.

Absorption

At the atomic level, neutron absorption is usually accompanied by the release of a photon to stabilize the nucleus. Many materials function as absorbers, including water and steel (due to hydrogen and largely iron, respectively). The absorption by hydrogen is most of the reason that there is a lower critical concentration limit. If the fissionable atoms only are included in a computation, (that is, no absorber materials), the neutron production-to-loss ratio in an infinite system is the average neutron production per fission, or in jargon K-infinity equals nu-bar.

Geometry

Geometry normally refers to the defined geometric form of a vessel (in the case of liquids) or solid shape. Geometries used in criticality control of processes frequently consist of cylinders and slabs. The geometry of a vessel has an effect on neutron leakage. When the shape of container provides enough leakage that a credible accumulation or concentration of fissile material cannot achieve criticality, it is commonly referred to as geometrically favorable.

Interaction

Interaction is a shorthand term for the number of neutrons leaving one fissile mass to cause fission in a nearby fissile mass. The sizes, distances, and intervening materials all influence interaction.

Concentration

The proportion of fissile material in the matrix greatly influences the critical mass or critical volume of matrix material. Concentration directly relates to the number of non-fissile atoms between a neutron and a fissionable atom, and thus the probability of the neutron being lost before interacting with a fissile atom. Concentration also influences the probability of neutron slowing to a more probable fission energy. Concentration is frequently discussed as H/fissile ratio (H/X, H/Pu, or H/U), since water or aqueous solutions, plastics, and cellulosic materials are common diluents for fissile material. The mass-concentration curve of Pu in water shows the effect of offsetting absorption and moderation effects as concentration changes. In solutions, the concentration of fissile material has a large impact on the critical mass and critical volume. For example, the minimum critical masses of ^{235}U and ^{239}Pu are about 800 and 500 grams, respectively, at solution concentrations of 30 to 40 grams/liter (g/l).

Moderation

When fissionable material is in solution, or present as finely divided particles, the presence of a neutron moderator, such as water or a hydrocarbon, can effect a significant reduction in the amount of fissile material required for criticality. The interaction of neutrons with light nuclei, such as hydrogen, lithium, beryllium, or carbon, reduces the neutron energy after only a few collisions. Slow neutrons interact more readily with nuclei, and therefore they have a greater probability of causing fission in ^{235}U or ^{239}Pu nuclei. There is an optimum degree of moderation because if the ratio of hydrogen nuclei to uranium nuclei becomes too large, neutron capture in the hydrogen becomes competitive with fission in the uranium.

Enrichment

Enrichment is the amount of fissile isotopes relative to the total amount of the element. Highly enriched uranium has more ^{235}U atoms than uranium ore. The higher the enrichment, the less material is required to achieve criticality. Enrichment is usually given as the ratio of the fissile isotope to the total element, in mass percent. Enrichment, especially in low enriched materials (typically uranium), has a large effect on the critical mass of the fissile isotope.

At low enrichments, the effect of resonance capture by non-fissile isotopes becomes an important consideration. A matrix of small pieces of fissile material is frequently more reactive than a homogeneous matrix for this reason. Natural uranium can be made critical by carefully sizing the fuel pieces and moderator properties to minimize the effect of resonance capture by ^{238}U. For these reasons, natural uranium is not an accidental criticality concern in nearly all situations. (An obvious exception would be CANDU fuel and heavy water in proximity, which is not credible in most DOE applications.)

Reflection

Reflection is the return of neutrons back into a region containing fissile material by collisions with atoms in surrounding materials. For example, a thin layer of cutting fluid on a fissile part being machined represents almost no reflection. Immersing that same part in a bucket of water, such that there are a few inches of water surrounding it, could reduce the amount of fissile material required for criticality by about a factor of two. Reflector materials are frequently also moderators, and neutron poisons are better reflectors than they are poisons. Reflection decreases the loss term of the neutron balance equation.

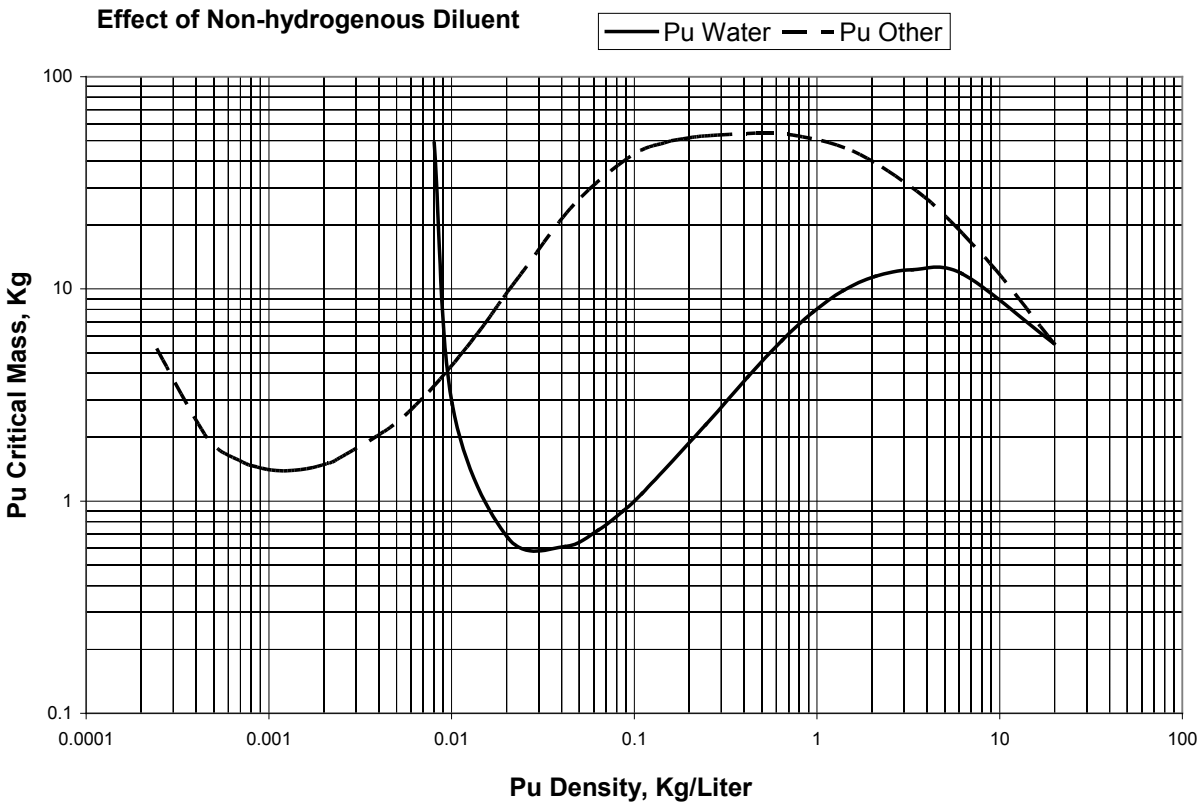
Volume

Volume is also an extensive property, and frequently easier to control than mass. For any fissile material, there is a minimum critical volume such that any smaller volume of that material (in isolation) is subcritical. However, small changes in the composition of the fissile material can result in large changes in the minimum critical volume, especially if carbon or other light scattering elements are involved. For hydrogenous systems, there is frequently a large portion of the critical volume vs. mass or volume vs. concentration curve where the critical volume changes slowly. This region often extends from somewhat dilute into the region where the system is no longer credibly liquid, making volume a very useful control for single units of solution, or a very useful secondary control.

b. Discuss the influence of the presence of non-fissionable materials mixed with, or in contact with, fissionable material on nuclear criticality safety. Include a discussion of the effects of mild absorbers (e.g., some absorption, but mostly scattering), and materials that behave as almost pure elastic scatterers, either with or without significant moderation per collision (e.g., describe the effect of diluting plutonium oxide with either wet or dry silica, contrast the two, and explain the effects from an interaction viewpoint) .

The addition of non-fissionable materials to a Pu or U metal system initially raises the minimum critical mass (more so with Pu than U), as the core density effect predominates. As more diluent is added, the moderation effect decreases the critical mass until the absorption properties of the diluent drive the system sub-critical. The general effect of

non-hydrogenous diluents is to raise the minimum critical mass and lower the minimum critical concentration as compared to hydrogenous diluents.



c. Discuss the effects of density, heterogeneity, and enrichment with respect to resonance escape and lumped fuel.

The following is taken from DOE-HDBK-1019/2-93.

The value of the resonance escape probability is determined largely by the fuel-moderator arrangement and the amount of enrichment of ^{238}U (if any is used). To undergo resonance absorption, a neutron must pass close enough to a ^{238}U nucleus to be absorbed while slowing down. In a homogeneous reactor the neutron does its slowing down in the region of the fuel nuclei, and this condition is easily met. This means that a neutron has a high probability of being absorbed by ^{238}U while slowing down; therefore, its escape probability is lower. In a heterogeneous reactor, however, the neutron slows down in the moderator where there are no atoms of ^{238}U present. Therefore, it has a low probability of undergoing resonance absorption, and its escape probability is higher.

The value of the resonance escape probability is not significantly affected by pressure or poison concentration. In water-moderated, low ^{238}U enrichment reactors, raising the temperature of the fuel will raise the resonance absorption in ^{238}U due to the Doppler effect. The increase in resonance absorption lowers the resonance escape probability, and the fuel temperature coefficient for resonance escape is negative. The temperature

coefficient of resonance escape probability for the moderator temperature is also negative. As water temperature increases, water density decreases. The decrease in water density allows more resonance energy neutrons to enter the fuel and be absorbed. The value of the resonance escape probability is always slightly less than one.

d. Discuss the effects of mixtures of different fissionable nuclides and the appropriate applications of the "rule of fractions" and "fissionable equivalent mass" concepts.

The following is taken from Westinghouse Savannah River Company, WSRC-TR-91-569, *Validity of the Rule-of-Fractions for Assuring Criticality Safety Margins*.

The rule of fractions has been used in many applications to ensure a minimum margin in K_{eff} for various homogeneous and heterogeneous mixtures of fissile and fissionable isotopes. In its simplest form the rule of fractions may be stated as follows:

$$\sum_i \frac{a_i}{A_i} \leq 1 \qquad (1)$$

where a_i is the actual amount of quantity i present, and A_i is an amount of quantity i that corresponds to a specified margin in $K_{ef\,f}$. The implication of equation 1 is that if the K_{eff} margin is known for each quantity A_i , then a margin at least equal to the smallest K_{eff} margin for any A_i is maintained when arbitrary amounts of quantities i are intermixed, and the conditions of equation 1 are met.

Assurance of safe margins in K_{eff} based on the above equation has been used in several applications at the Savannah River Site (SRS).

Three examples of the use of the rule of fractions to ensure a margin in K_{eff} will be shown below. Each example is a documented use of the rule of fractions at the SRS to ensure a justified margin of safety in K_{eff}.

Mixtures of Fissile Containers in Shipping Packing

An SRS shipping package is used to ship three types of fissile materials (referred to as type A, type B, and type C respectively). The package contains up to five bottles of fissile material in a vertical column.

The rule of fractions was used as a quick method to determine the K_{eff} for mixtures of bottles of the three fissile materials. This was done by weighting the infinite array reactivity of each type of material by itself by the fractional number of bottles of that kind present. The calculated K_{eff} for each infinite array was as follows:

PACKAGE FISSILE CONTENT	INFINITE ARRAY K_{eff}
5 bottles type A	0.770 ± 0.004
5 bottles type B	0.583 ± 0.004
5 bottles type C	0.517 ± 0.004

The rule of fractions was checked by calculation for loadings of five bottles of two types of fissile material (type A and type B). For this case the rule of fractions K_{eff} is given by:

$$K_{eff} = \frac{N_A}{5} 0.770 + \frac{N_B}{5} 0.583 \tag{2}$$

where N_A and N_B are the number of bottles of type A and type B, and the calculated values of K_{eff} are shown. These results show excellent agreement between the two methods, and provide reasonable assurance that a specified margin in K_{eff} is preserved. This is an example of the use of the rule of fractions for heterogeneous mixtures of fissile materials in geometrically same containers. The good agreement between calculation and rule of fractions appears to be as much dependent on the geometry as on the fissile material content.

Savannah River Laboratory (SRL) Method B

This particular method appeared in the SRL Criticality Control Procedure Manual as a method for maintaining criticality safety in mixtures of fissile/fissionable isotopes. The method starts by rewriting equation 1 in terms of a quantity Q_i defined by:

$$Q_i = 1/A_i \tag{3}$$

Equation 1 then becomes:

$$\sum_i a_i Q_i \leq 1 \tag{1A}$$

A table of values of Q_i are provided for each isotope i, hence satisfying equation lA assures criticality safety. The Q_i are defined primarily from the American National Standards Institute (ANSI)/American Nuclear Society (ANS) 8.1, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, and ANSI/ANS 8.15, *Nuclear Criticality Control of Special Actinide Elements*, standards by taking the mass limits from the standards as the A_i and rounding the resulting Q_i upward.

Isotopic Equivalents

A variety of equivalency expressions abound in the literature that express quantities of fissile/fissionable isotopes that are critically equivalent to another fissile/fissionable isotope. Such expressions are subject to the external conditions and many times are stated specifically for one condition only.

To express the combined mass of several fissile isotopes as an equivalent amount of ^{235}U one might proceed as follows. First write equation 1 as an equality because exact equivalency that ensures a specified margin in K_{eff} is the desired result. Define the a_i as the actual quantity of isotope i present and the A_i as the limiting quantities of isotope i that ensures a known margin in K_{eff}. Now write equation 1 as:

$$\sum_i \frac{'a_i}{A_i} = 1 - \frac{a_{235}}{A_{235}} = \frac{A_{235} - a_{235}}{A_{235}} \tag{4}$$

where the prime on the summation indicates the ^{235}U term was removed. Multiplying by A_{235} then yields the form:

$$A_{235}\text{-}a_{235} = \sum_i{}' a_i \frac{A_{235}}{A_i} \tag{4A}$$

The ratio, of A_{235}/A_i can now be defined from accepted mass quantities that ensure a specified K_{eff} margin, or simply taken from ANSI/ANS 8.1 or 8.15. The statement that equation 4A makes is that any quantity of ^{235}U mass in a mixture not explicitly present is equivalent to the sum of masses of isotopes weighted by the terms on the right side of equation 4A. Another way to state this is that the equivalent mass of ^{235}U represented by mass a_i is defined by:

$$\text{Equivalent } ^{235}\text{U mass} = a_i \frac{A_{235}}{A_i} \tag{5}$$

Equation 5 more often appears with the summation from the right side of equation 4A on its right side so that several isotopes may appear in the isotopic mixture.

The limitations of equivalency relationships in the form of equations 4A or 5 are the following:

- All A_i must be defined for the same environmental conditions.
- These equations assume that all isotopes behave the same no matter how the spectra is changed by the presence of other isotopes.

e. Discuss the concept of contingencies for checking the validity of criticality safety limits.

DOE-STD-3007-93, *DOE Standard, Guidelines for Preparing Criticality Safety Evaluations at Department of Energy Non-Reactor Nuclear Facilities* defines a contingency as a possible but unlikely change in a condition/control important to the nuclear criticality safety (NCS) of a fissionable material operation that would, if it occurred, reduce the number of barriers (either administrative or physical) that are intended to prevent an accidental nuclear criticality.

The following is derived from NCSET module 12.

The ultimate product of a criticality safety evaluation (CSE) is a document that prescribes the criticality safety controls and limits for an operation with fissionable material such that the operation is subcritical under all normal and credible abnormal conditions.

Seven steps should be performed to meet the objectives of a CSE. Two of these steps are directly related to contingencies:

- Identification of contingencies that could affect NCS
- Analysis of normal operations and contingencies

These steps are not independent and often overlap as the CSE is being developed. Normally, some of the contingency analysis is done at the same time that normal operations are analyzed, since operational limits will sometimes have to be modified based

on the contingency analysis results. Every CSE must cover all of the operations within the scope of the evaluation and credible contingency scenarios taking into account all materials and configurations that are expected to be present.

4. Criticality safety personnel must demonstrate a working level knowledge of alarm systems for criticality accidents.

a. Define the following terms:
- **Criticality accident**
- **Minimum accident of concern**
- **Process area**

The following is taken from ANSI/ANS 8.3.

Criticality Accident

The release of energy as the result of the accidental production of a self-sustaining or divergent fission chain reaction is referred to as a criticality accident.

Minimum Accident of Concern

The smallest accident, in terms of fission yield and dose rate, that a criticality alarm system is required to detect.

Process Area

A process area is any area involved in the chemical processing, mechanical processing, handling, or storage of fissionable materials.

b. Discuss the general principles associated with the use of criticality alarm systems including the following:
- **Installation**
- **Coverage**
- **Detection**
- **Alarms**
- **Dependability**
- **Removal**

The following is taken from ANSI/ANS 8.3.

Installation

A criticality alarm system meeting the requirements of ANSI/ANS 8.3 shall be installed in areas where personnel would be subject to an excessive radiation dose. For this purpose, the maximum fission yield integrated over the duration of the accident may be assumed not to exceed 2.0×10^{19} fissions. The basis for a different maximum fission yield shall be documented. Installation of an alarm system implies a nontrivial risk of criticality.

Coverage

The need for criticality alarm systems shall be evaluated for all activities in which the inventory of fissionable materials in individual unrelated areas exceeds 700 g of ^{235}U, 500 g of ^{233}U, 450 g of ^{239}Pu, or 450 g of any combination of these three isotopes. For operations involving significant quantities of other fissionable isotopes, this evaluation

shall be made whenever quantities exceed the subcritical mass limits specified in ANSI/ANS 8.15. Also this evaluation shall be made for all processes in which neutron moderators or reflectors more effective than water are present, or unique material configurations exist such that critical mass requirements may be less than the typical subcritical mass limits.

For this evaluation, individual areas may be considered unrelated when the boundaries between the areas are such that there can be no uncontrolled transfer of materials between areas, the minimum separation between material in adjacent areas is 10 cm, and the areal density of fissile material averaged over each individual area is less than 50 g/m^2. This stipulation is applicable only to the three specific isotopes noted.

In areas in which criticality alarm coverage is required, a means shall be provided to detect a criticality accident and to signal that prompt protective action is required.

Detection

Criticality alarm systems shall be designed to respond immediately to the minimum accident of concern. For this purpose, in areas where material is handled or processed with only nominal shielding, the minimum accident may be assumed to deliver the equivalent of an absorbed dose rate in free air of 0.2 Gy/min (20 rad/min) at 2 meters from the reacting material. The basis for a different minimum accident of concern shall be documented.

Alarms

Criticality alarm signals shall be for prompt evacuation or other protective actions. The criticality alarm signals should be uniform throughout the system. The signals shall be distinctive from other signals or alarms that require a response different from that necessary in the event of a criticality accident.

The signal generators shall be automatically and promptly actuated upon detection of a criticality accident.

After actuation, the signal generators shall continue to function as required by emergency procedures, even if the radiation falls below the alarm point. Manual resets, with limited access, should be provided outside areas that require evacuation.

A means for manual actuation of the criticality alarm signal may be provided.

For all occupied areas where personnel protective action is required in the event of criticality accident detection, the number and placement of criticality alarm signal generators shall be such that the signals are adequate to notify personnel promptly throughout those areas.

The audio generators should produce an overall sound pressure level of at least 75 decibel (dB), but not less than 10 dB above the maximum ambient noise level typical of each area for which audio coverage is provided.

In areas with very high audio background or mandatory hearing protection, visual signals or other alarm means should be considered.

Dependability
Consideration shall be given to the avoidance of false alarms. This may be accomplished by providing reliable single detector channels or by requiring concurrent response of two or more detectors to initiate the alarm. In redundant systems, failure of any single channel shall not prevent compliance with the detection criterion discussed previously.

Portable instruments may be used in special situations to augment an installed criticality accident alarm system (CASS). Examples of such situations include alarm system maintenance or testing, evacuation drills, activities in areas not normally occupied by personnel, or other special operations.

Removal
The following is taken from DOE-STD-3007-2007.

Removal of CAASs, whether fixed, permanent, temporary, portable, or transportable systems, from existing facilities should be justified with a CSE evaluating the need for a CAAS in terms of the overall risk benefit of such a system. For a CAAS removal evaluation, an important part of a strong argument is a thorough facility characterization detailing the quantity, form, and distribution of fissionable material in the facility. In order to support an assertion that the CAAS is not required, the potential holdup in a facility shall be addressed. New facilities that previously have not processed or handled fissionable materials obviously have no holdup. But holdup in older facilities shall be addressed. Utilization of operating personnel or facility experts with direct knowledge of operations spanning the full life cycle of the facility is important. When personnel with direct knowledge of past operations are not available, documentation relevant to the facility operations and off-normal events should be used. A thorough characterization should include

- a description of the operating history of the facility sufficient to support conclusions about the presence or absence of fissionable materials in various locations;
- a description of accidents or process upsets, particularly those that might have left significant quantities of fissionable materials in unexpected locations; and
- a description of current material inventories, including all accountable fissionable material, inventory differences, and comprehensive fissionable material assays.

The characterization should also include a brief description of assay methods used, their accuracy, potential weaknesses, comprehensiveness of the assays, and the meaning of any stated uncertainties.

c. Discuss the requirements for testing the criticality alarm system.

The following is taken from ANSI/ANS 8.3.

Initial tests, inspections, and checks of the system shall verify that the fabrication and installation were made according to design plans and specifications.

Following modifications or repairs, or events that call the system performance into question, there shall be tests and inspections adequate to demonstrate system operability.

System response to radiation shall be measured periodically to confirm continuing instrument performance. The test interval should be determined on the basis of experience. In the absence of experience, tests should be performed at least monthly. Records of tests shall be maintained. System designs may incorporate self-checking features to automated portions of this testing.

The entire alarm system shall be tested periodically. Each signal generator should be tested at least annually. Field observations shall establish that criticality alarm signals are functional throughout all areas where personnel could be subject to an excessive radiation dose. All personnel in affected areas shall be notified before testing of the criticality alarm signals.

When tests reveal inadequate performance, corrective action shall be taken without unnecessary delay. If portable instrument use is required, the criteria of section 4.4.2 of ANSI/ANS 8.3 shall be met.

Procedures for system testing shall minimize false alarms and inadvertent initiation of emergency response. The procedures shall also require that the systems be returned to normal operation immediately following tests.

5. Criticality safety personnel must demonstrate a working level knowledge of neutron absorbers.

a. Describe the use of neutron poisons. (See competency 19, ANSI/ANS 8.14 and 8.21.)

The following is taken from DOE-HDBK-1019/2-93.

Fixed Burnable Poisons

During operation of a reactor the amount of fuel contained in the core constantly decreases. If the reactor is to operate for a long period of time, fuel in excess of that needed for exact criticality must be added when the reactor is built. The positive reactivity due to the excess fuel must be balanced with negative reactivity from neutron-absorbing material. Moveable control rods containing neutron-absorbing material are one method used to offset the excess fuel. Using control rods alone to balance the excess reactivity may be undesirable or impractical for several reasons. One reason for a particular core design may be that there is physically insufficient room for the control rods and their large mechanisms.

To control large amounts of excess fuel without adding additional control rods, burnable poisons are loaded into the core. Burnable poisons are materials that have a high neutron absorption cross section that are converted into materials of relatively low absorption cross section as the result of neutron absorption. Due to the burnup of the poison material, the negative reactivity of the burnable poison decreases over core life. Ideally, these poisons should decrease their negative reactivity at the same rate the fuel's excess positive reactivity is depleted. Fixed burnable poisons are generally used in the form of compounds of boron or gadolinium that are shaped into separate lattice pins or plates, or introduced as additives to the fuel. Since they can usually be distributed more uniformly than control rods, these poisons are less disruptive to the core power distribution.

Soluble Poisons

Soluble poisons, also called chemical shim, produce a spatially uniform neutron absorption when dissolved in the water coolant. The most common soluble poison in commercial pressurized water reactors (PWR) is boric acid, which is often referred to as "soluble boron," or simply "solbor." The boric acid in the coolant decreases the thermal utilization factor, causing a decrease in reactivity. By varying the concentration of boric acid in the coolant (a process referred to as boration and dilution), the reactivity of the core can be easily varied. If the boron concentration is increased, the coolant/moderator absorbs more neutrons, adding negative reactivity. If the boron concentration is reduced (dilution), positive reactivity is added. The changing of boron concentration in a PWR is a slow process and is used primarily to compensate for fuel burnout or poison buildup. The variation in boron concentration allows control rod use to be minimized, which results in a flatter flux profile over the core than can be produced by rod insertion. The flatter flux profile is due to the fact that there are no regions of depressed flux like those that would be produced in the vicinity of inserted control rods.

DOE reactors typically do not use soluble neutron poisons during normal operation. Some DOE reactors do, however, include emergency shutdown systems that inject solutions containing neutron poisons into the system that circulates reactor coolant. Various solutions, including sodium polyborate and gadolinium nitrate, are used.

Fixed burnable poisons possess some advantages over chemical shim. Fixed burnable poisons may be discretely loaded in specific locations in order to shape or control flux profiles in the core. Also, fixed burnable poisons do not make the moderator temperature reactivity coefficient less negative as chemical shim does. With chemical shim, as temperature rises and the moderator expands, some moderator is pushed out of the active core area. Boron is also moved out, and this has a positive effect on reactivity. This property of chemical shim limits the allowable boron concentration because any greater concentration makes the moderator temperature coefficient of reactivity positive.

Non-Burnable Poisons

A non-burnable poison is one that maintains a constant negative reactivity worth over the life of the core. While no neutron poison is strictly non-burnable, certain materials can be treated as non-burnable poisons under certain conditions. One example is hafnium. The removal (by absorption of neutrons) of one isotope of hafnium leads to the production of another neutron absorber, and continues through a chain of five absorbers. This absorption chain results in a long-lived burnable poison which approximates non-burnable characteristics. Absorbers with low neutron absorption cross sections can also be treated as non-burnable under most conditions.

It is possible to make the reactivity of a poison material that is usually a burnable poison more uniform over core life through the use of self-shielding. In self-shielding, the poison material is thick enough that only the outer layer of the poison is exposed to the neutron flux. The absorptions that take place in the outer layers reduce the number of neutrons that penetrate to the inner material. As the outer layers of poison absorb neutrons and are converted to non-poison materials, the inner layers begin absorbing more neutrons, and the negative reactivity of the poison is fairly uniform.

The normal use of fixed non-burnable poisons is in power shaping, or to prevent excessive flux and power peaking near moderator regions of the reactor.

b. Explain the absorption characteristics of the following elements in terms of their cross-sections.
- **Cadmium**
- **Boron**
- **Chlorine**
- **Hydrogen**

The following was derived from information in AMSI/ANS 8.14 and 8.21 by a team of subject matter experts at the NNSA Service Center. Refer to the cross section plots for these elements in appendix 1 of this reference guide for additional detail.

Cadmium

3000 barns at thermal, large resonance band. The cadmium cross section is about 3 times that of the fission cross section for ^{235}U and ^{239}Pu, but drops below the fission cross sections through the resonance region. The resonance regions approximately overlap. The cadmium absorption cross section is about a tenth that of boron above the resonance region.

Boron

4000 barns at thermal, 1/v to about 1 MeV, then some resonance above 10^6 eV. Boron has an absorption cross section about 5 times as large as the fission cross sections in the thermal region. The boron cross section values pass approximately through the middle of the fission cross section resonance region for ^{235}U and ^{239}Pu, and are much smaller than the fission cross sections in the fast fission region.

Chlorine

50 barns at thermal, some resonance. Chlorine has an absorption cross section about 1/14 that of the fission cross section for ^{235}U and ^{239}Pu. Chlorine is an effective poison, but if used in calculations, the validation must be addressed carefully. The resonance parameters from ENDF-5 and previous were not well characterized. The chlorine resonance region is between 100 and 10^6 eV, while the fissile isotopes ^{235}U and ^{239}Pu have resonances between 1 and 10^5 eV.

Hydrogen

¼ barn at thermal, no resonance, mostly 1/v. Hydrogen has an absorption cross section about 1/2300 of the fission cross section for ^{235}U and ^{239}Pu below the resonance region. Hydrogen becomes an effective absorber in dilute systems ($H/X \geq \sim 3700$) because of the large number of hydrogen atoms.

c. Explain the purpose and use of Raschig Rings as a neutron poison. (See competency 19, ANSI/ANS 8.5.)

The following is taken from Oak Ridge Associated Universities (ORAU), Raschig Rings for Criticality Control, 1999.

Raschig rings are made of borosilicate glass. They are approximately 1 5/8 inch high and 1 ½ inch in diameter, and their purpose is to prevent an accidental criticality by absorbing neutrons. To be specific, it is the boron-10 (^{10}B) in the glass that absorbs the neutrons. The rings would normally be used to fill a container that under accidental circumstances might contain a solution with enough ^{239}Pu, ^{235}U or ^{233}U to go critical. Ideally, just the geometry (shape and size) of such a container would prevent a criticality, but mistakes happen and these solutions can end up in the wrong type of container. Raschig rings provide an additional level of safety. For example, an overflow tank from a floor drain in a room handling highly enriched U that should never receive enriched U is filled with rings just in case.

The shape of the rings is such that it maximizes their surface area while maintaining the maximum free space in the container. Because the rings might settle out over time, a periodic inspection is performed to ensure that the rings are still packed to the top of the containers. In addition, a few rings are removed, perhaps once per year, and the boron content confirmed. Over time, the ^{10}B content can be depleted due to neutron absorption. To prevent breakage, the glass is particularly hard.

The detailed requirements for the use of borosilicate-glass Raschig rings are found in ANSI/ANS-8.5-1996.

6. Criticality safety personnel must demonstrate a familiarity level knowledge of the functional interfaces between safety system software and control components and the system level design and function.

a. Identify how system level requirements are developed. Explain how these requirements are incorporated into an engineered system. Describe the methods a quality organization should use to verify that the "as installed" system meets the system level requirements as defined.

The following is taken from American national Standards Institute/Institute of Electrical and Electronics Engineers ANSI/IEEE 830.

System Level Requirements

The performance or functional requirements of the process system are defined first. From these, the control requirements can be developed. Once the controlled parameters and items are identified, control elements and methods can be identified and selected. This must include the sensor or detector response time in the system, the lag time for the control system (analog, digital, or manual) to respond to the detector and signal the control element, the lag time for the control element to operate, and the system response to action by the control element. These factors influence the type of control system selected. Digital systems have the advantage that the mathematical control algorithms can be changed without major changes to the hardware. Digital systems can also be used to provide software interlocks without having to route additional wiring and sensors or add relays.

Once the control requirements and methods are established, the control elements can be assigned to the process control points. The measurement methods may also impact the lag storage required for process material. As an example, if the key parameter, say impurity levels, can only be measured with a 24–hour response time, a day or two lag storage is

needed, and even the fastest digital control system is only an operator aid (i.e., the control system holds until the operator tells it to do something, for example, the operator re-aligns the valves).

To use automated control, a logic diagram is required for all but the simplest control systems. This will include sensor input, operator input, control and process modeling, and sensor output. In a digital system, the hardware either senses, acts, or amplifies and carries signal. The software responds to sensors and operator input, and displays status to the operator(s) and sends signals to control elements. The system-level requirements will identify those functions that must occur automatically, the alarms needed for operator intervention, and the condition the system should assume in an alarm condition. Failure modes of components must also be identified and perhaps specified (e.g., does a valve fail open, closed, or last on loss of power or loss of air?).

System-level requirements will drive the functional software requirements specifications. These will include

- performance of the intended control and alarm functions as identified from the process or system requirements;
- detection of incorrect hardware operation, and response to the detected condition (e.g., supervised circuits, flow verification after a valve opens);
- integrity, security, and configuration control issues (e.g., attempted incorrect operation, failure of operator to respond to alarm condition);
- human interface considerations (e.g., deadman circuits, display requirements, redundant input or "two key" requirements);
- system response to fault conditions (e.g., voting logic, and go to safe shutdown if independent systems disagree); and
- the ability to tolerate, detect, and recover, or safely shut down from fault or abnormal conditions.

Safety system software in this context is an integral part of the control system, and must be included in the hazards analysis and detailed safety analysis.

Quality Verification

The following is taken from DOE G 200.1-1, chapter 9.

Ensure the integrity and quality of the installed software product by executing the installation tests defined in the installation plan. Testing is performed to verify that the software product has been properly installed and is fully operational.

The installation test(s) is designed to validate all functions of the software product and should specify a standard set of test results and tolerances. If the software product being installed is a modification to an existing system, all remaining functions and code that may be affected by the new software should be tested.

Document any problems and identify corrective action. Select a diagnostic package that will pinpoint problems quickly and allow for timely corrections.

Retest all equipment and software after a repair, replacement, or modification. Certify each software component on successful completion of installation and checkout. When

installation is complete, rerun a portion or all of the system test and dry-run the acceptance test procedures to verify correct operation of the software product.

Conduct installation testing to verify the following:

- Security functions
- Integration with the current software
- Interfaces with other systems
- System functionality based on the requirements

b. Describe and discuss the advantages and disadvantages of the following automation approaches:

- **Analog control systems**
- **Hard wired relay logic**
- **Programmable Logic Controller (PLC) based systems**
- **Computer control systems**

Analog Control Systems

The following is taken from Cummins Power Generation.

Analog devices in power system control are discrete components (resistors, switches, capacitors, coils and relays) that coordinate input and output signals, and perform rudimentary logic for specific control functions. Adjustments to the system usually involve a physical adjustment such as increasing or decreasing the resistance of a variable resistor or substituting modules. Analog control hardware communicates system status and fault conditions with indicator lights, analog meters, or alarms.

In modern generating systems, there may be more than 200 typical alarm conditions having to do with the load, the utility, other paralleled generators, the engine or the alternator. As the number of potential status or alarm conditions has grown, analog control systems have not been able to communicate this information to operators effectively.

Hard Wired Relay Logic

The following is taken from University of Alabama, Logic Control Systems.

One of the disadvantages of the relay logic systems is the difficult nature of the programming. The program logic is hard wired by the interconnection of the relays, limit switches, timers, counters, etc. Changing the task performed requires physically moving the wires from the relays and limit switches and placing them in the desired new configuration. For circuits with only three or four components this is not difficult. However, systems containing ten to several hundred individual components are not uncommon in industrial automation systems.

Programmable Logic Controller (PLC) Based Systems

The following is taken from University of Alabama, Logic Control Systems.

The PLC was developed in the early 60's to overcome the deficiencies of relay logic systems. Programmable logic is implemented using a microcomputer instead of the hard-wired logic of the conventional hard-wired relay system.

The major advantage of PLCs is that the programming can be done in ladder logic, just like relay logic systems. Electricians and technicians can readily adapt to this familiar type of programming. A computer language like BASIC or Pascal might be too intimidating and is not required to implement straightforward machine logic.

Computer Control Systems

The following is taken from Cummins Power Generation: Computer Controls in Automation Systems.

New standards have recently been adopted in the United States, which clear the way for computer controls to be used for safety functions in automation systems. This means instead of a chain of wires connecting electromechanical switches, sophisticated computer controls can now be utilized in functional safety systems and can provide up-to-the-second data that will reduce downtime, catch problems before they occur and improve overall employee safety.

The use of programmable electronic equipment in safety applications delivers a high level of safety, reliability and quality, while allowing the user to better monitor processes. Because of the advantages offered by real-time monitoring, in many cases computer-controlled safety systems can actually increase productivity by allowing the user to catch problems before they become major enough to force a work stoppage or, in a worse case scenario, cause injuries.

In the past, most safety applications required hard-wired, push-button-type safety stops. For example, if an assembly line encountered a problem, a button would be pushed and the line would be shut down for safety inspection. If a sensor detected a problem, it might also trigger a shut-down and then the line would need to be inspected to determine the cause of the problem. Downtime could last minutes or days, depending on the severity of the problem. This type of situation rarely would occur with a computer-controlled functional safety system.

Computer-controlled systems allow the user to have ongoing diagnostic data on safety systems so operators can monitor machines and see where problems are developing. In many cases they can correct the problem, while the system remains online. Often, the amount of time that a machine needs to be shut down is reduced to minutes, because the user already has data in hand before shutting the machine down. The new computer-controlled systems just simply allow companies to have a better insight into how their equipment is operating and that data gives them the power to be more productive and have a safe workplace.

There are a number of industries that can benefit from computer-controlled safety systems. Virtually any company that uses industrial machinery or automatic production lines, such as robots, can realize significant gains. But any industry that depends on safety systems, such as the chemical industry, oil and gas companies, or nuclear power generators, can benefit from safety automation technology. Under consideration is the adaptation of computer-controlled safety systems in furnaces, power stations, elevators, radio-controlled hoists, or locomotives. Anywhere safety is a concern, computer-controlled systems can

provide the same or higher levels of safety compared to traditional electromechanical systems.

c. Discuss the limitations and pitfalls of automation as it relates to criticality safety. Identify areas that are appropriate to automate and areas where automation might be a detriment to safety.

The following is taken from Automation of System Safety Analysis: Possibilities and Pitfalls.

The benefits that can be expected from automating safety analyses are initially a result of the speed, consistency, reliability and accuracy of search and analysis algorithms.

At present, commercial tools are available to support many safety analysis techniques. However, their capabilities are usually limited to improving consistency in tasks that would be tedious and error-prone to implement manually, especially the application of mathematical methods, the storage, searching, and reliable retrieval of safety information, and neatly formatted presentation of results. Tools have traditionally been dedicated to one single analysis method. Tool set integration has tended to focus on the sharing of safety information, for example ensuring the consistency of data used in different views and analyses of an artifact. Some approaches provide computer-based support for safety argument structures, using these to integrate disparate data and tool-based results.

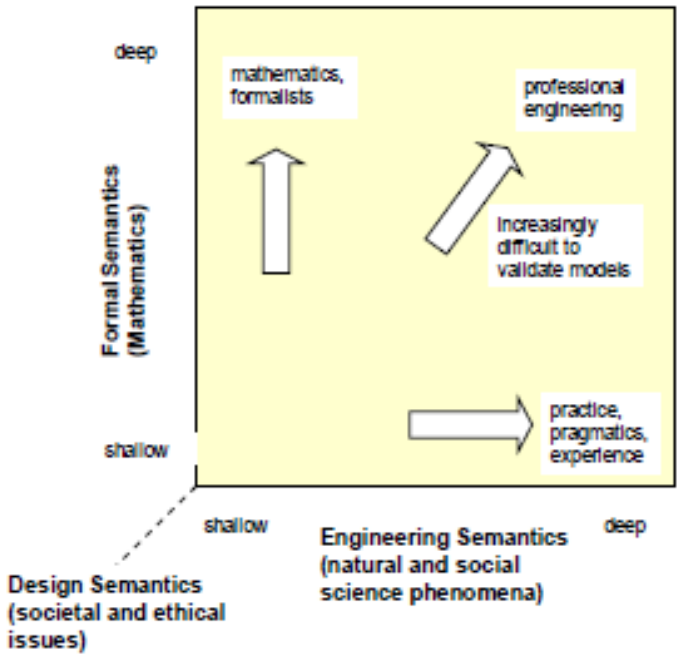
Few, if any, of the current tools automate the actual analysis; most require the safety engineer to manually identify hazards, failure modes, and failure effect propagation paths. In addition, many current tools require the criticality safety engineer to build a new model of the system and it is in supplying this information that the valuable and creative part of the safety engineer's role lies.

The use of system modeling in safety-related design and fault diagnosis has been a consistent strand of research over many years, involving different research communities. The key issue in automating the actual safety assessment, using models from the core design process, is how to replace the knowledge and insight contributed by the safety engineer. The most common approaches being investigated are either to explicitly extend the system design model to include faults and failures, or to construct a separate database of models of components, including their failure modes, and use a tool to compose these component models to give a complete system model. The disadvantage of the first approach is the complexity that it adds to the main system design model; the second approach avoids this, but there are issues of ensuring that the composed model accurately reflects true system behavior

One strand of research, pursued in the software engineering community, focuses on the use of formal modeling, where the term "formal" means a representation based in mathematics, and amenable to the construction of mathematically rigorous proofs. This approach depends on there being a common mathematical framework within which the specification (problem) and the designed artifact (solution) can both be expressed. Thus it is feasible to use discrete mathematics to formally prove that a software component is consistent with its specification, because of the logical nature of software.

Constructing a formal model often leads to greater understanding and precision of thought about a design or analysis problem, supports collaborative work, and opens up the possibility of mathematical techniques being deployed to solve the problem. There are limitations to mathematically formal approaches at the boundaries of these models, particularly at the interfaces with actuators and sensors and at the implementation of code in physical devices. However, work in this field is primarily concerned with fundamental questions about the role of mathematical formalisms in product development, questions which lie at the heart of analysis automation.

As figure 4 illustrates, the formal semantics of a model are different from its "engineering semantics—the correspondence between the model and the real physical or social phenomena that are being represented. Extending the depth and scope of a formal model increases the difficulty of model validation, and it becomes progressively more difficult to develop trust in more complex formal models.



Source: Automation of System Safety Analysis: Possibilities and Pitfalls

Figure 4. Three areas of meaning in engineering

The application of sophisticated mathematical techniques to design and analysis problems in such a way that deep engineering semantics are preserved is the hallmark of professional engineering. In traditional engineering fields, the development of trust in modeling approaches and tools is developed within a specialty over many years. Model validation is achieved gradually by repeated comparison between model predictions and observation.

Another area of meaning in engineering relates to the societal/ethical issues in product development; safety engineering, in common with other specialties, has its roots in ethical value judgments.

In the following paragraphs, the prospects for automation of safety analysis are explored. Systems and software are treated separately because the issues are slightly different in the two domains.

Automation in Systems Safety Analysis

The systems engineering field is characterized by having to manage disparate technologies and engineering specialties. System models are traditionally used to express designs and support trade-off studies between alternative architectures. Two different styles of models are usually identified: 1) models of specific properties or views of a system (e.g., reliability block diagrams, control loop models), and 2) integrating models, which represent global data about the design and ensure consistency between different views.

The use of these models for safety analysis presents potential benefits and drawbacks. If the design models are developed as part of the core design effort, additional safety-related modeling costs are minimized. There is potential for systematic and complete exploration of the safety-related behavior of the system as designed, and of the consequences of specified failure modes. The analysis results will be consistent and complete within the scope of the product models and failure modes represented. However, it is easier to base safety analyses upon the specific property (type 1) models than the integrating models (type 2), and there is again the risk that important safety properties will be missed because they are (cannot be) represented in the model.

We can distinguish three "boundaries" (sets of assumptions) about the scope of a system model and the associated failure modes:

- Assumptions about component failure modes. For example, the possibility of components failing in ways that have not previously been encountered may not be considered.
- Assumptions that the effects of component failures fall within the range of phenomena covered by the product model. For example, a product model based solely on electrical circuit theory would not capture the consequence of a component failure (such as overheating) that lies outside the electrical circuit domain.
- Assumptions about the safety implications for the total product of failure effects identified at the interface of the modeled system.

These assumptions are related to the engineering semantics of the models. A further general concern is the additional complexity that arises in dynamic systems, that is, systems which cannot be adequately represented by static structures. For example, finite-state machines are widely used to model the switching behavior of complex components. However, the behaviors exhibited by systems with dynamic feedback require different techniques. The field of systems engineering has developed the concept of "emergent property" to describe the non-obvious behavior of a system that arises as a result of the interaction of its components. It seems necessary for the safety engineer to assess different levels of aggregation of a system in order uncover "emergent" failure modes, that is failure modes that would not be exposed by considering component failure modes and their effects propagated within models of design intent.

Automation in Software Safety Analysis

Formal representations are relatively more powerful in software engineering because of the abstract nature of software. Many discrete formalisms have been investigated to support software development, including set theory, graph theory and group theory. Continuous formalisms found in other branches of engineering (differential calculus) have been approximated with discrete formalisms.

Software failures are systematic. Assuming the integrity of the hardware: in a deterministic system, the probability of a hazardous response is either zero—if the software has been designed to avoid it—or equivalent to the probability of the environmental conditions arising under which the software exhibits it.

Thus, if our probabilistic causal model factorizes out probabilities concerning the hardware on which the software runs, and the integrity of the process (e.g., analysis tools, correct compilers correct, product model and source code consistent, etc.) then one possible use of formalism is to predict whether the software exhibits a hazard. If so, an even more valuable use of formalism might be to predict which environmental conditions give rise to the software exhibiting such a hazard.

Formal descriptions of products and safety properties may also be used to derive test cases. However, testing cannot prove the absence of errors and placing an accurate probabilistic estimate of failure rates based on testing can be problematic.

There are significant practical problems associated with formalizing hazards so that they can be analyzed. Different formal methods emphasize different views of the system. For example, model-based methods emphasize the data and step operations on that data that the system may take. Other methods emphasize the valid sequences of data states that system may evolve through; or distinguish the patterns of communication a distributed system may exhibit. Each method comes with its own style of reasoning and effectively sets the agenda for the kinds of safety properties which may be analyzed. These may not necessarily be the kinds of safety properties the safety engineer wishes to express.

For example, a product model written in one method might describe the step-wise ways in which the data of the system can be altered, in terms of a set of inputs and producing a set of outputs. The kinds of reasoning classically available to the practitioner allows them to check whether an invariant holds over the system state, or perhaps to show whether an operation is a refinement of some abstract “safe” operation. However, the safety engineer might be interested in ensuring that a particular differential relationship on the system outputs holds. Moreover, they might want to derive the differential constraint on the input variables required to guarantee some differential constraint in the outputs. It is not easy to see how to do this given the classical application of the method.

Even if the engineer can express the safety property of interest within the chosen formalism, there are limitations to what can be achieved with the product models themselves. Safety critical systems are often complex systems. They can involve data structures, operations on those structures which must be implemented in code, asynchrony and communication, and real-time constraints. The older formal methods tend to focus on

one particular viewpoint making representation and reasoning about complex behavior extremely difficult. There are two extremes:

- A formal model is constructed which tries to encompass the full complexity of the system.
- Several formal models are constructed, each representing different facets of the system.

The first approach's main drawback is the lack of precedent. There are not many examples in the literature of formal models embracing many different viewpoints (e.g. time, data, communication). Another problem could well be the tractability of the resulting model when it comes to analysis. A complex model will almost certainly involve a large state-space, probably too large for conventional model-checking limitations. If theorem proving is preferred, there is still likely to be a large axiom base describing the behavior of the system under each combination of viewpoints. This may make the already expensive theorem proving approach infeasible.

The second approach also has its drawbacks. Firstly, different viewpoints are not necessarily independent of each other. For instance, communication patterns can depend on the data in the system, or the time taken for some event to occur. One must be careful not to "over-abstract" when producing different formal viewpoints of the system, as the resulting model may not contain the information necessary to verify a safety property. Secondly, there is a rather extreme validation problem. How does one know that each formal model faithfully captures some abstraction of the system—especially when we do not understand what that abstraction is in formal terms. Indeed, when each model is not necessarily based on the same formalism, how can we even be sure the models are consistent with each other?

Automation: Challenges and Potential Pitfalls

The automated safety analysis of a design has to be based on two sources of explicit data;: 1) a design model or description, held in computer-processable form, and 2) data and algorithms which can be applied to the design data and which implement a safety analysis. The quality and completeness of the safety analysis is limited by the quality of the design data and of the formalized safety information. This situation can be summarized by characterizing the design and tool data as existing in a virtual world; validating such models against real world phenomena is a key responsibility of the safety engineer.

Validating virtual-world models and tools against real phenomena is problematic because the real world is informal; there is no one-off test to demonstrate "correctness". Instead, trust has to be established over time, with repeated use and demonstration of consistency between modeled and real-world outcomes. Trust is easier to establish in generic applications, where standardization and wide use establishes trust within a community of practitioners. It is more difficult to build trust in specific tools and models. The more formally complex (the deeper the formal semantics) embodied in a tool, the more difficult is the validation task against real world phenomena. The rate of change in an application domain compared with the rate of building of trust in a model is a factor in the usability of automation in a particular setting. A safety engineer may trust his/her engineering judgment (based on "engineering semantics" more readily than a tool which, although formal and rigorous, is less familiar) and where the process by which results are generated

is unclear. Visibility of (and participation in) tool development are perhaps the best ways to build trust, but are rarely practical strategies.

Tools cannot at present help with "fundamental safety"—what we might call safety by design. The system model, which is formally represented and subject to automated analysis, is an input. To consider automating design decisions as a result of safety analysis raises many issues, and is not feasible today.

Automating an analysis does not remove problems which often arise in the application of tools. Typical adverse behaviors include

- optimization for "best" result from tool, rather than real safety;
- use of tools just to be seen to be generating numbers, or to comply with externally imposed requirements;
- inappropriate use and/or belief in the outputs of tools; and
- over-reliance on tools and tendency not to assess the gaps and joins between formally supported aspects of a design.

A further problem with analysis tools is the generation of results which are difficult to interpret, possibly due to the volume of data generated and a lack of prioritization of importance or significance. Tools which assimilate data, highlight significant features and offer conclusions would be desirable provided the trust and validation concerns can be satisfied.

The demand for increased levels of automation in safety analysis can be interpreted in several ways. If the motivation is one only of efficiency gains and short-term cost reductions, we must be wary. Business managers may be placing the interests of the shareholder over those of the at-risk user of the system product. Tools vendors are interested in sustaining profitable businesses and do not carry responsibility for the appropriate use of their products.

The deployment of automation in criticality safety analysis focuses on the automatic analysis of product designs to derive safety properties. The possible gains from deploying automation in safety analysis include greater efficiency in the conduct of analysis, more systematic and complete application of techniques and analyses independent of particular interests or biases.

However, there are potential shortcomings of automated analyses. The validation of the system models and analytical methods embodied in tools is a major concern. Because real-world phenomena are informal, virtual-world models cannot be tested in a once-and-for-all manner against real physical systems. Instead, trust has to be developed over time and by a community of users. The applicability of automation depends on

- the generality/specificity of the models used;
- the rate of change of the modeled domain and the time needed to build confidence in a system model; and
- the feasibility of validating the safety–related assumptions in tools (e.g., postulated failure modes and failure effect propagation mechanisms) as well as the models of intended design.

Even with extended tool integration and automatic application to design models, the criticality safety engineer remains responsible for the outputs of the analysis and has to interpret them in the context of a wider safety analysis process. Such tools effectively hold and reapply past knowledge of product designs and component failure modes. The appropriateness of such knowledge for a current project is a "fitness-for-purpose" decision which the criticality safety engineer must make.

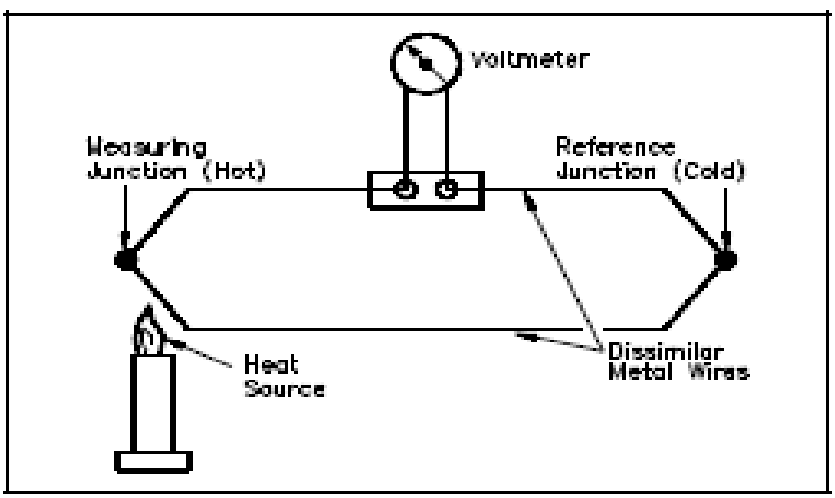
d. Describe the effect of the following on control of a process or experimental system:

- **Sensing elements, (e.g., thermocouples, position sensors, level sensors, flow sensors, pressure sensors, power level sensors)**
- **Control logic element (e.g., the hardware and/or software that actuates the control action elements)**
- **Control action element and control action (e.g., induction furnace power, resistance furnace voltage, cooling coil flow control, refrigeration unit, modulating valve position, block valve position, pump speed, control rod position, scram system action)**
- **Controlled system response to control action (e.g., change in temperature position, level, flow, pressure, power level)**

The following is taken from DOE-HDBK-1013/1-92.

Thermocouples

Thermocouples will cause an electric current to flow in the attached circuit when subjected to changes in temperature. The amount of current that will be produced is dependent on the temperature difference between the measurement and reference junction; the characteristics of the two metals used; and the characteristics of the attached circuit. Figure 5 illustrates a simple thermocouple circuit.



Source: DOE-HDBK-1013/1-92

Figure 5. Simple thermocouple circuit

Position Sensor

Position detection devices provide a method to determine the position of a valve or control rod. The four types of position indicators are limit switches, reed switches, potentiometer

valve position indicators, and linear variable differential transformers (LVDTs). Reed and limit switches act as on/off indicators to provide open and closed indications and control functions.

Reed switches can also be used to provide coarse, incremental position indication. Potentiometer and LVDT position indicators provide accurate indication of valve and rod position throughout their travel. In some applications, LVDTs can be used to indicate open and closed positions when small secondary windings are used at either end of the valve stem stroke.

The indicating and control circuitry provides for remote indication of valve or rod position and/or various control functions. Position indications vary from simple indications such as a light to meter indications showing exact position. Control functions are usually in the form of interlocks. Pump isolation valves are sometimes interlocked with the pump. In some applications, these interlocks act to prevent the pump from being started with the valves shut. The pump/valve interlocks can also be used to automatically turn off the pump if one of its isolation valves shut or to open a discharge valve at some time interval after the pump starts.

Valves are sometimes interlocked with each other. In some systems, two valves may be interlocked to prevent both of the valves from being opened at the same time. This feature is used to prevent undesirable system flow paths.

Control rod interlocks are normally used to prevent outward motion of certain rods unless certain conditions are met. One such interlock does not allow outward motion of control rods until the rods used to scram the reactor have been withdrawn to a predetermined height. This and all other rod interlocks ensure that the safety of the reactor remains intact.

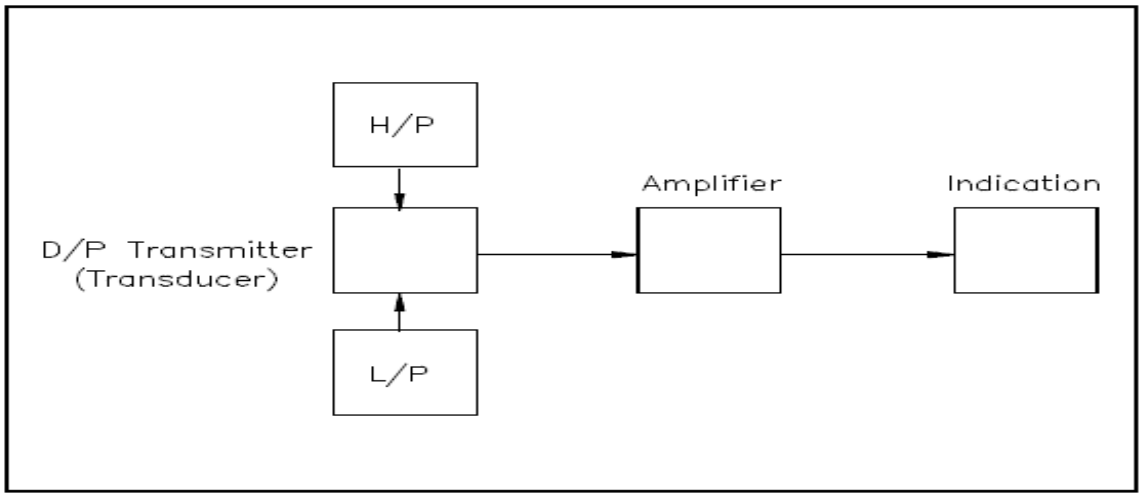
Level Sensors

Remote indication is necessary to provide transmittal of vital level information to a central location, such as the control room, where all level information can be coordinated and evaluated. There are three major reasons for utilizing remote level indication:

1. Level measurements may be taken at locations far from the main facility.
2. The level to be controlled may be a long distance from the point of control.
3. The level being measured may be in an unsafe/radioactive area.

Figure 6 illustrates a block diagram of a typical differential pressure (D/P) detector. It consists of a D/P transmitter (transducer), an amplifier, and level indication. The D/P transmitter consists of a diaphragm with the high pressure (H/P) and low pressure (L/P) inputs on opposite sides. As the D/P changes, the diaphragm will move.

The transducer changes this mechanical motion into an electrical signal. The electrical signal generated by the transducer is then amplified and passed on to the level indicator for level indication at a remote location. Using relays, this system provides alarms on high and low level. It may also provide control functions such as repositioning a valve and protective features such as tripping a pump.

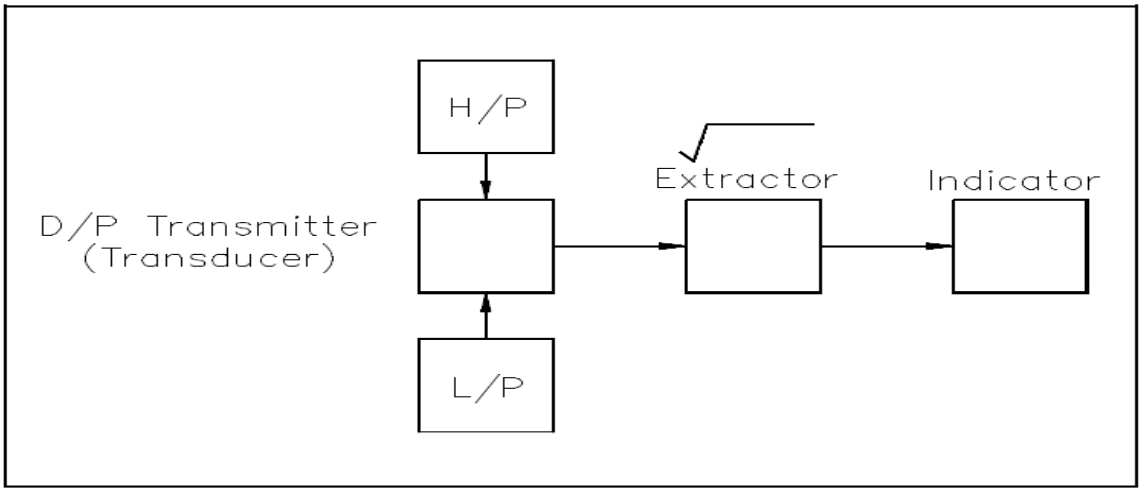


Source: DOE-HDBK-1013/1-2

Figure 6. Block diagram of a D/P level detection circuit

Flow Sensors

Figure 7 shows a block diagram of a typical D/P flow detection circuit. The D/P transmitter operation is dependent on the pressure difference across an orifice, venturi, or flow tube. This D/P is used to position a mechanical device such as a bellows. The bellows acts against spring pressure to reposition the core of a differential transformer. The transformer's output voltage on each of two secondary windings varies with a change in flow.

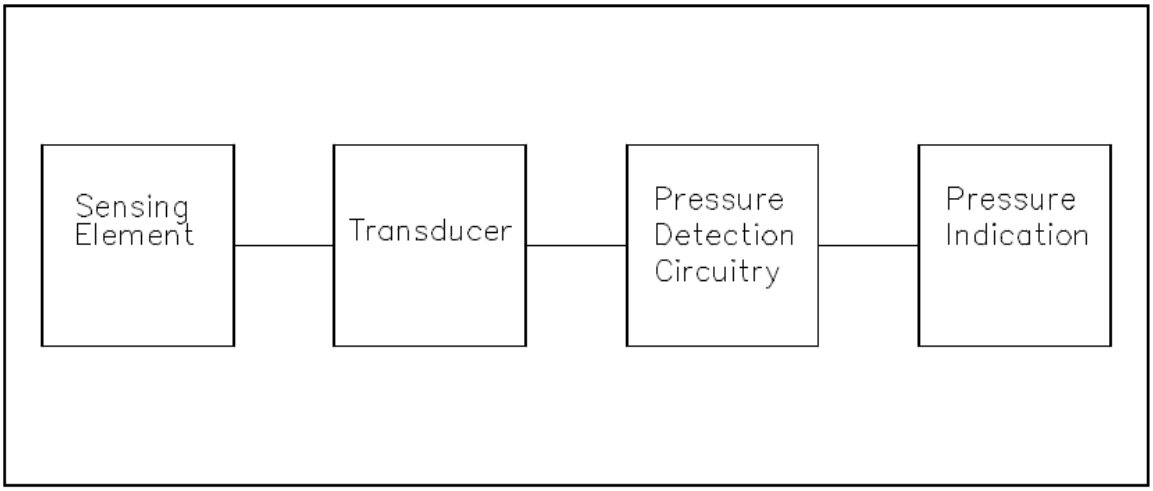


Source: DOE-HDBK-1013/1-92

Figure 7. Differential pressure flow detection block diagram

Pressure Sensors

Figure 8 shows a block diagram of a typical pressure detection circuit.



Source: DOE-HDBK-1013/1-92

Figure 8. Typical pressure detection block diagram

The sensing element senses the pressure of the monitored system and converts the pressure to a mechanical signal. The sensing element supplies the mechanical signal to a transducer, as discussed above. The transducer converts the mechanical signal to an electrical signal that is proportional to system pressure. If the mechanical signal from the sensing element is used directly, a transducer is not required and therefore not used. The detector circuitry will amplify and/or transmit this signal to the pressure indicator. The electrical signal generated by the detection circuitry is proportional to system pressure. The exact operation of detector circuitry depends upon the type of transducer used. The pressure indicator provides remote indication of the system pressure being measured.

Power Level Sensors

The following is taken from DOE-HDBK-1013/2-92.

A change in reactor power level can result in a change in reactivity if the power level change results in a change in system temperature.

The power level at which a reactor is producing enough energy to make up for the energy lost to ambient is commonly referred to as the point of adding heat. If a reactor is operating well below the point of adding heat, then variations in power level produce no measurable variations in temperature. At power levels above the point of adding heat, temperature varies with power level, and the reactivity changes will follow the convention for temperature variations.

The inherent stability and power-turning ability of a negative temperature coefficient are ineffective below the point of adding heat. If a power excursion is initiated from a very low power level, power will continue to rise unchecked until the point of adding heat is reached, and the subsequent temperature rise adds negative reactivity to slow, and turn,

the rise of reactor power. In this region, reactor safety is provided by automatic reactor shutdown systems and operator action.

Control Logic Element and Control Action Elements

The following is taken from DOE-HDBK-1013/2-92.

A control system is a system of integrated elements whose function is to maintain a process variable at a desired value or within a desired range of values. The control system monitors a process variable or variables, then causes some action to occur to maintain the desired system parameter. In a central heating unit, the system monitors the temperature of the house using a thermostat. When the temperature of the house drops to a preset value, the furnace turns on, providing a heat source. The temperature of the house increases until a switch in the thermostat causes the furnace to turn off.

Two terms which help define a control system are input and output. Control system input is the stimulus applied to a control system from an external source to produce a specified response from the control system. In the case of the central heating unit, the control system input is the temperature of the house as monitored by the thermostat.

Control system output is the actual response obtained from a control system. In the central heating unit, the temperature dropping to a preset value on the thermostat causes the furnace to turn on, providing heat to raise the temperature of the house.

In the case of nuclear facilities, the input and output are defined by the purpose of the control system. A knowledge of the input and output of the control system enables the components of the system to be identified. A control system may have more than one input or output.

Control systems are classified by the control action, which is the quantity responsible for activating the control system to produce the output. The two general classifications are open-loop and closed-loop control systems.

An open-loop control system is one in which the control action is independent of the output. An example of an open-loop control system is a chemical addition pump with a variable speed control. The feed rate of chemicals that maintain proper chemistry of a system is determined by an operator, who is not part of the control system. If the chemistry of the system changes, the pump cannot respond by adjusting its feed rate (speed) without operator action.

A closed-loop control system is one in which control action is dependent on the output. The control system maintains water level in a storage tank. The system performs this task by continuously sensing the level in the tank and adjusting a supply valve to add more or less water to the tank. The desired level is preset by an operator, who is not part of the system.

Feedback is information in a closed-loop control system about the condition of a process variable. This variable is compared with a desired condition to produce the proper control action on the process. Information is continually fed back to the control circuit in response to control action. In the previous example, the actual storage tank water level, sensed by

the level transmitter, is feedback to the level controller. This feedback is compared with a desired level to produce the required control action that will position the level control as needed to maintain the desired level.

Controlled System Response

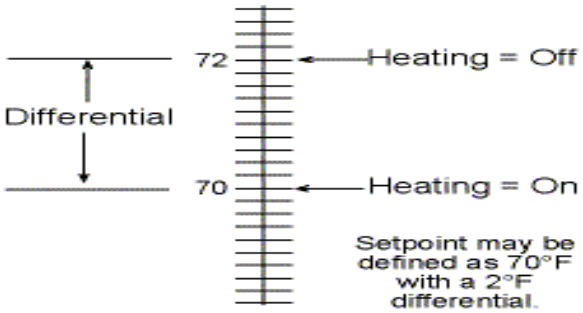
The following is taken from DDC Online, Control Response.

Two-Position Control

Two-position control compares the value of an analog or variable input with instructions and generates a digital (two-position) output. The instructions involve the definition of an upper and lower limit. The output changes its value as the input crosses these limit values. There are no standards for defining these limits. The most common terminology used is setpoint and differential. The setpoint indicates the point where the output "pulls-in," "energizes" or is "true." The output changes back or "drops-out" after the input value crosses through the value equal to the difference between the setpoint and the differential.

Two-position control can be used for simple control loops (temperature control) or limit control (freezestats, outside air temperature limits). The analog value can be any measured variable including temperature, relative humidity, pressure, current and liquid levels.

Time can also be the input to a two-position control response. This control response functions like a time clock with pins. The output "pulls-in" when the time is in the defined "on" time and drops out during the defined "off" time.



Source: DDC Online

Figure 9. Two-position control response

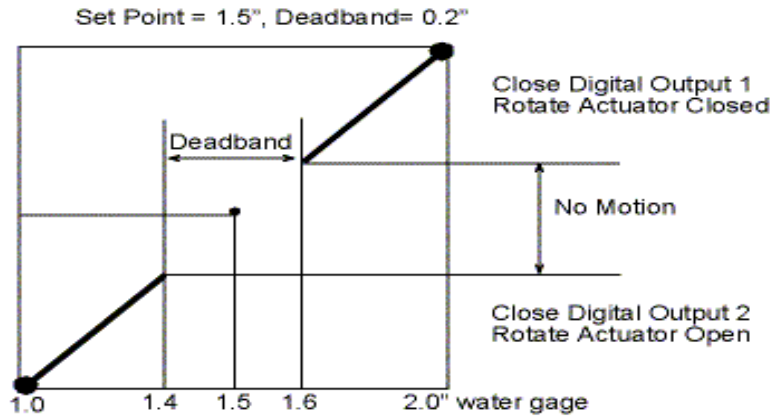
Figure 9 shows an example of two-position control in a home heating system, where the thermostat is set to energize the heating system when the space temperature falls below 70 degrees F and turn off when the temperature rises to 72 degrees F in the space. This is an example of a setpoint of 70 degrees F with a two-degree differential.

Floating Control

Floating control is a control response that produces two possible digital outputs based on a change in a variable input. One output increases the signal to the controlled device, while the other output decreases the signal to the controlled device. This control response also involves an upper and lower limit with the output changing as the variable input crosses these limits. Again, there are no standards for defining these limits, but the terms setpoint

and dead band are common. The setpoint sets a midpoint and the dead band sets the difference between the upper and lower limits.

When the measured variable is within the dead band or neutral zone, neither output is energized and the controlled device does not change—it stays in its last position. For this control response to be stable, the sensor must sense the effect of the controlled device movement very rapidly. Floating control does not function well where there is significant thermodynamic lag in the control loop. Fast airside control loops respond well to floating control. An example of floating controls is shown in figure 10.



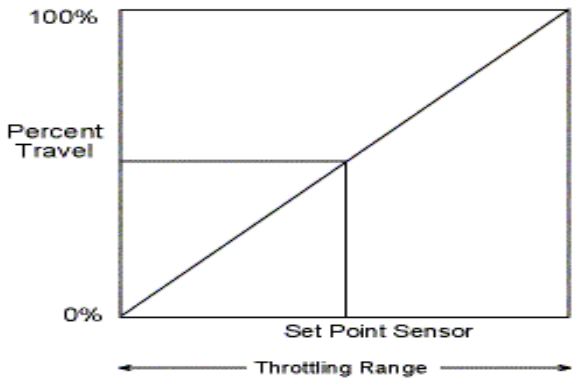
Source: DDC Online

Figure 10. Floating control response

Proportional Control

A proportional control response produces an analog or variable output change in proportion to a varying input. In this control response, there is a linear relationship between the input and the output. A setpoint, throttling range and action typically define this relationship. In a proportional control response, there is a unique value of the measured variable that corresponds to full travel of the controlled device and a unique value that corresponds to zero travel on the controlled device. The change in the measured variable that causes the controlled device to move from fully closed to fully open is called the throttling range. It is within this range that the control loop will control, assuming that the system has the capacity to meet the requirements.

The action dictates the slope of the control response. In a direct acting proportional control response, the output will rise with an increase in the measured variable. In a reverse acting response, the output will decrease as the measured variable increases. The setpoint is an instruction to the control loop and corresponds to a specified value of the controlled device, usually half-travel. An example is shown in figure 11.



Source: DDC Online

Figure 11. Proportional control response

In a proportional control system, the value of the measured variable at any given moment is called the control point. Offset is defined as the difference between the control point and the desired condition. One way to reduce offset is to reduce throttling range. Reducing the throttling range too far will lead to instability. The more quickly the sensor "feels" the effect of the control response, the larger the throttling range has to be to produce stable control.

Proportional Plus Integral (PI) Control

Proportional plus integral control involves the measurement of the offset or "error" over time. This error is integrated and a final adjustment is made to the output signal from the proportional part of this model. This type of control response will use the control loop to reduce the offset to zero. A well set-up PI control loop will operate in a narrow band close to the setpoint. It will not operate over the entire throttling range.

Proportional plus integral control loops do not perform well when setpoints are dynamic, where sudden load changes occur or if the throttling range is small.

Proportional Plus Integral Plus Derivative (PID) Control

Proportional plus integral plus derivative control adds a predictive element to the control response. In addition to the proportional and integral calculation, the derivative or slope of the control response will be computed. This calculation will have the effect of dampening a control response that is returning to setpoint so quickly that it will "overshoot" the setpoint.

Proportional plus integral plus derivative is a precision process control response and is not always required for heating, ventilation, and air conditioning applications. The routine application of PID control to every control loop is labor intensive and its application should be selective.

Definition of Direct Digital Control (DDC)

Direct digital control consists of microprocessor-based controllers with the control logic performed by software. Analog-to-digital converters transform analog values into digital signals that a microprocessor can use. Analog sensors can be resistance, voltage or current generators. Most systems distribute the software to remote controllers to eliminate the

need for continuous communication capability (stand-alone). The computer is primarily used to monitor the status of the energy management system, store backup copies of the programs and record alarming and trending functions. Complex strategies and energy management functions are readily available at the lowest level in the system architecture. If pneumatic actuation is required, it is accomplished with electronic to pneumatic transducers. Calibration of sensors is mathematical; consequently the total man-hours for calibration are greatly reduced. The central diagnostic capabilities are a significant asset. Software and programming are constantly improving, becoming increasingly user-friendly with each update.

e. Discuss the effects of time dependence in sensing and control systems in relation to the system dynamics. A possible example is a shock driven safety block in a fast burst reactor, as compared to a thermocouple sensor with motor driven reactivity removal in such a reactor.

The following is taken from Case Western Reserve University, Electrical Engineering and Computer Science Department, *Time-Dependent Dynamics in Networked Sensing and Control*.

A networked sensing and control system (NSCS) combines networked sensors with control and actuation units so as to control a physical environment. A fundamental problem in NSCS is that sensor and control signals are useless or dangerous if they are delivered too late. In particular, a late control can jeopardize the stability, safety, and performance of the controlled physical environment. As a result, the physics of an NSCS critically depends on the real-time network behavior.

In general, an NSCS will experience two distinct delays:

1. A delay from the sensor to the controller
2. A delay from the controller to the actuator

7. Criticality safety personnel must demonstrate a familiarity level knowledge of non-destructive assay techniques for quantification of fissile and fissionable materials.

a. For the following types of stationary assay equipment:

- **Calorimeter**
- **Gamma spectrometer**
- **Segmented gamma scanner**
- **Package gamma scanner or "package counter"**
- **High efficiency neutron counter**
- **Passive/Active neutron counter**

1. **Briefly describe each type of assay machine.**
2. **Describe the strengths and weaknesses of each type of machine.**
3. **Identify the types of materials that will grossly bias the assay, both high and low.**

Calorimeter

The following is taken from DOE-HDBK-1129-99.

Calorimetry is the quantitative measurement of heat. A calorimeter is an apparatus for measuring heat quantities generated in or emitted by materials in processes such as chemical reaction, changes of state, and formation of solutions. Heat is generally measured in calories or joules. A calorie is a unit of heat energy equal to the heat energy required to raise the temperature of a gram of water from 14.5 to 15.5°C, at a constant pressure of 1 atmosphere. A calorie is equal to 4.186 joules.

A calorimeter designed to be used in processes that continually generate heat (power sources) and measures power instead of heat is called a constant heat flow (CHF) calorimeter. A CHF calorimeter measures the power (joules/second) of a source, not the heat output (joules) of a source. The power is usually measured in watts, which is a unit of power equal to 1 joule/second.

A radioactive material is a power source, which deposits the energy due to decay in the radioactive material itself and in the materials surrounding the radioactive material. The power generated by the decay of tritium has been measured and is equal to 0.3240 ± 0.0009 watts/gram of tritium.

The CHF calorimeters are generally designed to meet the specific needs of the items to be assayed and are limited in application by the following:

- Physical size of the calorimeter measurement chamber
- Wattage range of the measurement system
- Precision and accuracy of the measurement for the size and wattage range of the item to be measured
- Throughput or number of samples to be measured per day

The CHF calorimeters have been designed in many different configurations, such as over/under, and twin. Most CHF systems in use today use digital control systems operated by a stored program and are easy to operate. The steps in making a CHF measurement are generally as follows:

- Install a dummy mass in the calorimeter container, pack steel or copper wool around the dummy mass, and install it in the measurement chamber.
- Make a zero baseline run at a wattage level (Wzbl), which is at a wattage level greater than the unknown wattage level of the sample to be measured.
- During the baseline run, the digital control system establishes a calorimeter bridge voltage value for a known Wzbl input.
- Remove the calorimeter container from the measurement chamber, remove the dummy sample from the container, replace it with the sample to be measured, place it back in the measurement chamber, and make an unknown sample run.
- During the unknown sample run, the digital control system decreases the power in the calorimeter until the bridge voltage is the same as that measured in the zero baseline run.
- The power input to the calorimeter during this unknown sample run (Wusr) is measured.

- The power of the sample being measured (Ws) is calculated by subtracting the wattage value measured during the zero baseline run from the wattage measured during the unknown sample run to find the wattage of the sample. In equation form:

$$Ws = Wzbl - Wusr$$

The calorimeter factor for tritium used at most DOE sites for the purposes of reporting accountable quantities of tritium to DOE is 0.3240 +/- 0.0009 Watts/g of tritium. CHF calorimetry can be used to measure tritium in solid form. CHF is the most accurate method available for the measurement of tritium quantities if the chamber size and wattage level of the item to be measured are well matched to the specifications of the CHF system being used. CHF systems, however

- do not provide any information about the different gases present in a container,
- only measure the quantity of tritium in the container,
- are not currently available for items larger than 11 inches in diameter and 16 inches long, and
- take several hours to complete a single measurement.

Gamma Spectrometer

Gamma spectroscopy is a radiochemistry measurement method that determines the energy and count rate of gamma rays emitted by radioactive substances. Gamma spectroscopy is an extremely important measurement. A detailed analysis of the gamma-ray energy spectrum is used to determine the identity and quantity of gamma emitters present in a material.

The equipment used in gamma spectroscopy includes a detector, a pulse sorter (multichannel analyzer), and associated amplifiers and data readout devices. The detector is normally a sodium iodide (NaI) scintillation counter.

The multichannel pulse height analyzer is a device that will separate pulses based on pulse height. Each energy range of pulse height is referred to as a channel. The pulse height is proportional to the energy lost by a gamma ray. Separation of the pulses, based on pulse height, shows the energy spectrum of the gamma rays that are emitted.

Multichannel analyzers typically have 100 or 200 channels over an energy range of 0 to 2 MeV. The output is a plot of pulse height and gamma activity, as shown in figure 12. By analyzing the spectrum of gamma rays emitted, the user can determine the elements which caused the gamma pulses.

Segmented Gamma Scanner

The following is taken from LANL, Segmented Gamma-Ray Scanner

The segmented gamma-ray scanner (SGS) is a rugged nondestructive assay (NDA) instrument used to quantify the radioisotope content of low-density scrap and waste containers. The SGS is most often used to measure waste containers holding special nuclear material (SNM), principally the isotopes of uranium and plutonium. The SGS combines physical motion and active radiation detection in the assay. The SGS has been

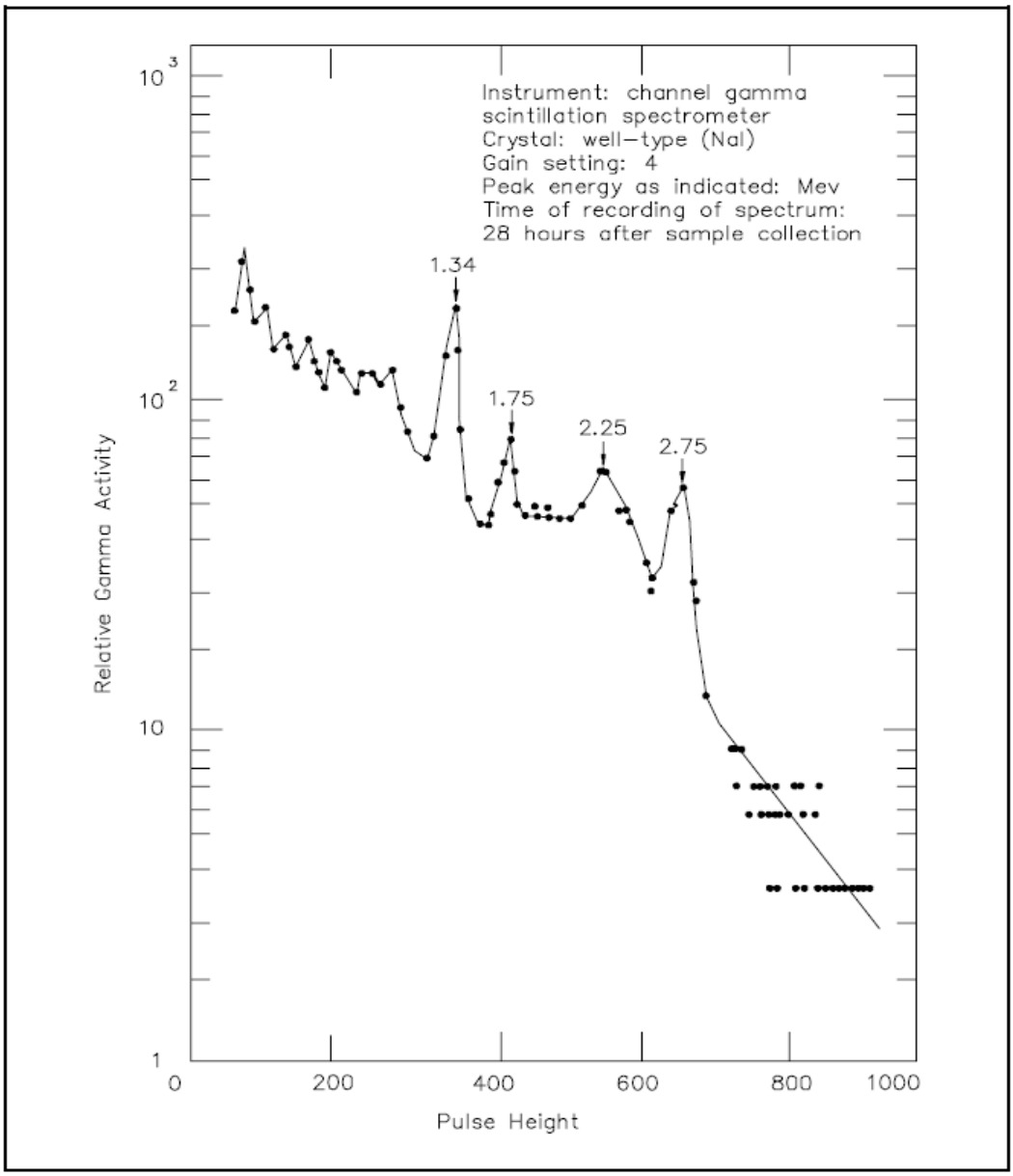
constructed in various configurations allowing measurements on items ranging from small vials to 200-liter drums. This device is well adapted to in-plant operation; its design is rugged, its components are reliable, and it is available from commercial sources. It may be the most widely used gamma-ray based NDA instrument.

The SGS measures gamma radiation from natural decay. Correction factors and refinements are then applied to the measurement to improve its accuracy. These enhancements fall into three categories: 1) gamma-ray attenuation and absorption corrections, 2) sample rotation and vertical scanning, and 3) rate-loss corrections.

Gamma-ray attenuation and absorption corrections compensate for absorption by the components of the sample. The SGS measures how much the assay item attenuates its own gamma rays by determining the attenuation of gamma rays by the sample from an external radioactive source. A correction that is a function of this measured attenuation is applied to gamma rays produced within the sample that are absorbed before they escape. A second correction uses the measurement of several gamma rays emitted from the sample to correct for the increased self-absorption by large, dense particles or lumps of SNM in the matrix compared with the case in which the SNM is uniformly distributed in the matrix. The total SNM content of the sample is obtained by summing the assay results from the individual segments.

Rate-loss corrections are made for count-rate-related losses in the electronics of the gamma-ray spectroscopy system. The relative rate of a small radioactive source fixed to the detector provides a basis for this correction.

The corrections make the SGS method applicable to a wide range of low-density SNM-bearing materials such as paper, plastics, ash, sand, and liquids. Dispersed higher-mass constituents, such as metals, can also be measured with somewhat lower accuracy, if the containers are too big.



Source: DOE-HDBK-1013/2-92

Figure 12. Multichannel analyzer output

Package Gamma Scanner

The following is taken from ORTEC/ANTEC, Comparison of Gamma-Ray Nondestructive Assay Measurement Techniques.

This type of scanner is a shielded waste assay instrument intended for the measurement of low-density waste in 200-liter (55-gal U.S.) drums or similar packages or bags. Other systems of the same general type exist. It is particularly suited for the measurement of low-level waste. Shielding is used to reduce the gamma-ray background, so the scanner can be used to consign waste that may be suitable for free release. The instrument includes an integrated load cell to weigh the waste item and to eliminate manual data entry

operations and errors. The measured sample weight is needed to determine sample density and estimate the average gamma-ray attenuation.

This system consists of three uncollimated high purity germanium (HPGe) coaxial detectors in a fixed, vertical array. One detector is located next to the mid-plane of the bottom third of the drum, one next to the mid-plane of the middle third and one next to the mid-plane of the top third of the drum. The detectors are most sensitive to gamma rays emanating from the closest third of the drum, but the detectors record gamma rays from the whole drum or package volume, increasing the total detection efficiency. The drum is rotated during the measurement. In its basic analysis mode, this system sums the individual gamma-ray spectra from the three uncollimated HPGe detectors to produce the assay result. In addition, it implements a more sophisticated analysis to produce much smaller errors and contains a diagnostic function to identify non-homogeneity in the measured sample.

As stated above, this system is used to measure low density waste in 200-liter (55-gal U.S.) drums or similar packages or bags. In practice, the maximum waste density for which the system is suitable is about 1.5 g/cm3. It can be used up to 2 g/cm3 with larger errors. The waste should be relatively homogeneous. This system can be used to measure non-homogeneous waste but with larger errors, which cannot be easily quantified.

Measurement errors are strongly dependent on counting statistics and hence the activity of the waste item and the degree of attenuation of the waste matrix as well as measurement time. For waste items with reasonable counting statistics (relatively high activity and low matrix density) measurement errors (accuracy) are typically 15 percent to 25 percent. If the waste item is not homogeneous, for example, the matrix contains regions of strong self-absorption or the activity distribution is very non-uniform, measurement errors can be very large, perhaps greater than 500 percent.

Passive Neutron Counter

The following is taken from LA-UR-00-4788.

This test method is useful for determining the plutonium content of items such as impure Pu oxide, mixed Pu/U oxide, oxidized Pu metal, Pu scrap and waste, Pu process residues, and weapons components.

Measurements made with this test method may be suitable for safeguards or waste characterization requirements such as

- nuclear materials accountability
- inventory verification
- confirmation of nuclear materials content
- resolution of shipper/receiver differences
- excess weapons materials inspections
- safeguards termination on waste
- determination of fissile equivalent content

A significant feature of neutron multiplicity counting is its ability to capture more information than neutron coincidence counting because of the availability of a third

measured parameter, leading to reduced measurement bias for most material categories. This feature also makes it possible to assay some in-plant materials that are not amenable to conventional coincidence counting, including moist or impure plutonium oxide, oxidized metal, and some categories of scrap, waste, and residues.

Calibration for many material types does not require representative standards. Thus, the technique can be used for inventory verification without calibration standards, at somewhat reduced accuracy.

The precision of the measurement results is related to the quantity of nuclear material, the (α,n) reaction rate, and the count time of the measurement.

For certain materials such as small Pu items of less than 1g, some Pu bearing waste, or very impure Pu process residues where the (α,n) reaction rate overwhelms the triples signal, multiplicity information may not be useful because of the limited precision of the triple coincidences within practical counting times.

For pure Pu metal, pure oxide, or other well-characterized materials, the additional multiplicity information is not needed, and conventional coincidence counting will provide better precision and sufficient accuracy. Conventional coincidence information can be obtained either by changing to a coincidence counter, or analyzing the multiplicity data in coincidence mode.

High Efficiency Neutron Counter

The following is taken from LA-UR-99-2214.

A super high efficiency neutron counter (Super HENC) has been designed for the non-destructive assay of Pu waste materials contained in large boxes, drums, or crates (2000-1 standard waste boxes for waste isolation pilot plant). The counter uses passive neutron counting to measure the spontaneous fission and (α,n) neutrons from Pu and Am-241. To make multiplicity counting practical, the counter has been designed with an efficiency of 40 percent, higher than any prior large scale counter. The counter makes use of 260 ^{3}He proportional counters at 10-atm pressure to obtain high efficiency and short neutron die-away time.

Neutron multiplicity counting allows the singles, doubles, and triples neutron coincidences to be measured, and thus solve for the Pu mass, the (α,n) rate, and the effective efficiency. The system will use the new advanced multiplicity shift register to perform multiplicity analysis and to reject cosmic-ray neutron events giving a Pu detectability capability that is about an order of magnitude better than prior systems. The complete detector system will be mounted in a mobile trailer that will move to several U.S. decommissioning sites.

Passive/Active Neutron Counter

The following is taken from LA-UR-89-3740.

The following describes a passive/active neutron counter based on a californium shuffler that has been delivered to the Portsmouth Gaseous Diffusion Plant in Piketon, Ohio.

A californium shuffler has four major components: the storage module for the californium source; the sample chamber and detectors; the control electronics (including the drive system for the californium source); and the software.

The shuffler is based on the principle that about 1 percent of the neutrons released following fission of ^{235}U come from the fission products with half-lives ranging from a fraction of a second to a few minutes (delayed neutrons). We interrogate the sample for several cycles; in each, the californium is moved close to the sample for 10s, it is withdrawn, and the delayed neutrons are counted for 7s. The source storage module shields both the instrument operator and the sample chamber from the californium source neutrons.

The shuffler has a detection efficiency for fission neutrons of about 20 percent, huge enough to make the counter suitable for passive coincidence counting. A shift-register circuit for passive neutron coincidence counting is also included. Neutron-flux monitors are used to correct for matrix effects. There are two modes of interrogation to correct for isotopic variations. Either thermal or epicadmium neutrons can be used in the active interrogation. The epicadmium mode is achieved with a steel reflector behind the ^{252}Cf source and a cadmium liner completely covering the inner surface of the counting chamber. In the thermal mode a 2.5-cm-thick polyethylene liner is added between the drum and the cadmium liner. Dual-energy interrogation allows correction for the ^{238}U effects in samples of unknown uranium isotopic distribution.

b. Discuss the various types of detectors (e.g., NaI, GeLi, HPGe, Geiger-Mueller, ^{3}HE, BF_3) used, and the strengths and weaknesses of each.

The following is taken from DOE-HDBK-1122-1999.

NaI

Inorganic crystals are comprised of inorganic salts, normally halides, which contain small quantities of impurities, called activators. The most commonly used inorganic crystal scintillator is sodium iodide, activated with thallium—commonly subscripted NaI(Tl). NaI(Tl) crystals have a high density— 3.7 g/cm^3, which allows for improved gamma photon response. The decay time is about 3 E-7 seconds (0.3 µsec). NaI(Tl) has a high response to beta particles; however, the need to hermetically seal a NaI(Tl) crystal to prevent deterioration, limits the actual beta response. Inorganic crystals operate as follows:

- An incident photon interacts with the crystal atoms (NaI) exciting the atom and raising valence band electrons to the conductance band, leaving a hole in the valence band.
- Some of these electrons and holes recombine to form an exciton. The excitons, free holes, and free electrons drift through the crystal.
- The impurity centers (T1) capture the excitons, free holes, and free electrons. This capture raises the impurity center to an excited state.
- The impurity center will decay back to the ground state, and in doing so, emits a light photon, which is proportional to the energy of the incident radiation.

NaI(Tl) scintillation detectors are commonly used in applications where high gamma sensitivity and a high energy resolution is desired.

- The solid nature of the crystal offers more targets to a photon than does a Geiger-Mueller (GM) detector. For this reason, gamma scintillators typically have higher yields than equivalently sized GM detectors.
- The light output of the crystal is a function of the incident photon energy. The output signal of the photomultiplier tube is a function of the light input, and therefore is proportional to the energy of the incident radiation. This characteristic allows scintillators to be used to perform pulse height analysis for radiation energy. The NaI(Tl) scintillator has a higher energy resolution than a proportional counter, allowing for more accurate energy determinations. Resolution is the characteristic of a detector to be able to differentiate between two close radiation energies. The higher the resolution, the closer the radiation energies can be to each other and still be differentiated. It should be noted that recent advances with semiconductor detectors have provided detectors with even better resolution than NaI(Tl).

Advantages of Scintillation Detectors

- Ability to discriminate between alpha, beta, gamma radiations, and between different radiation energies with a moderate resolution
- (NaI(Tl)): High gamma sensitivity
- (Liquid): Extremely low energy response
- (ZnS(Ag): Most advantageous alpha detector

Disadvantages of Scintillation Detectors

- (NaI(Tl)): No beta or alpha response, poor low energy gamma response
- (Liquid): Relatively cumbersome; solution is one time use only
- Requires a regulated power supply for pulse height analysis
- (NaI(Tl) and ZnS(Ag)): Detector is not a solid state device, needs to be handled with care

GeLi

A typical GeLi detector system consists of a vacuum enclosed GeLi crystal which is coaxial in shape and attached to a copper cold finger through an agate insulator. The crystal is under a vacuum to prevent frost forming on the crystal, and damage caused by impurities in the air. The cold finger is immersed in liquid nitrogen in a dewar. The crystal is subject to failure should its temperature be raised to room temperature due to lithium ion drift and increased electron noise.

Advantages of GeLi Detectors

GeLi detectors offer an advantage of high resolution. They can resolve the 3 photopeaks of ^{95}Zr/Nb whereas this would appear as 1 photopeak with NaI(Tl) detectors. GeLi also has a short response time and a more linear energy response than NaI(Tl). Small-size crystals offer the best resolution but the efficiency of detection is lowered.

Disadvantages of GeLi Detectors

GeLi systems can only be used for gamma photon detection. A cryogenic (liquid nitrogen) system is required to cool the detector and this adds to the initial cost as well as a continuing operation cost. Because detectors are presently limited to less than 100-300 cc

in size, the efficiency of counting is low compared to NaI. Counting times in excess of 1,000 minutes are necessary for environmental samples.

HPGe

If an extremely pure germanium crystal is created, the crystal's resistance will be sufficiently big so that a depletion region of 10mm can be obtained by using a reverse bias voltage, instead of drifting lithium through the germanium as is done in GeLi detectors. This pure germanium crystal would have, like GeLi crystals, semiconductor properties, and by applying the correct voltage could be used to collect electrons induced by radiation. Recently, processes for creating germanium in a very high state of purity have been developed. This, in turn, has led to the development of a semiconductor that can be stored at room temperature. The pure germanium crystals are usually called intrinsic germanium or abbreviated HPGe detectors. HPGe crystals are perhaps the most highly purified material that has even been produced.

Note that it was stated that the HPGe crystals could be stored at room temperature. This is because HPGe crystals are not lithium drifted. However, there was no mention of operating detectors with the crystals at room temperature. In order to reduce unwanted detector noise (caused by reduced resistance at room temperature) the detector must be operated at the temperature of liquid nitrogen. Experience has shown that the crystals can be cycled between room and cold temperatures without damage. Most manufacturers, however, recommend that the detectors be continuously maintained at liquid nitrogen temperatures if possible.

Advantages of HPGes

HPGes offer high resolution as an advantage. Additionally, the dewar used for storing the liquid nitrogen coolant for HPGe detectors can be smaller than the dewar used for GeLi detectors. HPGe detectors are more portable. The main advantage is that should the detector be allowed to rise to room temperature (with no voltage applied) the detector need only be returned to and stabilized at liquid nitrogen temperatures to be used.

Disadvantages of HPGes

Even though the dewar is smaller for the HPGe detector they still require liquid nitrogen cooling and tend to be fairly expensive.

Geiger-Mueller

As the voltage on the detector is increased beyond the proportional region, the detector enters the limited proportional region. As mentioned before, this region is unusable for radiological control purposes. In this region the small individual avalanches which occur within the tube start to interfere with each other. This interference is unpredictable and reduces the overall output signal.

As the voltage is increased further, the secondary ions are also accelerated to very high velocities and gain sufficient energy to cause ionization themselves. These tertiary ionizations spread rapidly throughout the tube causing an avalanche. The avalanche, caused by a single ionization, results in a single very large pulse. The avalanche continues until the fields created by the produced ions interfere with the field created by the high

voltage potential across the detector. When this occurs, the amount of acceleration decreases preventing further secondary ionization and halting the avalanche.

The output pulse size is a function of the gas amplification which occurs. In a GM tube, the gas amplification can range upwards from about 1 E8. Since the number of ions eventually produced and collected has no relation to the initial incident ionizing event, the pulse size is independent of radiation energy or specific ionization (a 0.1 MeV gamma creates the same size pulse as a 0.5 MeV gamma). For this reason, GM tubes cannot discriminate against different radiation types or radiation energies.

Any radiation event with sufficient energy to create the first ion pair can create a large pulse. For this reason, the GM detector is more sensitive than the ion chamber or proportional counter.

A GM detector can also be avalanched by the small amount of energy released by a positive ion when it is neutralized at the cathode. To prevent this undesirable occurrence, a quenching gas is added to the counting gas. Thus, instead of causing ionization, this excess energy is expended in dissociating the quenching gas molecules.

The following sequence of events explains the processes involved in GM detection.

- At time zero, the voltage potential across the detector is maximum. An incident radiation causes ionization, resulting in an ion pair.
- These ion pairs are accelerated towards the center electrode, thereby gaining energy.
- The primary ion pairs cause secondary ionization. The ion pairs created by the secondary ionization begin to accelerate towards the center electrode, thereby gaining energy. Since the potential is greatest near the center electrode, the bulk of the ionization occurs near the center electrode.
- The secondary ion pairs cause additional ionization and ion pairs. These ion pairs are accelerated and begin to cause ionization of their own. This process continues and an avalanche occurs.
- The negative ions (electrons) are collected by the center electrode and form a pulse. The positive ions form a cloud surrounding the center electrode. This ion cloud reduces the voltage potential across the detector. With a reduced voltage potential the gas amplification factor decreases such that secondary ionization stops, thereby halting the avalanche.

The events described above occur very rapidly, in the range of a fraction of microsecond. During this period the positive ion cloud is relatively stationary. The positive ion cloud is the cause of both the dead time and recovery time. Continuing:

- The positive ion cloud starts to drift towards the shell of the detector.
- As the cloud drifts, the voltage potential starts to increase.
- After about a microsecond (typically) the voltage potential is high enough to collect the electrons from another ionization should they occur. This is the end of the dead time. If another event does occur, the pulse will be very small and probably not measurable as the detector voltage is in the ion chamber region.
- As the ion cloud continues to drift, the voltage potential continues to increase and gas amplification starts to occur. The detector is now in the proportional region.

An event which occurs now will result in a large pulse. Whether or not this pulse is measured is a function of the input sensitivity of the electronic package.

- Eventually the gas amplification factor will increase to the point where an avalanche can occur when the positive ions reach the detector shell and are neutralized. At this point the detector has recovered and is ready for another radiation event. This time is about μ100-300 sec in typical detectors.
- During neutralization, the positive ions may release photons which in themselves could cause an avalanche if no quenching gas was present. Instead, the photons react with the molecules of the quenching gas, thereby dissipating their energy.

Advantages of GM Detectors

- GM detectors are relatively independent of the pressure and temperature effects which affect ion chamber detectors. This is because of the magnitude of the output pulse.
- GM detectors require less highly regulated power supplies. This is because the pulse repetition rate is measured and not the pulse height.
- GM detectors are generally more sensitive to low energy and low-intensity radiations than are proportional or ion chamber detectors.
- GM detectors can be used with simpler electronics packages. The input sensitivity of a typical GM survey instrument is 300-800 millivolt, while the input sensitivity of a typical proportional survey instrument is 2 millivolt.

Disadvantages of GM Detectors

- GM detector response is not related to the energy deposited; therefore GM detectors cannot be used to directly measure true dose, as can be done with an ion chamber instrument.
- GM detectors have a typically large recovery time. This limits their use in extremely high radiation fields. Dead time in a GM detector can be reduced by reducing the physical size of the detector. However, the smaller the detector, the lower the sensitivity. For this reason, wide-range GM survey instruments, such as the Teletector or the E520, commonly have two GM detectors—one for the low ranges, one for the high ranges.
- GM detectors cannot discriminate against different types of radiation (α, β, γ), nor against various radiation energies. This is because the size of the GM avalanche is independent of the primary ionization which created it.

Typical Applications

GM detectors are widely used in portable survey instruments at nuclear power facilities due to their ruggedness and the simplicity of the associated electronics. GM detectors are also used for personal monitoring for contamination (friskers), for process monitoring, and for area radiation monitoring. In addition, GM detectors are often used for laboratory counting when just a gross count is desired

^3He

The following is taken from LA-7890-C.

Large single-vessel two-dimensional systems are almost invariably ^3He (rather than BF_3) in view of safety considerations. Desirable design features are:

- High overall throughput including encoding. Two parameters are significant: the dead time or minimum pulse-to-pulse recording time, which should be less than 0.25 µs, and the maximum rate at which data may be stored, which should be greater than 10^5 sec^{-1}.
- Spatial and time resolution. These requirements are strongly dependent on specific instruments. The highest spatial resolution is needed for single-crystal diffractometers and neutron radiography, and the highest time resolution for epithermal neutron detection.
- Large area.
- Ability to cover arbitrary shapes and curved surfaces, and to encode in forms of spherical coordinates.
- High efficiency over an extended wavelength range.
- Uniformity of efficiency. This requirement also reaches its extreme for single-crystal diffractometers.
- Low cost, both for acquisition and for maintenance. The complexity and cost of the encoding electronics are a major consideration.
- Availability.

BF_3

The following is taken from DOE-HDBK-1122-99.

Each type of radiation has a specific ionization factor in a particular gas. In addition, each different detector gas has a different response to various radiation energies. By employing the most advantageous gas, a detector can be constructed that will have a higher yield for a specific radiation type or radiation energy than it will for other radiation types or energies.

A specific example of this is the use of BF_3 gas in proportional detectors to measure neutrons. In these detectors, the incident neutron fissions boron into lithium and an alpha particle. This alpha particle has a much higher specific ionization than does a gamma photon. The pulses created by neutrons are much larger than those created by gamma. The electronics sort out the pulses by pulse height.

Neutrons are measured. Alpha and beta radiation are not detected because they do not penetrate the detector shielding. Gamma radiation passes through the detector shielding but is rejected by the BF_3 proportional chamber up to 500 R/hr (dependent on high voltage setting and desired rejection level).

Since the BF_3 detector is operated in the proportional region, the pulses from the alpha particles are larger than pulses from other interactions and trigger a pulse height discriminator in the instrument circuitry. The mode of operation for the instrument is the pulse mode so that individual pulses can be discriminated and counted. Pulses from gammas are rejected. Alpha and beta particles do not penetrate the polyethylene sphere.

c. Discuss the physics and mathematics that relate count time, amount of material, and precision of the assay.

The following is taken from DOE-HDBK-1122-99.

Minimizing the statistical error present in a single sample count is limited to several options. If we look at the factors present in the calculation below, we can see that there are varying degrees of control over these factors. The standard deviation (σ) is calculated here in terms of count rate.

$$\sigma_{\text{rate}} = \sqrt{\frac{R_{S+B}}{T_s} + \frac{R_B}{T_B}}$$

R_{S+B} is the sample count rate. We really have no control over this.

R_B is the background count rate. We do have some control over this. On any counting equipment the background should be maintained as low as possible. In most counting applications, however, the relative magnitude of the background count rate should be extremely small in comparison to the sample count rate if proper procedures are followed. This becomes an issue when counting samples for free release or environmental samples. However, some reduction in error can be obtained by increasing the background counting time, as discussed below.

T_B and T_S are the background and sample counting times, respectively. These are the factors we have absolute control over. A count under given circumstances may be reproduced with a certain confidence level, and the larger the number of counts the greater the reliability. The condition is that the count is taken within a given time. In order to get more precise results, many counts must be observed. Therefore, for have low count rates, the counting time must be increased to obtain many counts, thereby making the result more precise (or reproducible).

The total counting time required depends on the sample and background count rates. For high-sample activities the sample count time can be relatively short compared to the background count time. For medium count rates we must increase the sample count time to increase precision. As the sample activity gets even lower, we approach the case where we must devote equal time to the background and source counts. In other words, by counting low-activity samples for the same amount of time as that of the background determination, we increase the precision of our sample result. However, we must never count a sample for a period of time longer than that of the system background.

In summary, by minimizing the potential error present, we improve statistical validity of our measurements.

d. Discuss the types of non-destructive assay equipment used for in-situ measurements.

The following is taken from DOE/EM-0477.

The in situ object counting system (ISOCS) developed by Canberra Industries, Inc. is a portable, in situ germanium-based spectroscopy system that is designed to provide information on the type and amount of radioactive material. The ability to provide quantitative information in real time reduces costly delays from offsite analysis.

The Canberra ISOCS consists of an ISOCS characterized germanium detector with portable cryostat; a cart support for holding the detector, lead shielding and collimators; an InSpector portable spectroscopy analyzer; a portable computer with Genie-PC software; and the in situ calibration software. The ISOCS characterized detector is a germanium detector whose response to a series of point sources surrounding it has been characterized using a Monte Carlo code. The steel-jacketed lead shielding can be mounted around the germanium detector to provide 1 or 2 inches of shielding from background radiation, and to change the field of view between 30, 90, or 180 degrees. The detector rotates on the cart for alignment with the target. The computer controls the InSpector analyzer and the Genie-PC software provides peak identification, data and error analysis, and sample quality assurance (QA). The ISOXSW software automatically determines the relationship between the radioactive source geometry, the measured count rate, and the amount of radioactive material present using the ISOCS characterized detector data.

The following is taken from American Society for Testing and Materials (ASTM) C-1133-03.

Segmented gamma-ray scanning provides a nondestructive means of measuring the nuclide content of scrap and waste where the specific nature of the matrix and the chemical form and relationship between the nuclide and matrix may be unknown.

The procedure can serve as a diagnostic tool that provides a vertical profile of transmission and nuclide concentration within the item.

Sample preparation is generally limited to good waste/scrap segregation practices that produce relatively homogeneous items that are required for any successful waste/inventory management and assay scheme, regardless of the measurement method used. Also, process knowledge should be used, when available, as part of a waste management program to complement information on sample parameters, container properties, and the appropriateness of calibration factors.

This test method covers the transmission-corrected NDA of gamma-ray emitting SNMs, most commonly ^{235}U, ^{239}Pu, and ^{241}Am, in low-density scrap or waste, packaged in cylindrical containers. The method can also be applied to NDA of other gamma-emitting nuclides including fission products. High-resolution gamma-ray spectroscopy (HRGS) is used to detect and measure the nuclides of interest and to measure and correct for gamma-ray attenuation in a series of horizontal segments (collimated gamma detector views) of the container. Corrections are also made for counting losses occasioned by signal processing limitations.

e. Discuss the types of equipment and limitations of assay when the material of interest is the following:
- **Shielded by containers or process equipment**
- **Low activity**

- **High activity**
- **Characteristic radiations are low energy**
- **Characteristic radiations are high energy**

The following is taken from McElroy, Robert D. Jr.; S. Croft, B, Young, *Non-Destructive Assay Box Counter*.

A feasibility study for an integrated waste assay system (IWAS) for the characterization of suspect TRU waste was undertaken by Canberra Industries as part of the U.S. DOE Program Research and Development Announcement. The integrated crate interrogation system (ICIS) represents one of several system types examined in that study. The following paragraphs discuss the performance of the super-IWAS system for the characterization of wastes within large containers such as the solid waste liner box and the ten-drum over pack.

The assay of large waste containers is not a new application. The currently deployed box counter technologies have been used with varying degrees of success for a variety of matrix types and measurement conditions. However, they all share important limitations. First, all existing box counter measurements, whether gamma or neutron based, are dependent on waste matrix composition and source distribution. Matrix composition parameters that affect measurements include elemental form, bulk density, presence and distribution of multiple matrix materials, and the concentration of interfering materials (such as neutron moderators and absorbers) in the matrix. Although matrix effects are generally smaller for passive neutron measurements than for gamma-ray or active neutron instruments, all of the existing techniques are susceptible to some extent. Self shielding is another effect that has (provided the moderator content is not excessive) the potential for increasing the bias in active neutron and gamma-ray measurements. Although passive neutron analysis is also less vulnerable to this effect, it suffers from poorer sensitivity and significantly higher background effects. All existing systems are affected by non-uniform distributions of radioactive sources. Non-uniform source distribution, including non-uniform plutonium composition, can bias NDA measurements either high or low.

Several attempts have been made in the recent past to correct biases due to matrix composition uncertainty, self shielding, source distribution effects, and background irregularities in nondestructive measurements. Some of these attempts include random triggering of the accidentals gate to reduce backgrounds in passive neutron measurements; matrix non-uniformity correction algorithms for gamma-ray analysis; neutron imaging and tomography for neutron and gamma-ray analysis, respectively; the “add-a-source” technique to compensate for matrix effects, etc. In addition, there have been attempts to provide matrix-specific calibrations for each different matrix form that the system is intended to measure. To some extent, all of these approaches have led to improvement in measurement accuracy. However, the improvements have generally been over a limited range of matrix types, under a limited set of conditions, or have required great effort to implement. Most of the correction approaches have also been applied to drum-sized, or smaller, containers. Their usefulness for analysis of boxed waste has not yet been validated.

The ICIS concept attempts to improve the likelihood of successfully characterizing radioactive wastes by integrating the assay results from three distinct assay modes, high-

efficiency passive neutron counting, active neutron interrogation, and box-segmented gamma scanning. The ICIS would consist of two physically separate counters, the super-IWAS passive/active neutron system and the box gamma segmented gamma scanner. These systems would be based on modifications to existing counter designs and techniques with an automated data integration technique tailored to these large systems.

The Multi-Modal Approach

The ICIS provides three distinct assay modes. The different analysis methods are complementary in that the weaknesses of one assay mode are often the strengths of another. The ICIS approach provides all the data that can be gathered on the waste container to minimize misclassification of wastes.

Passive neutron counting provides the following benefits:

- The passive neutron assay does not suffer from self-shielding or self-absorption effects, providing generally more accurate assay results than the HRGS or active neutron assays for higher plutonium and uranium loadings.
- More accurate than active neutron interrogation because it is less sensitive to matrix effects including thermal absorbers and less sensitive to source distributions. Matrix effects in passive neutron are smaller and more predictable, therefore, the errors from matrix effects can be bounded. This feature is necessary for approval of the total measurement uncertainty method for Waste Isolation Pilot Project (WIPP) certification.
- Significantly less sensitive to the presence of high-Z material in the waste matrix than HRGS.
- Interference from uranium is small. For passive assay ^{238}U is a weak interference but its effects can be corrected for in the neutron analysis when combined with the gamma-ray assay.
- The total neutron rate is proportional to the total alpha activity that must be reported as part of the characterization requirements for WIPP. While this measurement is subject to the chemical form of the alpha emitter it can be used to bound the sample's alpha activity.
- The ^{240}Pu-effective from the passive measurement can be combined with a gamma isotopic measurement to provide reliable reporting of plutonium and other isotopes.

High resolution gamma spectroscopy provides:

- Measurement of the plutonium, americium and/or uranium isotopic ratios. Standard isotopic analysis codes such as multi-group analysis and FRAM code (functions, responsibilities and authorities) are available and have been approved for isotopic measurements by DOE Carlsbad area office during multiple site audits.
- Quantitative measurement of plutonium and uranium for low-density matrices. By summing the spectra from all eight detectors, the proposed system can achieve detection levels well below the 60 nCi/g TRU detection level for plutonium wastes.
- Detection levels for ^{235}U < 0.5 grams for uniform source/matrix distributions for matrix density <1.2 g/cc.

- Direct measurement of other gamma emitters in the waste that are not identified in the isotopic measurement (e.g., 244Cm).
- Basic positional information to improve the accuracy of both the gamma and the neutron measurement.

Active neutron interrogation provides:

- Measurement of plutonium in the presence of interfering neutron emitters such as curium and californium.
- Measurement of plutonium in the presence of (alpha, n) neutron interferences.
- Confirmation that the container is below fissile limits when the uranium content cannot be definitively tied to the passive neutron measurement.
- "Arbitration" when passive neutron and quantitative gamma results disagree.
- Rapid screening for low-level wastes that are difficult to assay by means of the passive neutron or gamma modes.
- Measurement of uranium in the absence of plutonium.

f. Briefly discuss how to derive detection criteria and select the appropriate Non-Destructive Analysis (NDA) methods for stationary and in-situ applications.

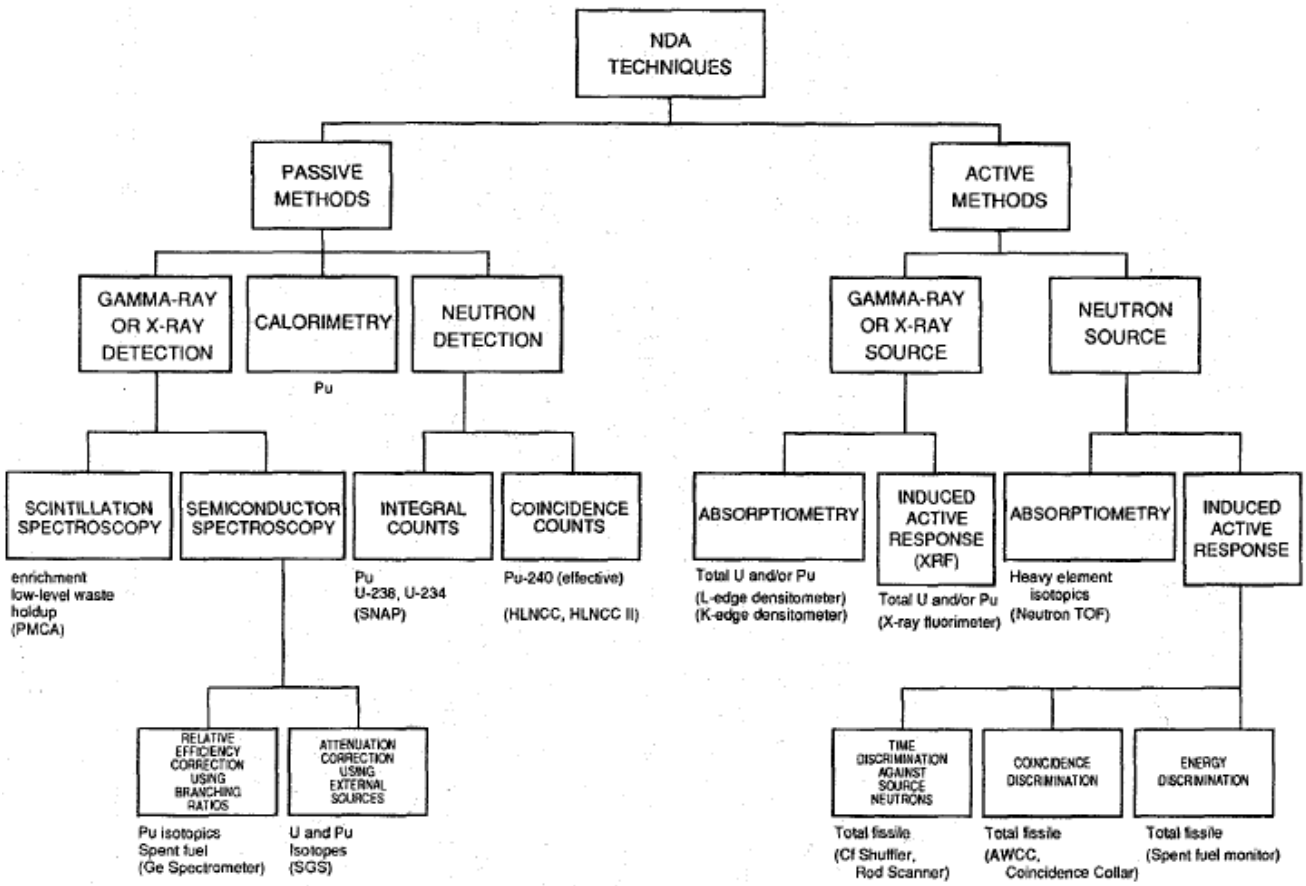
The following is taken from LA-UR-90-732.

Gamma rays follow radioactive decay and carry energy information that uniquely identifies the nuclides present in the sample. This information is usually preserved by the detection process. The principal difficulty for gamma-ray assay is to accurately correct for sample attenuation. Gamma-ray attenuation increases with atomic number and material density, so gamma-ray assay techniques work best for materials with low atomic number (<25) and low density (<1 g/cm3).

Neutrons carry less specific energy information, and even this is lost in the typical detection process. Neutron radiation does not provide information to identify the nuclear species present in the sample. On the other hand, neutrons penetrate dense, high-atomic-number materials (for example lead and uranium) with ease. They have more difficulty with very low atomic number materials, especially anything containing hydrogen, such as water or polyethylene.

Gamma-ray and neutron assay techniques are, therefore, complementary because of their different sensitivities to density and material type. In general, passive assay techniques work well on plutonium samples because plutonium decay reactions (alpha decay and spontaneous fission) have high specific activities. The very low specific activities of the uranium decay modes often dictate the use of active measurement techniques.

Figure 13 illustrates the variety of techniques available to the assayist. In the various nuclear fuel cycles, plutonium and uranium appear in many forms, and this can often make it difficult to select the appropriate technique for the measurement needs.



Source: LA-UR-90-732

Figure 13. An overview of passive and active NDA techniques, including the names of some common instruments

g. Briefly discuss how the geometry models for detector and source material (e.g. generalized geometry, plane source, point source, line source) affect the interpretation of raw NDA data.

The following is taken from LA-UR-90-732.

The absolute full-energy detector efficiency varies approximately as the inverse square of the distance between the detector and gamma-ray source. Consider a point source emitting I gamma rays per second. The gamma-ray flux F at a distance R is defined as the number of gamma rays per second passing through a unit area on a sphere of radius R centered at the source. Because the area of the sphere is 4 R^2, the expression for F is

$$F = \frac{I}{4\pi R^2}.$$

The detector count rate is proportional to the incident flux, and if the detector face can be approximated by a portion of a spherical surface centered at the source, the count rate has the same $1/R^2$ dependency as the flux. When low-intensity samples are counted, there is a clear motivation to reduce the sample-to-detector distance and increase the count rate.

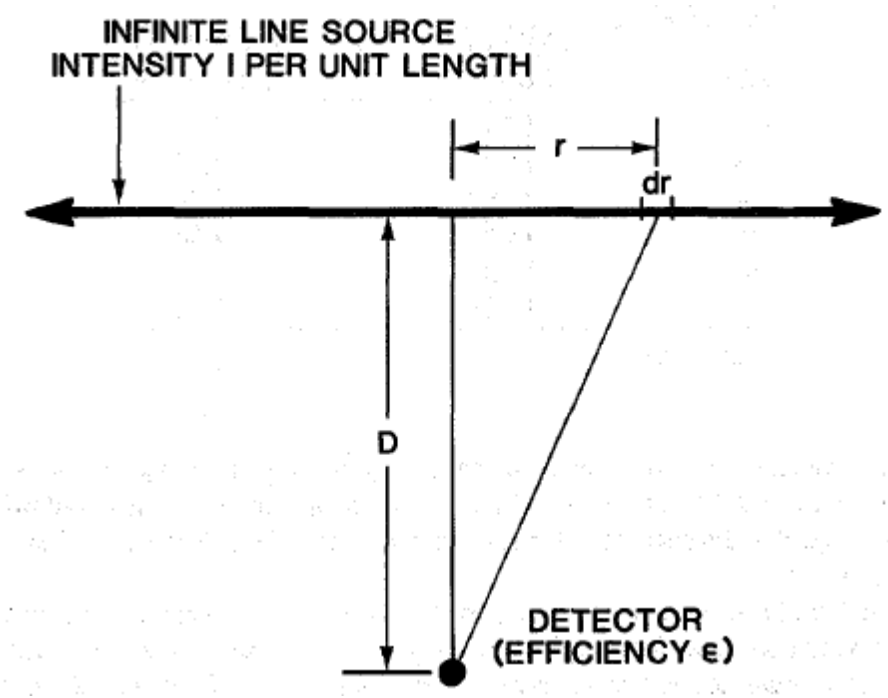
Unfortunately, when the sample-to-detector distance is so small that different parts of the sample have significantly different distances to the detector, the count rates from different parts of the sample vary significantly. This variation can cause an assay error when the distribution of emitting material is non-uniform.

The overall count rate from samples of finite extent does not follow the simple law; usually the variation is less strong than l/R^2. Knowledge of a few simple cases can help to estimate overall count rates and response uniformity.

The simplest extended source is a line, which is often an adequate model of a pipe carrying radioactive solutions. Consider an ideal point detector with intrinsic efficiency ε at a distance D from an infinitely long source of intensity I per unit length (figure 14). The count rate from this source can be expressed as

$$CR = 2\int_0^{\infty} \frac{I\varepsilon \ dr}{r^2 + D^2} = \frac{\pi I\varepsilon}{D}$$

In this ideal case, the count-rate dependence is l/R rather than l/R^2; when pipes are counted at distances much smaller than their length, the count-rate variation will be approximately l/R.



Source: LA-UR-90-732

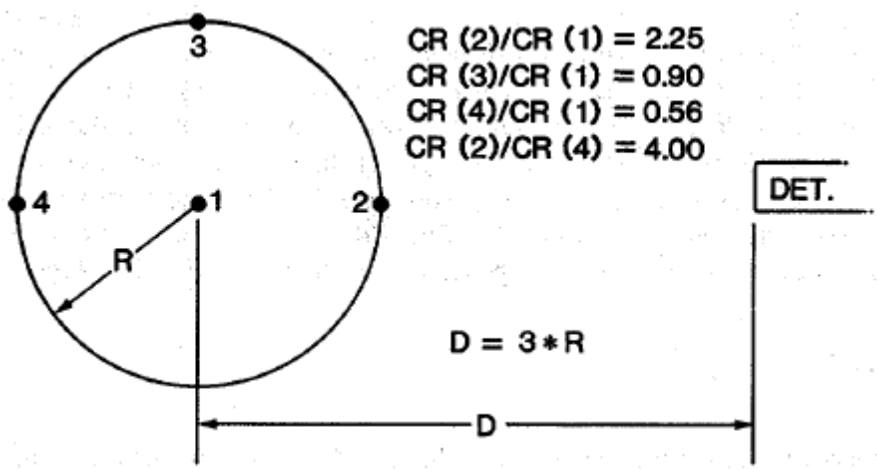
Figure 14. Geometry for computing the response of a point detector to a line source

The count rate from a point detector at a distance R from an infinite plane surface does not depend on R at all. For a detector near a 'uniformly contaminated glovebox wall, count rates vary little with wall-to-detector distance changes.

When possible the variation of response with position inside a sample should be minimized. The sample-to-detector distance can be increased, but the penalty is a severe loss of count rate. A better strategy is to rotate the sample. Consider the cross section of a cylindrical sample of radius R whose center is at a distance D from a detector (figure 15). Unless D is much greater than R, the count rates for identical sources at positions 1, 2, 3, and 4 vary considerably. The figure shows that if D = 3R, the maximum count-rate ratio is CR(2)/CR(4) = 4. The ratio of the response of a rotating source at radius R to the response at the center (position 1 of figure 15) is

$$\frac{CR(R)}{CR(1)} = \frac{1}{1 + (R/D)^2}$$

The response is the same as that obtained for a uniform non-attenuating circular source of radius R whose center is at a distance D from a detector. Table 1 gives the value of this function for several values of R/D compared to CR(2)/CR(l) for the non-rotating source of figure 15. For relatively large values of R/D, rotation improves the uniformity of response. The response variation is even larger when attenuation is considered. Rotation only reduces l/R^2 effects; it does not eliminate them completely.



Source: LA-UR-90-732

Figure 15. Cross section through a cylindrical sample and a point detector showing how count rate varies with position

Table 1. The effect of sample rotation on count rate variation

R/D	CR(R)/CR(1) (Rotating)	CR(2)/CR(1) (Nonrotating)
1/2	1.33	4.0
1/3[a]	1.125	2.25
1/4	1.067	1.78
1/5	1.042	1.56
1/6	1.029	1.44
1/7	1.021	1.36

[α] See Figure 15.
Source: LA-UR-90-732

Rotation reduces response variations caused by radial positioning, but does little to compensate for height variations. If the source height is less than one-third of the sample-to-detector distance, the decrease in response is less than 10 percent relative to the normal position.

The choice of sample-to-detector distance is a compromise between minimizing the response variations and maintaining an adequate count rate. A useful guideline is that the maximum count-rate variation is less than ±10 percent if the sample-to-detector distance is equal to three times the larger of the sample radius or the half-height. If a sample cannot be rotated, it helps considerably to count it in two positions 180 degrees apart.

8. Criticality safety personnel must demonstrate a familiarity level knowledge of the relationship between human factors, human performance, and implementation of criticality safety controls.

a. Identify and discuss aspects of person-machine interface that can degrade or enhance the safety performance of personnel.

The following is taken from DOE-HDBK-1028/1-2009, chapter 2.

When individuals believe that equipment is reliable, they may reduce their level of vigilance or even suspend monitoring of the equipment during operation. Automation, such as level and pressure controls, has the potential to produce such a dependency.

Boring tasks and highly repetitive monitoring of equipment over long periods can degrade vigilance or even tempt a person to violate inspection requirements, possibly leading to the falsification of logs or related records. Monitoring tasks completed by a computer can also lead to complacency. In some cases, the worker becomes a “common mode failure” for otherwise independent facility systems, making the same error or assumption about all redundant trains of equipment or components.

Diminishing people’s dependencies on equipment can be addressed by:

- applying forcing functions and interlocks;
- eliminating repetitive monitoring of equipment through design modifications;
- alerting personnel to the failure of warning systems;
- staggering work activities on redundant equipment at different times or assigning different persons to perform the same task;
- diversifying types of equipment or components, thereby forcing
- the use of different practices, for example for turbine-driven and motor-driven pumps;
- training people on failure modes of automatic systems and how they are detected;
- informing people on equipment failure rates; and
- minimizing the complexity of procedures, tools, instrumentation, and controls.

The human-machine environment contains several opportunities to “control” human error. Human-centered designs consider human error and its potential consequences, eliminating

or minimizing error traps with equipment. Consideration is given to the habitability and accessibility of the physical work environment.

Unnecessary human interactions with facility equipment are either eliminated or automated. Otherwise, interlocks and error-tolerant designs have to be used to mistake-proof human-machine interactions, especially those with risk-important systems and critical components.

Interlocks and protection systems are provided to prevent improper operator actions and to initiate automatic protective actions when necessary protection systems will not prevent all possible operator errors, but they can substantially reduce the risks if they are properly maintained. Supervisors initiate modifications to eliminate or minimize errors associated with workarounds and human-machine interface deficiencies. These actions are especially important at critical steps. Other important elements relevant to effective engineered controls include configuration control, material condition, foreign material exclusion, and housekeeping practices. Problems with environmental conditions, labeling, accessibility, lighting, and habitability are resolved, if possible, to minimize their impact on performance, especially on risk-important equipment. These are administrative controls (ACs) in support of the engineered controls.

Common Flaws with Engineered Controls

The following list highlights some of the more common equipment-related conditions that challenge worker performance and can contribute to facility events:

- Out-of-service equipment, controls, alarms, and indicators
- Workarounds, temporary repairs, or long-term temporary modifications/alterations
- Nuisance alarms and disabled annunciators
- Excessive noise
- Missing labels or labels oriented such that they cannot be seen or read easily
- Poor lighting
- High temperatures or high humidity (heat stress factors)
- Unusual plant or equipment conditions
- Poor accessibility, cramped conditions, or awkward layout of equipment

b. Identify and discuss how written procedures are conducive to reliable or unreliable performance of activities important to safety.

The following is taken from DOE-HDBK-1028/1-2009, chapter 4.

Administrative controls, such as procedures, inform people about what to do, when to do it, where it is to be done, and how well to do it, and are usually documented in various written policies, programs, and plans. Administrative controls rely on human judgment, training, and personal initiative to follow the direction contained in documents. Consequently, ACs are not as reliable as engineered controls.

Example: Administrative Controls

A wide range of management methods exists to ensure proper facility operations and to control various hazards. Administrative controls that significantly impact human performance include the following:

- Strategic business planning (e.g., goals, budgeting, priorities, plans, and resource acquisition)
- Formal organizational structure, lines of authority, roles, and responsibilities
- Policies, programs, and processes for the conduct of production work activities (e.g., preventive maintenance, procedure development, modifications, configuration control, and operations)
- Communication methods (e.g., conversations, e-mail, logs, meetings, reports, newsletters, signs, postings, telephones, radios, alarms, and so on)
- Technical and administrative procedures (e.g., clearances/tagging, foreign material exclusion, industrial safety, human performance, troubleshooting, records, parts and materials, self assessment, and corrective action)
- Training programs
- Work management processes (e.g., work initiation, prioritization, review and approval, planning, and scheduling)
- Human resources policies and practices related to staffing levels, overtime, and discipline
- Human performance tools, expectations, and standards
- Information technology and information handling

Common Flaws with ACs

The following administrative conditions, among others, can be causes or contributing factors in facility events:

- Two or more actions embedded in one procedure step
- Vague expectations and standards
- Superficial document reviews or the lack of a "qualified reviewer" process for technical procedure development
- Critical steps not identified in procedures and work packages
- Excessive work package backlog that exceeds planner resources
- Work packages planned without inclusion of operating experience
- Unresponsive procedure revision process
- Unavailable foreign material exclusion caps and covers
- Excessive deferred preventive maintenance
- Insufficient staffing leading to excessive overtime, workload, and fatigue
- Routine authorization to exceed overtime limits (leading to chronic fatigue)
- Inadequate time for direct supervision of work in the field
- Unclear qualification standards
- Incomplete or missing electrical load lists to aid in ground isolation.

c. Identify and discuss how personnel training programs can be conducive to safety or prone to error.

The following is taken from *Improve Safety by Improving Human Performance*.

Training ensures that workers possess the basic skills necessary to effectively perform their functions. Proper training of workers is critical to reducing human errors. Several types of training have proven most effective in reducing human errors, including initial skill training, refresher training, and management systems training.

Classroom knowledge-based training provides the basic expertise, technical background, and understanding of the procedures. Initial skills training is a bridge between the knowledge-based training and on-the-job performance. It is generally conducted in the classroom and supplemented with on-the-job experience, and should be viewed as the means for preparing workers for real experiences they will routinely encounter, as well as those they will infrequently encounter. If skill training does not include the critical infrequent events or situations, the likelihood of successfully handling them will depend solely on the problem-solving and decision-making skills of the worker (which may not be sufficient).

In addition to the initial training, refresher training on non-routine or modified tasks will minimize worker errors and reduce the potential for a worker's skills to deteriorate. A refresher training (or practice) program that regularly challenges the worker's knowledge and provides opportunities to practice skills is needed to assist workers in developing and maintaining a high skill level, and enhancing a new worker's skills beyond the initial training level.

To round out a training program, regular training is required to ensure that workers can readily identify and follow relevant management systems. To prevent a common-cause human error, many companies have policies for redundant checking of critical tasks by different personnel. For example, the equipment may be aligned by one worker, checked by another, and verified by the supervisor before the procedure is started. Examples of other management systems include management-of-change procedures, safe work practices, and pre-startup reviews.

d. Identify and discuss how staffing and qualification of operational personnel are conducive to safe versus unsafe operations.

The following is taken from DOE M 426.1-1A.

The objectives related to staffing and qualification of operational personnel are:

- Identify and document the functional competencies individual employees must possess to ensure that DOE defense nuclear facilities and programs are operated in accordance with applicable safety, health, and environmental requirements.
- Establish a program that clearly identifies and documents the process used by senior line management to demonstrate employee technical competence, consistent with applicable industry standards for similar occupations.
- Ensure that employees maintain their technical competencies.
- Maintain a cycle of continuous performance improvement through structured, individualized training and development programs and through review and assessment of headquarters and field element programs.

e. Identify and discuss the influence of management and organizational factors upon safety of operations.

The following is taken from DOE-HDBK-1028/1-2009, chapter 3.

Organizational effectiveness is the ability of an organization to accomplish its goals efficiently. To achieve organizational effectiveness, the management team must organize

its resources, especially its people. Organizing involves determining the division of labor and coordinating the effort. Establishing functions, goals, roles and responsibilities, structure, and job assignments determines the division of labor. Managers pay attention to the tools of the organization—things typically written on paper (the AC system). They use formal policies, business plans, priorities, directives, goals and objectives, programs, processes, planning and scheduling, action plans, and expectations and standards to provide direction and controls to accomplish the facility's mission. The purpose of controls is to make processes (or tasks) go smoothly, properly, and according to high standards.

Managers shoulder the responsibility for overall facility performance. To discharge their responsibilities, managers use work processes as the primary mechanism to coordinate work. Functions carried out by managers to establish work processes include:

- Decide the administrative and functional structure needed to establish a standardized sequence of tasks to be accomplished.
- Develop and approve procedures to direct workers production and maintenance tasks.
- Train people to do the work, specifying what, how, why, and when they are expected to accomplish their tasks.
- Establish processes that provide feedback and identify opportunities for improvement.
- Set priorities of the organization.

The effectiveness of work processes is improved when managers communicate clear expectations to the workers, when they promote open communication, and when they strive for quality procedures and make use of an effective corrective action program.

Organizational factors have a strong influence on human performance. Organizational factors encompass all the ways management uses to direct and coordinate the work of the facility, which together shape the behavior of the people performing their jobs. Collectively, they are the hub of all that goes on at the facility. Organizational factors reveal themselves in engineered controls, ACs, cultural controls, and oversight controls (corporate and independent). Some of the more important organizational factors known to impact performance are the following:

- Communication methods and practices
- Management styles and degree of workforce
- Participation
- Tools and resources
- Procedure development and review
- Cleanliness of the work environment
- Layout of facilities and structures
- Staffing levels
- Experience level of the workforce
- Design and modification
- Work processes
- Management visibility
- Human resources policies and practices
- Training programs

- Priorities (production and safety)
- Expectations and standards
- Emphasis on health and safety
- Work planning and scheduling

f. Identify and discuss the methods used to estimate the probability of significant mistakes made by personnel and the relationship to Probabilistic Risk Assessment (PRA).

The following is taken from University of Pittsburgh, THERP: Technique for Human Reliability Analysis.

THERP, the technique for human error rate prediction, was developed by Alan Swain at Sandia National Laboratories in the 1950's as a quality control method for estimating errors in the assembly of nuclear warheads. Although the full-blown methodology incorporates distributional assumptions needed for sensitivity analysis, it is exceedingly simple in point estimate form. Types of errors, such as reading or omitting an instructional step, or choosing the wrong switch, are presumed to occur at constant rates. If the tasks that a person performs can be broken down into subtasks for which these types of errors can be predicted, then the probability of the successful completion of the overall task can be predicted.

As with other methods for reliability analysis, the usefulness of THERP depends on the type of prediction. Where tasks are rote, there is little stress, and each step is crucial to successful completion, THERP works very well. If any of these conditions are not met, it may produce estimates that deviate substantially from actual failures. THERP is often criticized for its assumption that human error rates can be accurately quantified and predicted. The diagrams provided in THERP: *Technique for Human Reliability Analysis* should suggest why this criticism is less telling than it may seem. The component failure rate in these models can be doubled or halved without affecting their relative differences in reliability in any substantial way (1.6×10^{-3}vs. .59) for doubling rates for instance. The tabled error rates used by THERP have evolved over a 30-year period based on a combination of statistical data and expert judgment and are presumed to be accurate within an order of magnitude. When studies have been conducted to verify this assumption, estimates are usually found to be much closer, varying by factors of 2 or 3 rather than 10. The modeling assumptions typically made by human reliability analysts are another matter. As in any form of reliability analysis the validity of results depends crucially on the modeling of dependencies. Although paying lip service to this nostrum, human reliability analysis as a practical tool, almost always presumes independence unless there is overwhelming evidence to the contrary.

Although the probability that a person will successfully complete a long task is a complex conditional probability involving a myriad of possible combinations of errors, it becomes exceptionally simple if these errors just happen to be independent. When this is the case, the conditional probabilities are just the same as the simple ones. If it also just so happens that every step is necessary to the successful completion of the task, then this probability becomes simply, one minus the joint probability that each of the steps were successfully completed. This very special case (although with slight modifications to accommodate dependence when it can't be avoided) forms the basis of THERP's event tree (ET)

methodology. Tasks are represented as a tree of constituent task steps, each of which can either be successfully completed or be unsuccessful due to error. The error branches of the tree are usually left undeveloped, resulting in a tree having a single success path with errors at each of the steps represented by undeveloped leaf nodes. The probability of successful task completion can then be found simply by taking the product of the simple probabilities along the success path.

The only other prominent feature of THERP lies in modeling the recovery of errors. Just as people are presumed to make errors at fixed rates, they are presumed to fail to notice they have made errors at particular rates. So, for example, if when setting a clock with a display which blinked until a particular step was successfully completed, if the step were omitted, noticing the blinking digits would provide a second opportunity to correct the error. This type of error recovery plays an important role in reliable human performance. To see why, consider an error which is made at rate, a, and recovered at rate, 1- b. The probability of failure at that step is now reduced to a times b. These error-recovery cycles in the tree can make persisting errors extremely rare for some steps. Another feature of ACs rewarded by this model of error recovery is the use of checklists and subsequent verifications. If these fail at a rate of ~.1, each additional check increases the system's reliability by an order of magnitude.

g. Identify and discuss the methods for assessing the reliability of administrative controls in maintaining criticality safety.

Refer to competency statement 8b for a discussion of ACs.

9. Criticality safety personnel must demonstrate both a working level knowledge of calculational methods used in criticality safety evaluations, and must have demonstrated the ability to use such methods.

a. Identify and discuss the application of several common hand calculation methods (e.g., buckling method, solid angle method, surface density, and density analog). Reference LA-14244, *Hand Calculation Methods for Criticality Safety – A Primer*, Bowen and Busch, November 2006.

The following is taken from LA-14244.

Single Unit Methods

- One-group diffusion theory
- One-group modified diffusion theory
- Buckling conversions
- Core-density conversions

Array Unit Methods

- Surface density method
- Density analog method
- Limiting surface density (NBN2) method
- Solid angle method

Single Unit Methods

The methods listed in table 2 are valid for single fissile units only. The methods discussed for single units are the one-group and modified one-group diffusion theories, buckling conversions, and core-density conversions. These methods can be used to resolve a wide variety of criticality safety problems as summarized table 2.

Table 2. Single unit methods and applicability summary

Single Unit Hand Calculation Method	Applicability Summary
One-group and Modified One-Group Diffusion Theories	• Good for large, homogeneous systems with isotopes that have low neutron absorption. • Caution: Diffusion theory is not a good method to use for small systems, near boundaries, or in or near strong neutron absorbing materials.
Buckling Conversions	• Useful for simple geometries such as spherical, slab, or cylindrical systems. • Can convert the neutron leakage characteristics for a critical simple geometry to another simple geometry that has equivalent leakage characteristics, as long as critical data exist for a particular system.
Core-Density Conversions	• Can be applied to homogeneous, critical systems if the volume or density of the system changes uniformly. • Applicable to bare systems or those with a close-fitting reflector, as long as the reflector density remains constant.

Source: LA 14244-M

Array Unit Methods

The methods listed in table 3 are valid for fissile units arranged in certain array configurations. The methods discussed in this section are the surface density method, density analog method, the solid angle method, and the limiting surface density method or the NBN2 method. These methods can be used to resolve a wide variety of criticality safety problems in which fissile materials are arranged into various multiple-unit configurations.

Table 3. Array methods and applicability summary

Array Hand Calculation Method	Applicability Summary
Surface Density Method	• Useful for determining the subcritical center-to-center spacing for fissile materials stored or staged in finite array configurations where the size of the array is controlled in one direction • Useful for irregular shapes such as equipment stored on the floor • Considers 15.5 cm of water reflection on the top and bottom of the array
Density Analog Method	• Useful for determining the subcritical center-to-center spacing for fissile materials stored or staged in array configurations of any shape • Useful for irregular shapes such as equipment stored on the floor • Considers 20.0 cm of water reflection on the top and bottom of the array
Limiting Surface Density (NB_N^2) Method	• Useful for determining the critical center-to-center spacing for fissile materials stored or staged in array configurations of any shape (>64 units) • Useful for irregular shapes such as equipment stored on the floor • Data exists for powders, metals, up to an H/X of about 20 for some fissile materials • Data exists for arrays reflected by concrete instead of water • Can be used to calculate trends due to a change in unit shape or density • Considers 20.0 cm of water reflection on the top and bottom of the array
Solid Angle Method	• Useful for small numbers of moderated fissile units, because the basis for this method is experiments with aqueous solutions of fissile materials. • The multiplication factor for any individual unit cannot exceed 0.8, and the unit must be subcritical with a thick close-fitting water reflector. • The minimum separation distance between fissile units should be at least 0.3 meters, and the total allowed solid angle should not exceed 6 steradians. • Reflectors that are more effective than a thick water reflector should not be considered for this method. • Concrete reflection on three sides of the fissile material is considered bounded by this method.

Source: LA 14244-M

b. Prepare an example using each one of the hand calculational methods listed above, or others as accepted or identified by the qualifying official.

This is a performance-based competency. The Qualifying Official will evaluate its completion.

10. Criticality safety personnel must demonstrate a working level knowledge of critical and subcritical experiments.

a. Describe the types of data derived from critical experiments and its use in criticality safety.

The following is taken from the introduction to the International Handbook of Criticality Benchmark Experiments.

Critical experiment data is derived from experiments involving seven types of fissile materials:

1. Plutonium systems
2. Highly enriched uranium systems (weight percent U 235 ≥ 60)
3. Intermediate and mixed enrichment uranium systems (10 < weight percent U-235 < 60)

4. Low enriched uranium systems (weight percent U-235 < 10)
5. Uranium-233 systems
6. Mixed plutonium–uranium systems
7. Special isotope systems

Each of these material types is then categorized by the physical form of the fissile material:

- Metal systems
- Compound systems
- Solution systems
- Miscellaneous systems

These physical forms are then categorized by fast, intermediate, thermal, and mixed spectra systems depending upon where the majority of the fissions occur. In general, fast, intermediate, and mixed subdivisions are not applicable to solution systems; however, for certain solution systems (e.g., some heavy-water-moderated systems) the majority of the fissions occur above the thermal range.

In this handbook, fast, intermediate, and thermal systems are defined as systems in which over 50 percent of the fissions occur at energies over 100 keV, from 0.625 eV to 100 keV, and less than 0.625eV, respectively. Systems for which over 50 percent of the fissions do not occur in any one of these three energy ranges are classified as mixed spectra systems.

Some experiments can be categorized into more than one subsection. In these special cases, the data are assigned where they appear to fit best, with an attempt to include cross-references. Therefore, experimental data are presented only once.

Critical experiment data is used by criticality safety analysts to validate calculationa1 techniques and cross sections.

b. Discuss the physics of critical experiments including fundamental concepts associated with critical experiments (e.g., six factor formula, approach to critical, reactivity insertion, multiplication, reactor kinetics, reactivity changes, etc.).

The following is taken from DOE-HDBK-1019/2-93.

Start with the four factor formula. The discussion of the four factor formula uses ^{235}U as the fuel, and ^{238}U as a resonance absorber. However, the concepts apply to plutonium systems also.

Infinite Multiplication Factor, K_∞

Not all of the neutrons produced by fission will have the opportunity to cause new fissions because some neutrons will be absorbed by non-fissionable material. Some will be absorbed parasitically in fissionable material and will not cause fission, and others will leak out of the reactor. For the maintenance of a self-sustaining chain reaction, however, it is not necessary that every neutron produced in fission initiate another fission. The minimum condition is for each nucleus undergoing fission to produce, on the average, at

least one neutron that causes fission of another nucleus. This condition is conveniently expressed in terms of a multiplication factor.

The number of neutrons absorbed or leaking out of the reactor will determine the value of this multiplication factor, and will also determine whether a new generation of neutrons is larger, smaller, or the same size as the preceding generation. Any reactor of a finite size will have neutrons leak out of it. Generally, the larger the reactor, the lower the fraction of neutron leakage. For simplicity, we will first consider a reactor that is infinitely large, and therefore has no neutron leakage. A measure of the increase or decrease in neutron flux in an infinite reactor is the infinite multiplication factor, k_∞. The infinite multiplication factor is the ratio of the neutrons produced by fission in one generation to the number of neutrons lost through absorption in the preceding generation. This can be expressed mathematically as shown below.

$$K_\infty = \frac{\text{neutron production from fission in one generation}}{\text{neutron absorption in the preceding generation}}$$

Four Factor Formula

A group of fast neutrons produced by fission can enter into several reactions. Some of these reactions reduce the size of the neutron group while other reactions allow the group to increase in size or produce a second generation. There are four factors that are completely independent of the size and shape of the reactor that give the inherent multiplication ability of the fuel and moderator materials without regard to leakage. This four factor formula accurately represents the infinite multiplication factor as shown in the following equation.

$$k_\infty = \varepsilon \rho f \eta$$

where:
ε = fast fission factor
ρ = resonance escape probability
f = thermal utilization factor
η = reproduction factor

Each of these four factors, which are explained in the following subsections, represents a process that adds to or subtracts from the initial neutron group produced in a generation by fission.

Fast Fission Factor, (ε)

The first process that the neutrons of one generation may undergo is fast fission. Fast fission is fission caused by neutrons that are in the fast energy range. Fast fission results in the net increase in the fast neutron population of the reactor core. The cross section for fast fission in uranium-235 or uranium-238 is small; therefore, only a small number of fast neutrons cause fission. The fast neutron population in one generation is therefore increased by a factor called the fast fission factor. The fast fission factor (ε) is defined as the ratio of the net number of fast neutrons produced by all fissions to the number of fast neutrons produced by thermal fissions. The mathematical expression of this ratio is shown in the following equation.

$$\varepsilon = \frac{\text{number of fast neutrons produced by all fissions}}{\text{number of fast neutrons produced by thermal fissions}}$$

In order for a neutron to be absorbed by a fuel nucleus as a fast neutron, it must pass close enough to a fuel nucleus while it is a fast neutron. The value of ε will be affected by the arrangement and concentrations of the fuel and the moderator. The value of ε is essentially 1.00 for a homogenous reactor where the fuel atoms are surrounded by moderator atoms. However, in a heterogeneous reactor, all the fuel atoms are packed closely together in elements such as pins, rods, or pellets. Neutrons emitted from the fission of one fuel atom have a very good chance of passing near another fuel atom before slowing down significantly. The arrangement of the core elements results in a value of about 1.03 for ε in most heterogeneous reactors. The value of ε is not significantly affected by variables such as temperature, pressure, enrichment, or neutron poison concentrations. Poisons are non-fuel materials that easily absorb neutrons and will be discussed in more detail later.

Resonance Escape Probability, (ρ)

After increasing in number as a result of some fast fissions, the neutrons continue to diffuse through the reactor. As the neutrons move they collide with nuclei of fuel and non-fuel material and moderator in the reactor losing part of their energy in each collision and slowing down. While they are slowing down through the resonance region of uranium-238, which extends from about 6 eV to 200 eV, there is a chance that some neutrons will be captured. The probability that a neutron will not be absorbed by a resonance peak is called the resonance escape probability. The resonance escape probability (ρ) is defined as the ratio of the number of neutrons that reach thermal energies to the number of fast neutrons that start to slow down. This ratio is shown below.

$$\rho = \frac{\text{number of neutrons that reach thermal energies}}{\text{number of fast neutrons that start to slow down}}$$

The value of the resonance escape probability is determined largely by the fuel-moderator arrangement and the amount of enrichment of uranium-235 (if any is used). To undergo resonance absorption, a neutron must pass close enough to a uranium-238 nucleus to be absorbed while slowing down. In a homogeneous reactor the neutron does its slowing down in the region of the fuel nuclei, and this condition is easily met. This means that a neutron has a high probability of being absorbed by uranium-238 while slowing down; therefore, its escape probability is lower. In a heterogeneous reactor, however, the neutron slows down in the moderator where there are no atoms of uranium-238 present. Therefore, it has a low probability of undergoing resonance absorption, and its escape probability is higher.

The value of the resonance escape probability is not significantly affected by pressure or poison concentration. In water moderated, low uranium-235 enrichment reactors, raising the temperature of the fuel will raise the resonance absorption in uranium-238 due to the Doppler effect (an apparent broadening of the normally narrow resonance peaks due to thermal motion of nuclei). The increase in resonance absorption lowers the resonance escape probability, and the fuel temperature coefficient for resonance escape is negative (explained in detail later). The temperature coefficient of resonance escape probability for the moderator temperature is also negative. As water temperature increases, water density decreases. The decrease in water density allows more resonance energy neutrons to enter

the fuel and be absorbed. The value of the resonance escape probability is always slightly less than one (normally 0.95 to 0.99).

The product of the fast fission factor and the resonance escape probability ($\varepsilon\rho$) is the ratio of the number of fast neutrons that survive slowing down (thermalization) compared to the number of fast neutrons originally starting the generation.

Thermal Utilization Factor, (f)

Once thermalized, the neutrons continue to diffuse throughout the reactor and are subject to absorption by other materials in the reactor as well as the fuel. The thermal utilization factor describes how effectively thermal neutrons are absorbed by the fuel, or how well they are utilized within the reactor. The thermal utilization factor (f) is defined as the ratio of the number of thermal neutrons absorbed in the fuel to the number of thermal neutrons absorbed in any reactor material. This ratio is shown in the following equation.

$$f = \frac{\text{number of thermal neutrons absorbed in the fuel}}{\text{number of thermal neutrons absorbed in all reactor materials}}$$

The thermal utilization factor will always be less than one because some of the thermal neutrons absorbed within the reactor will be absorbed by atoms of non-fuel materials. An equation can be developed for the thermal utilization factor in terms of reaction rates as follows.

$$f = \frac{\Sigma_a^U \phi^U V^U}{\Sigma_a^U \phi^U V^U + \Sigma_a^m \phi^m V^m + \Sigma_a^p \phi^p V^p}$$

The superscripts U, m, and p refer to uranium, moderator, and poison, respectively. In a heterogeneous reactor, the flux will be different in the fuel region than in the moderator region due to the high absorption rate by the fuel. Also, the volumes of fuel, moderator, and poisons will be different. Although not shown in the above equation, other non-fuel materials, such as core construction materials, may absorb neutrons in a heterogeneous reactor. These other materials are often lumped together with the superscript designation OS, for “other stuff.” To be completely accurate, the above equation for the thermal utilization factor should include all neutron-absorbing reactor materials when dealing with heterogeneous reactors. However, for the purposes of this text, the above equation is satisfactory.

In a homogeneous reactor the neutron flux seen by the fuel, moderator, and poisons will be the same. Also, since they are spread throughout the reactor, they all occupy the same volume. This allows the previous equation to be rewritten as follows:

$$f = \frac{\Sigma_a^U}{\Sigma_a^U + \Sigma_a^m + \Sigma_a^p}$$

This equation gives an approximation for a heterogeneous reactor if the fuel and moderator are composed of small elements distributed uniformly throughout the reactor.

Since absorption cross sections vary with temperature, it would appear that the thermal utilization factor would vary with a temperature change. But, substitution of the temperature correction formulas in the above equation will reveal that all terms change by the same amount, and the ratio remains the same. In heterogeneous water-moderated reactors, there is another important factor. When the temperature rises, the water moderator expands, and a significant amount of it will be forced out of the reactor core. This means that N^m, the number of moderator atoms per cm^3, will be reduced, making it less likely for a neutron to be absorbed by a moderator atom. This reduction in N^m results in an increase in thermal utilization as moderator temperature increases because a neutron now has a better chance of hitting a fuel atom. Because of this effect, the temperature coefficient for the thermal utilization factor is positive. The amount of enrichment of uranium-235 and the poison concentration will affect the thermal utilization factor in a similar manner as can be seen from the equation above.

Reproduction Factor, (η)

Most of the neutrons absorbed in the fuel cause fission, but some do not. The reproduction factor (η) is defined as the ratio of the number of fast neutrons produced by thermal fission to the number of thermal neutrons absorbed in the fuel. The reproduction factor is shown below.

$$\eta = \frac{\text{number of fast neutrons produced by thermal fission}}{\text{number of thermal neutrons absorbed in the fuel}}$$

The reproduction factor can also be stated as a ratio of rates as shown below.

$$\eta = \frac{\text{rate of production of fast neutrons by thernal fission}}{\text{rate of absorption of thermal neutrons by the fuel}}$$

The rate of production of fast neutrons by thermal fission can be determined by the product of the fission reaction rate ($\Sigma_f^u \phi^u$) and the average number of neutrons produced per fission (ν). The average number of neutrons released in thermal fission of ^{235}uranium is 2.42. The rate of absorption of thermal neutrons by the fuel is $\Sigma_a^u \phi^u$. Substituting these terms into the equation above results in the following equation.

$$\eta = \frac{\Sigma_f^U \ \phi^U \ \nu}{\Sigma_a^U \ \phi^U}$$

Table 4 lists values of ν and η for fission of several different materials by thermal neutrons and fast neutrons.

Table 4. Average number of neutrons liberated in fission

Fissile Nucleus	Thermal Neutrons		Fast Neutrons	
	ν	η	ν	η
^{233}Uranium	2.49	2.29	2.58	2.40
^{235}Uranium	2.42	2.07	2.51	2.35
^{239}Plutonium	2.93	2.15	3.04	2.90

Source: DOE-HDBK-1019/2-93

As temperature varies, each absorption and fission microscopic cross section varies according to the 1/v relationship. Since both the numerator and the denominator change equally, the net change in η is zero. Therefore, η changes only as ^{235}uranium enrichment changes. η increases with enrichment because there is less ^{238}uranium in the reactor making it more likely that a neutron absorbed in the fuel will be absorbed by ^{235}uranium and cause fission.

To determine the reproduction factor for a single nuclide rather than for a mixture, the calculation may be further simplified to the following.

$$\eta = \frac{\sigma_f \nu}{\sigma_a}$$

Effective Multiplication Factor

The infinite multiplication factor can fully represent only a reactor that is infinitely large, because it assumes that no neutrons leak out of the reactor. To completely describe the neutron life cycle in a real, finite reactor, it is necessary to account for neutrons that leak out. The multiplication factor that takes leakage into account is the effective multiplication *factor* (k_{eff}), which is defined as the ratio of the neutrons produced by fission in one generation to the number of neutrons lost through absorption and leakage in the preceding generation. The effective multiplication factor may be expressed mathematically as shown in the following equation.

$$k_{eff} = \frac{\text{neutron production from fission in one generation}}{\text{neutron absorption in the preceding generation} + \text{neutron leakage in the preceding generation}}$$

So, the value of k_{eff} for a self-sustaining chain reaction of fissions, where the neutron population is neither increasing nor decreasing, is one. The condition where the neutron chain reaction is self-sustaining and the neutron population is neither increasing nor decreasing is referred to as the *critical* condition and can be expressed by the simple equation $k_{eff} = 1$.

If the neutron production is greater than the absorption and leakage, the reactor is called supercritical. In a supercritical reactor, k_{eff} is greater than one, and the neutron flux increases each generation. If, on the other hand, the neutron production is less than the absorption and leakage, the reactor is called subcritical. In a subcritical reactor, k_{eff} is less than one, and the flux decreases each generation.

When the multiplication factor of a reactor is not equal to exactly one, the neutron flux will change and cause a change in the power level. Therefore, it is essential to know more about how this factor depends upon the contents and construction of the reactor. The balance between production of neutrons and their absorption in the core and leakage out of the core determines the value of the multiplication factor. If the leakage is small enough to be neglected, the multiplication factor depends upon only the balance between production and absorption, and is called the infinite multiplication factor ($k\infty$) since an infinitely large core can have no leakage. When the leakage is included, the factor is called k_{eff}.

The effective multiplication factor for a finite reactor may be expressed mathematically in terms of the infinite multiplication factor and two additional factors which account for neutron leakage as shown below.

Fast Non-Leakage Probability ($\mathcal{L}_f$)

In a realistic reactor of finite size, some of the fast neutrons leak out of the boundaries of the reactor core before they begin the slowing-down process. The fast non-leakage probability is defined as the ratio of the number of fast neutrons that do not leak from the reactor core to the number of fast neutrons produced by all fissions. This ratio is stated as follows.

$$\mathcal{L}_f = \frac{\text{number of thermal neutrons that do not leak from reactor}}{\text{number of neutrons that reach thermal energies}}$$

Thermal Non-Leakage Probability ($\mathcal{L}_t$)

Neutrons can also leak out of a finite reactor core after they reach thermal energies. The thermal non-leakage probability is defined as the ratio of the number of thermal neutrons that do not leak from the reactor core to the number of neutrons that reach thermal energies. The thermal non-leakage probability is represented by the following.

$$\mathcal{L}_t = \frac{\text{number of thermal neutrons that do not leak from reactor}}{\text{number of neutrons that reach thermal energies}}$$

The fast non-leakage probability and the thermal non-leakage probability ($\mathcal{L}_t$) may be combined into one term that gives the fraction of all neutrons that do not leak out of the reactor core. This term is called the total non-leakage probability and is given the symbol $\mathcal{L}_T$, where $\mathcal{L}_T = \mathcal{L}_f\mathcal{L}_t$. $\mathcal{L}_f$ and $\mathcal{L}_t$ are both affected by a change in coolant temperature in a heterogeneous water-cooled, water-moderated reactor. As coolant temperature rises, the coolant expands. The density of the moderator is lower; therefore, neutrons must travel farther while slowing down. This effect increases the probability of leakage and thus decreases the non-leakage probability. Consequently, the temperature coefficient (defined later) for the non-leakage probabilities is negative, because as temperature rises, $\mathcal{L}_f$ and $\mathcal{L}_t$ decrease.

Six Factor Formula

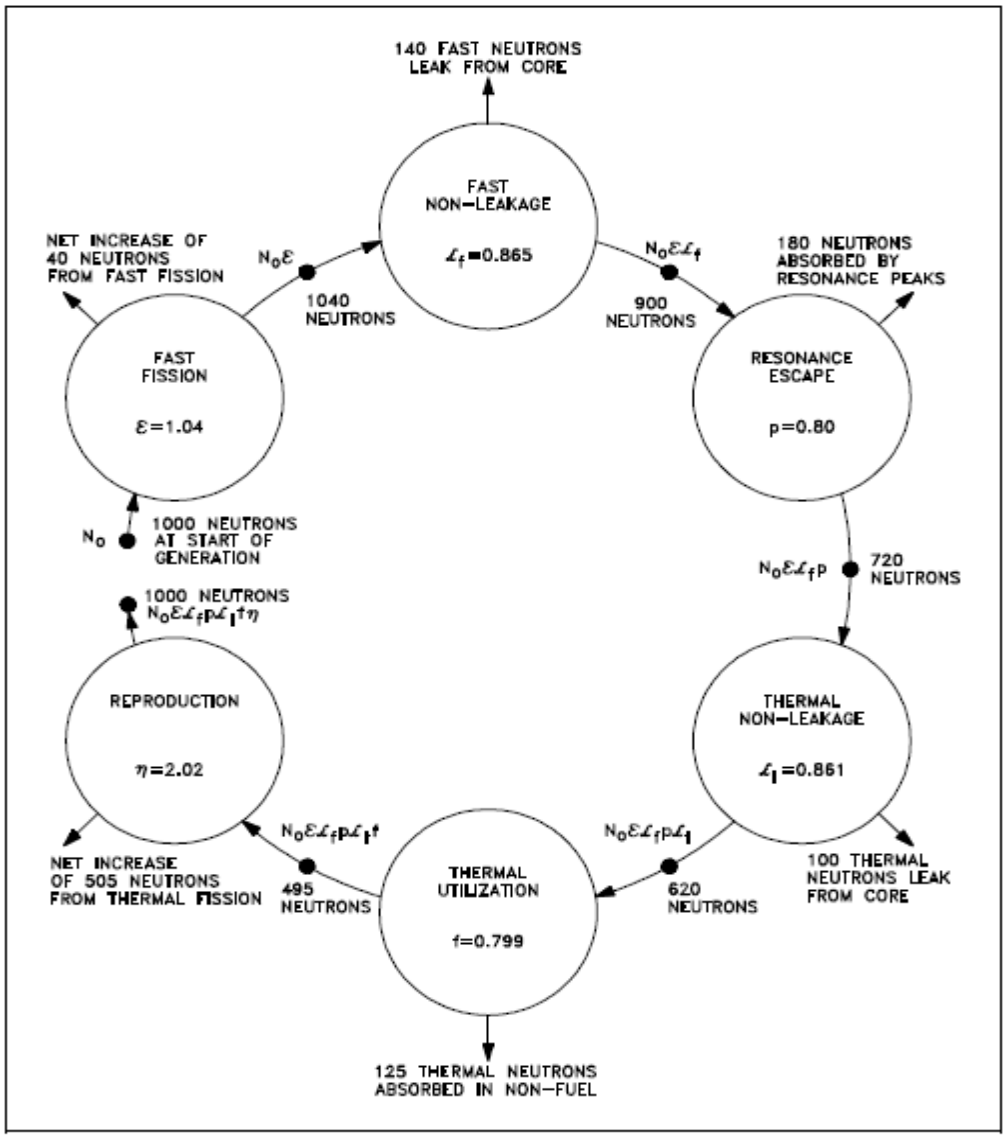
With the inclusion of these last two factors it is possible to determine the fraction of neutrons that remain after every possible process in a nuclear reactor. The effective multiplication factor (k_{eff}) can then be determined by the product of six terms.

$$k_{eff} = \varepsilon\, \mathcal{L}_f\, p\, \mathcal{L}_t\, f\, \eta$$

This equation is called the six factor formula. Using this six factor formula, it is possible to trace the entire neutron life cycle from production by fission to the initiation of subsequent fissions. Figure 16 illustrates a neutron life cycle with nominal values provided for each of the six factors. Refer to figure 16 for the remainder of the discussion on the neutron life cycle and sample calculations. The generation begins with 1000 neutrons. This initial number is represented by N_0. The first process is fast fission and the

population has been increased by the neutrons from this fast fission process. From the definition of the fast fission factor it is possible to calculate its value based on the number of neutrons before and after fast fission occurs.

$$\varepsilon = \frac{\text{number of fast neutrons produced by all fissions}}{\text{number of fast neutrons produced by thermal fissions}}$$
$$= \frac{1040}{1000}$$
$$= 1.04$$



Source: DOE-HDBK-1019/2-93

Figure 16. Neutron life cycle with $K_{eff} = 1$

The total number of fast neutrons produced by thermal and fast fission is represented by the quantity $N_o\varepsilon$. Next, it can be seen that 140 neutrons leak from the core before reaching

the thermal energy range. The fast non-leakage probability is calculated from its definition, as shown below.

$$\mathcal{L}_f = \frac{\text{number of fast neutrons that do not leak from reactor}}{\text{number of fast neutrons produced by fissions}}$$

$$= \frac{1040 - 140}{1040}$$

$$= 0.865$$

The number of neutrons that remain in the core during the slowing down process is represented by the quantity

$$N_o \, \varepsilon \, \mathcal{L}_f.$$

After reaching thermal energies, 100 neutrons leak from the core. The value for $\mathcal{L}_f$ can be calculated by substitution of the known values in the definition as shown below.

$$\mathcal{L}_f = \frac{\text{number of thermal neutrons that do not leak from reactor}}{\text{number of neutrons that reach thermal energies}}$$

$$= \frac{620}{720}$$

$$= 0.861$$

The number of thermal neutrons available for absorption anywhere in the core is represented by the quantity

$$N_o \, \varepsilon \, \mathcal{L}_f \, p \, \mathcal{L}_t.$$

Figure 16 indicates that 125 neutrons were absorbed in non-fuel materials. Since a total of 620 thermal neutrons were absorbed, the number absorbed by the fuel equals 620 - 125 = 495. Therefore, the thermal utilization factor can be calculated as follows.

$$f = \frac{\text{number of thermal neutrons absorbed in the fuel}}{\text{number of thermal neutrons absorbed in any reactor material}}$$

$$= \frac{495}{620}$$

$$= 0.799$$

The final factor numerically describes the production of fission neutrons resulting from thermal neutrons being absorbed in the fuel. This factor is called the reproduction factor (h). The value for the reproduction factor can be determined as shown below.

$$\eta = \frac{\text{number of fast neutrons produced by thermal fission}}{\text{number of thermal neutrons absorbed in the fuel}}$$

$$= \frac{1000}{495}$$

$$= 2.02$$

The number of fission neutrons that exist at the end of the life cycle which are available to start a new generation and cycle is represented by the quantity

$$N_o \, \varepsilon \, \mathcal{L}_f \, p \, \mathcal{L}_t \, f \, \eta.$$

Approach to Critical

Because the sub-critical multiplication factor is related to the value of k_{eff}, it is possible to monitor the approach to criticality through the use of the sub-critical multiplication factor. As positive reactivity is added to a sub-critical reactor, k_{eff} will get nearer to one. As k_{eff} gets nearer to one, the sub-critical multiplication factor (M) gets larger. The closer the reactor is to criticality, the faster M will increase for equal step insertions of positive reactivity. When the reactor becomes critical, M will be infinitely large. For this reason, monitoring and plotting M during an approach to criticality is impractical because there is no value of M at which the reactor clearly becomes critical. Instead of plotting M directly, its inverse (1/M) is plotted on a graph of 1/M versus added reactivity.

Reactivity Insertion

In practice, the reference count rate used is the count rate prior to the beginning of the reactivity change. The startup procedures for many reactors include instructions to insert positive reactivity in incremental steps with delays between the reactivity insertions to allow time for sub-critical multiplication to increase the steady-state neutron population to a new, higher level and allow more accurate plotting of 1/M. The neutron population will typically reach its new steady-state value within 1-2 minutes, but the closer the reactor is to criticality, the longer the time will be to stabilize the neutron population.

Multiplication

The infinite multiplication factor can fully represent only a reactor that is infinitely large because it assumes that no neutrons leak out of the reactor. To completely describe the neutron life cycle in a real, finite reactor, it is necessary to account for neutrons that leak out. The multiplication factor that takes leakage into account is the effective multiplication factor (k_{eff}), which is defined as the ratio of the neutrons produced by fission in one generation to the number of neutrons lost through absorption and leakage in the preceding generation.

So, the value of k_{eff} for a self-sustaining chain reaction of fissions, where the neutron population is neither increasing nor decreasing, is one. The condition where the neutron chain reaction is self-sustaining and the neutron population is neither increasing nor decreasing is referred to as the critical condition and can be expressed by the simple equation $k_{eff} = 1$.

If the neutron production is greater than the absorption and leakage, the reactor is called supercritical. In a supercritical reactor, k_{eff} is greater than one, and the neutron flux increases each generation. If, on the other hand, the neutron production is less than the absorption and leakage, the reactor is called sub-critical. In a sub-critical reactor, k_{eff} is less than one, and the flux decreases each generation.

Reactor Kinetics and Reactivity Changes

The reactor period is defined as the time required for reactor power to change by a factor of "e," where "e" is the base of the natural logarithm and is equal to about 2.718. The reactor period is usually expressed in units of seconds. From the definition of reactor period, it is possible to develop the relationship between reactor power and reactor period that is expressed by

$$P = P_o \, e^{t/\tau}$$

where:
P = transient reactor power
P_o = initial reactor power
τ = reactor period (seconds)
t = time during the reactor transient (seconds)

The smaller the value of τ, the more rapid the change in reactor power. If the reactor period is positive, reactor power is increasing. If the reactor period is negative, reactor power is decreasing.

There are numerous equations used to express reactor period. The first term in this equation is the prompt term and the second term is the delayed term.

$$\tau = \frac{\ell^*}{\rho} + \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho + \dot{\rho}}$$

where:
* = prompt generation lifetime
$\bar{\beta}_{eff}$ = effective delayed neutron fraction
ρ = reactivity
λ_{eff} = effective delay neutron precursor decay constant
= rate of change of reactivity

The delayed neutron fraction, β, is the fraction of all fission neutrons that are born as delayed neutrons. The value of β depends on the actual nuclear fuel used. The delayed neutron precursors for a given type of fuel are grouped on the basis of half-life.

The term (pronounced beta-bar) is the average delayed neutron fraction. The value of is the weighted average of the total delayed neutron fractions of the individual types of fuel. Each total delayed neutron fraction value for each type of fuel is weighted by the percent of total neutrons that the fuel contributes through fission. If the percentage of fissions occurring in the different types of fuel in a reactor changes over the life of the core, the average delayed neutron fraction will also change. For a light water reactor using

low enriched fuel, the average delayed neutron fraction can change from 0.0070 to 0.0055 as uranium-235 is burned out and plutonium-239 is produced from uranium-238.

Delayed neutrons do not have the same properties as prompt neutrons released directly from fission. The average energy of prompt neutrons is about 2 MeV. This is much greater than the average energy of delayed neutrons (about 0.5 MeV). The fact that delayed neutrons are born at lower energies has two significant impacts on the way they proceed through the neutron life cycle. First, delayed neutrons have a much lower probability of causing fast fissions than prompt neutrons because their average energy is less than the minimum required for fast fission to occur. Second, delayed neutrons have a lower probability of leaking out of the core while they are at fast energies because they are born at lower energies and subsequently travel a shorter distance as fast neutrons. These two considerations (lower fast fission factor and higher fast non-leakage probability for delayed neutrons) are taken into account by a term called the importance factor (I). The importance factor relates the average delayed neutron fraction to the effective delayed neutron fraction.

In a small reactor with highly enriched fuel, the increase in fast non-leakage probability will dominate the decrease in the fast fission factor, and the importance factor will be greater than one. In a large reactor with low enriched fuel, the decrease in the fast fission factor will dominate the increase in the fast non-leakage probability and the importance factor will be less than one.

Another term in the reactor period (τ) equation is λ_{eff} (pronounced lambda effective), the effective delayed neutron precursor decay constant. The decay rate for a given delayed neutron precursor can be expressed as the product of precursor concentration and the decay constant (λ) of that precursor. The decay constant of a precursor is simply the fraction of an initial number of the precursor atoms that decays in a given unit of time. A decay constant of 0.1 sec^{-1}, for example, implies that one-tenth, or ten percent, of a sample of precursor atoms decays within one second. The value for the effective delayed neutron precursor decay constant, λ_{eff}, varies depending upon the balance existing between the concentrations of the precursor groups and the nuclide(s) being used as the fuel.

If the reactor is operating at a constant power, all the precursor groups reach an equilibrium value. During an up-power transient, however, the shorter-lived precursors decaying at any given instant were born at a higher power level (or flux level) than the longer-lived precursors decaying at the same instant. There is, therefore, proportionately more of the shorter-lived and fewer of the longer-lived precursors decaying at that given instant than there are at constant power. The value of $\boldsymbol{\lambda_{eff}}$ is closer to that of the shorter-lived precursors.

During a down-power transient the longer-lived precursors become more significant. The longer-lived precursors decaying at a given instant were born at a higher power level (or flux level) than the shorter-lived precursors decaying at that instant. Therefore, proportionately more of the longer-lived precursors are decaying at that instant, and the value of λ_{eff} approaches the values of the longer-lived precursors.

Approximate values for λ_{eff} are 0.08 sec for steady-state operation, 0.1^{-1} sec. for a power increase, and 0.05^{-1}sec. for a power decrease. The exact values will depend upon the materials used for fuel and the value of the reactivity of the reactor.

If the positive reactivity added is less than the value of $_{eff}$, the emission of prompt fission neutrons alone is not sufficient to overcome losses to non-fission absorption and leakage. If delayed neutrons were not being produced, the neutron population would decrease as long as the reactivity of the core has a value less than the effective delayed neutron fraction. The positive reactivity insertion is followed immediately by a small immediate power increase called the prompt jump. This power increase occurs because the rate of production of prompt neutrons changes abruptly as the reactivity is added.

Conversely, in the case where negative reactivity is added to the core, there will be a prompt drop in reactor power. The prompt drop is the small immediate decrease in reactor power caused by the negative reactivity addition.

If the amount of positive reactivity added equals the value of $_{eff}$, the reactor period equation becomes

$$\tau = \frac{\ell^*}{\rho}$$

In this case, the production of prompt neutrons alone is enough to balance neutron losses and increase the neutron population. The condition where the reactor is critical on prompt neutrons, and the neutron population increases as rapidly as the prompt neutron generation lifetime allows, is known as prompt critical. The prompt critical condition does not signal a dramatic change in neutron behavior. The reactor period changes in a regular mannerbetween reactivities above and below this reference. Prompt critical is, however, a convenient condition for marking the transition from delayed neutron to prompt neutron time scales. A reactor whose reactivity even approaches prompt critical is likely to suffer damage due to the rapid rise in power to a very high level. For example, a reactor that has gone prompt critical could experience a several thousand percent power increase in less than one second.

Because the prompt critical condition is so important, a specific unit of reactivity has been defined that relates to it. The unit of reactivity is the dollar ($), where one dollar of reactivity is equivalent to the effective delayed neutron fraction $_{eff}$. A reactivity unit related to the dollar is the cent, where one cent is one-hundredth of a dollar. If the reactivity of the core is one dollar, the reactor is prompt critical. Because the effective delayed neutron fraction is dependent upon the nuclides used as fuel, the value of the dollar is also dependent on the nuclides used as fuel.

For normal reactor operating conditions, the value of positive reactivity in the reactor is never permitted to approach the effective delayed neutron fraction, and the reactor period equation is normally written as follows:

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho + \dot{\rho}}$$

This equation is referred to as the transient period equation since it incorporates the term to account for the changing amount of reactivity in the core. The $^{*}/\rho$ term (prompt period) is normally negligible with respect to the remainder of the equation and is often not included.

For conditions when the amount of reactivity in the core is constant (= 0) and the reactor period is unchanging, this equation can be simplified further to the following equation which is known as the stable period equation:

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho}$$

c. Participate in a criticality experiment demonstration.

This is a performance-based competency. The Qualifying Official will evaluate its completion.

11. Criticality safety personnel must demonstrate both a working level knowledge of computer codes used in criticality safety evaluations, and must have demonstrated the ability to use such methods.

a. Develop input model for one Monte Carlo and one deterministic code or code system (e.g., MONK, VIM, SCALE/KENO, MCNP, PARTISN, SCALE/XSDRN, and COG).

This is a performance-based competency. The Qualifying Official will evaluate its completion.

b. Describe how cross section data impact Monte Carlo and deterministic codes.

The following provided by a team of subject matter experts at the NNSA Service Center.

Cross section data are the fundamental physical data on which Monte Carlo and deterministic codes rest. The cross section is the probability of a particular interaction between a particular nucleus and a neutron of a particular energy. The basis of the physical model in Monte Carlo is converting this data to probability density functions and collecting the tallies to form the solution to the integral equations. The cross section data are also fundamental to the deterministic codes. Although they use a different mathematical algorithm and do not use the stochastic methods, the neutron-nuclei interactions are still the basis of the code.

c. Discuss the advantages and disadvantages of point-wise continuous and multi-group cross-section data.

The following was derived from the MCNP5 Manual by a team of subject matter experts at the NNSA Service Center.

Point-wise continuous cross sections can most accurately calculate the interaction effect of the various cross sections of the materials of concern. This is particularly important in the resonance regions of the cross sections. However, calculations with multi-group cross-section sets require much less computer memory and are much quicker.

d. Briefly discuss the effect of geometry and spectral assumptions on the development of multi-group cross sections.

The following was provided by a team of subject matter experts at the NNSA Service Center.

All multi-group cross section sets are developed by collapsing larger sets of data by using simplifying assumptions. These assumptions include geometry and spectral effects. If the assumed spectrum, geometry, and spatial effects are incorrect for the system being calculated, significant errors can occur.

e. Describe the importance of validation of computer codes and how it is accomplished. (See competency 19, ANSI/ANS 8.1 & 8.24)

The following was derived from ANSI/ANS 8.1 and 8.24 by a team of subject matter experts at the NNSA Service Center.

The codes are only theoretical without validation. Validation is the comparison of the code results to known conditions. It is accomplished by modeling known critical conditions, known in validation practice as benchmarks. The results of the code vs. the benchmarks are compared, the variances in the differences are analyzed, and trends are searched out. Typical parameters are main fissile isotope, main diluent, reflectors, and some measure of energy range. Bias, variance in the bias, experimental uncertainties, and code variance are also considered. This work is often reduced to a maximum acceptable K_{eff}, or preferably a maximum acceptable k_{eff}, for each type of system likely to be analyzed by the validator (person or organization). When this is done, the proper goal is to have a maximum acceptable K_{eff}, such that the validator (and user, if different) are highly confident that the maximum acceptable K_{eff}, is subcritical.

f. Describe the methodology supporting Monte Carlo codes and deterministic codes.

The following is taken from NCSET modules 6 and 7.

Diffusion theory is derived from the equations of continuity, and is applicable to many types of systems, with the caveat that at each boundary a new diffusion equation must be applied. If there are too many discontinuities, the solutions become intractable. Also, diffusion theory does not treat any angular dependence of neutron reactions, which further degrades accuracy near material discontinuities. In spite of failing to treat angular dependence correctly, diffusion theory works well with large thermal systems.

Deterministic codes expand on the method of solving the diffusion equation in terms of spatial mesh and energy groups by adding an explicit consideration of the directional properties.

For the discrete ordinates approximation, the cosine of the polar angle, $\mu = \cos\theta$, is divided into a finite number of intervals and the angular distribution is assumed constant over each interval. The discrete ordinates approximation has led to a large family of computer codes, commonly called the S_n approximations, such as DTF, ANISN, XSDRN, DOT, TWOTRAN, TWODANT, THREEDANT, PARTISN, and TORT. The higher the

level of approximation, S4, S8, etc., the better the approximation. The S_n codes are most useful to the criticality specialist for calculating simple geometries in one dimension (e.g., doing parametric studies for spheres, long cylinders or large slabs).

Monte Carlo codes solve the transport equation by tracking an individual particle from reaction to reaction, with direction, distance between reactions, and reaction type estimated using probability density functions. A large number of particles are tracked, and the reactions tallied, and the averages and variance of the reaction type or number of particles produced is used to estimate the solution of the equation or fundamental eigen value (e.g. K_{eff}).

g. Describe advantages and pitfalls of Monte Carlo calculations and deterministic codes.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Deterministic codes are much faster than Monte Carlo, due to the fewer number of calculations required to iterate to a solution. They are limited by the approximations of spatial mesh, directional grouping, and energy grouping. Properly validated, they can be useful for a wide range of single unit problems. They can be misleading if the group structure of the cross sections has not been properly derived for the energy spectrum of the system being calculated. The limitation on energy spectrum is somewhat mitigated by using a large number of groups.

Monte Carlo codes have the advantage of being able to handle almost any geometry. Some codes also use point-wise continuous cross section data, avoiding the need to derive the group structures for an energy spectrum. Their disadvantages are that they can be treated like a black box, and some understanding of statistics is necessary to understand the output. Further, systems can be modeled which the code may calculate incorrectly, especially if important regions are not sampled with particles. This can be detected only by correctly designing the input to provide some output check of the source sampling of each important region. Also, if review is done by reviewing the input deck, errors in geometry and material descriptions are easy to miss.

h. The diffusion theory model is not strictly valid for treating fissile systems in which neutron absorption, voids, and/or material boundaries are present. In the context of these limitations, identify a fissile system for which a diffusion theory solution would be adequate.

The following is taken from LA-14244-M.

Diffusion theory is appropriate for systems without wide variations in neutron energy, or where most of the neutrons are at the same energy and there are minimal discontinuities in the material compositions. Hence, diffusion theory works well for large homogenous systems. Early reactor design was done using diffusion theory and homogenization approximations.

Diffusion theory is derived from the equations of continuity and is applicable to many types of systems with the caveat that, at each boundary, a new diffusion equation must be applied. If there are too many discontinuities, the solutions become intractable. Also, diffusion theory does not treat any angular dependence of neutron reactions, which further degrades accuracy near material discontinuities. Diffusion theory does work well with large thermal systems.

i. Discuss the *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, including its purpose, accessibility, and application to computer code validation.

The following is taken from the International Criticality Safety Benchmark Evaluation Project website.

The *International Handbook of Evaluated Criticality Safety Benchmark Experiments* was prepared by a working group comprised of experienced criticality safety personnel from the United States, the United Kingdom, Japan, the Russian Federation, France, Hungary, Republic of Korea, Slovenia, Serbia, Kazakhstan, Israel, Spain, Brazil, Czech Republic, Poland, India, Canada, China, Sweden and Argentina. The handbook contains criticality safety benchmark specifications that have been derived from experiments that were performed at various nuclear critical facilities around the world. The benchmark specifications are intended for use by criticality safety engineers to validate calculational techniques used to establish minimum subcritical margins for operations with fissile material. The example calculations presented do not constitute a validation of the codes or cross section data.

The handbook contains 516 evaluations with benchmark specifications for 4405 critical, near-critical, or subcritical configurations and 24 criticality alarm placement/shielding configurations with multiple dose points for each, and 200 configurations that have been categorized as fundamental physics measurements that are relevant to criticality safety applications. Experiments that are found unacceptable for use as criticality safety benchmark experiments are discussed in the handbook evaluations; however, benchmark specifications are not derived for such experiments. Approximately 770 experimental configurations are categorized as unacceptable for use as criticality safety benchmark experiments. Additional evaluations are in progress and will be added to the handbook periodically.

12. Criticality safety personnel must demonstrate both a working level knowledge of development of criticality safety evaluations, and the ability to develop such evaluations.

a. Prepare criticality safety evaluations for two different applications selected from those listed in h., i., and k. below.

This is a performance-based competency. The Qualifying Official will evaluate its completion.

b. Describe development of contingency analysis and controls.

The following is taken from DOE-STD-3007-2007.

This section and its contents shall be included in the CSE. All normal conditions and credible abnormal conditions (credible contingencies) shall be analyzed and documented. This section shall document that operations are subcritical under all normal conditions and that no credible abnormal condition can lead to an accidental criticality. This section shall identify those controls that have been developed.

All credible contingencies shall be identified, analyzed, and documented. The following three basic steps should be included in performing the contingency analysis:

1. Know the operation and system being evaluated. The criticality safety engineer should directly observe the processes and equipment if they currently exist. Facility and equipment drawings should be reviewed as well as process flow sheets or descriptions. The safety analysis for the facility (documented safety analysis (DSA) or safety analysis report) is an appropriate source of information on failure modes which should be considered, such as sprinkler activation, glovebox rupture, rack collapse and natural phenomena, as potential initiators of criticality accidents.
2. Identify potential contingency scenarios. Contingencies shall be identified. A disciplined method should be used to identify contingencies. Examples of acceptable methods (see brief descriptions in DOE-STD-1027-92, CN 1, section 4.1) are
   - what if methods;
   - qualitative event or fault trees;
   - quantitative probabilistic risk assessment (PRA) methods;
   - hazard and operability analysis (HAZOP); and/or
   - failure modes and effects analysis (FMEA).

   Input should be obtained from operations personnel and process specialists thoroughly familiar with the operations and possible abnormal conditions. Whenever practical, identified contingencies should be eliminated by modification of the process.

   Establish controls. Controlled parameters and their associated limits shall be identified in this section. Examples of parameters subject to control include, but are not limited to, fissionable material mass, volume, concentration, moderation, interaction, etc. Appropriate operations staff, engineering staff, and/or process experts should review the postulated contingencies and the proposed controls to ensure practicality. The preferred hierarchy of controls shall be: () passive engineered features, 2) active engineered features, and 3) ACs. Inspections, periodic surveillances, or other QA measures should be developed and implemented to defend the reliability of the selected controls. Other factors that influence the selection of controls and should be considered, include the

   - implementation complexity of the control;
   - ability of personnel to recognize the failure of the control;
   - potential for common mode failure of controls; and
   - final reliability of the set of controls.

The unlikely changes in process conditions (contingencies) and the controls that make these abnormal changes unlikely may be summarized in a table. Table 5 is an example of a contingency control table.

Table 5. Example contingency control table

Controlled Parameter	Contingency (i.e., unlikely process condition)	Controls
Mass	Single double batched container in the workstation in an array of properly loaded containers.	1. Procedure requiring mass measurement 2. Prejob review of applicable criticality safety limits 3. Procedure requiring mass to be checked prior to transfer of material 4. Procedure requiring mass to be verified by independent operator prior to beginning operation
Interaction	Two containers in contact in one workstation immediately adjacent to two containers in contact in adjoining workstation making a total of four containers in contact.	1. Prejob review of applicable criticality safety limits 2. PC-3 qualified rack holding a maximum of three approved containers 3. Procedures and limits permit only one container out of the rack at a time in each workstation 4. Seismically qualified racks

Source: DOE-STD-3007-2007

c. Describe practical ways to minimize the use of administrative controls, and discuss how to evaluate whether the evaluation develops a proper mix of engineered and administrative controls.

The following is taken from NCSET, module 12.

Understanding of the operator and supervisor knowledge, training process, procedure use and level of conduct of operations employed at the facility is also needed to determine the level of AC implementation. Other human factors issues also need to be understood to judge the frequency of failure of ACs that may be placed on the operation. For example, an AC that prevents drums from being stacked is less likely to be violated if drums are not stacked anywhere in the facility compared to if some drums types are stacked and some are not.

The analyst should hold discussions with operations staff throughout the nuclear criticality safety evaluation (NCSE) development process to obtain their concurrence with the controls and limits. Controls proposed by the analyst may conflict with requirements imposed on the operation by other disciplines and/or their experience may lead to a better control scheme. Determination that the controls cannot be implemented in the facility after the NCSE is written and peer reviewed will result in significant additional work and associated costs.

Other site programs and requirements should also be considered when writing NCS controls. The robustness of conduct of operations, configuration management, and other programs at the site is a factor in how the controls are implemented. For example, inspection criteria for passive design features may need to be stated in the NCSE unless the configuration management program has an element for assessing degradation of such components. Also, NCS controls need to be robust to ensure that the upset conditions evaluated as contingencies are indeed unlikely.

This determination often requires knowledge of the NCS program within the facility. For example, if drums are not allowed to be stacked anywhere in the facility, the upset of an operator stacking drums is more unlikely, due to the consistently applied limit and its reinforcement in training, than if drums are stacked in one process but not in another in the same facility. Where ACs are complicated or inconsistent, the criticality safety program (CSP) should consider whether independent verification of control compliance is needed, and if required, explicitly state it in the control.

The following is taken from DOE-STD-3007-2007

A process to examine the collection of controls developed in the CSEs to determine their importance in DSA space should be developed and documented as part of the CSP. The process should be agreed upon by both the NCS and safety analysis staff and formally documented as part of the CSP description document.

Eight evaluation criteria that should be used in this process are listed below:

1. The selection of NCS controls for the DSA should be performed using a team of criticality safety, nuclear safety, and operations personnel.
2. The consequence of criticality should be examined for the purpose of establishing whether a particular control is safety class or safety significant, equipment important to safety, or merely provides defense-in-depth.
3. The CSEs that cover the fissionable material operations addressed by the DSA should be examined to ensure that bounding assumptions or analysis conditions are considered as potential DSA/TSR (technical safety requirement) controls.
4. All passive engineered features credited in the CSE should be considered for selection as a DSA design feature.
5. All active engineered features credited in the CSE should be considered for selection as a safety class or safety significant structure, system, or component (SSC) with associated safety limits (SLs) or limiting condition of operations as appropriate.
6. Not all engineered features need be selected for inclusion in the DSA/TSR (technical safety requirement). The minimum set of controls selected for inclusion should be those that meet the following conditions: (A) loss of the single control under consideration could directly result in a criticality accident, (B) loss of the control could result in a singly contingent condition, and (C) active controls requiring calibration. Additional engineered features beyond the minimum set may be selected as appropriate.
7. If all of the credible scenarios are shown to be subcritical by engineered features, then specific administrative controls (SACs) are not required.
8. Only ACs meeting at least one of the following criteria should be considered for inclusion: (A) credible violation of the control could directly lead to a criticality

accident, (B) if the safety function were to be equivalent to a safety class or safety significant engineered control, and (C) general references to control philosophy.

d. Give an example of a practical method for controlling each of the following parameters:

- **Mass**
- **Absorption**
- **Geometry**
- **Interaction**
- **Concentration**
- **Moderation**
- **Enrichment**
- **Reflection**
- **Volume**

The following is taken from NCSET, module 12, and describes controlling the above parameters during the criticality safety analysis for a beaker leaching (BL) process. Refer to module 12 for additional detail and graphics.

Mass

Uranium mass is controlled for the dissolving and filtering operations by limiting the amount of feed material that can enter the process; each dissolution workstation is limited to one feed unit. The overall uranium mass in the hood is not controlled because the uranium mass in the pans on the shelf is unquantifiable.

The uranium mass in a feed unit is limited by the NCS requirements on the operation that generated that unit, not the BL process (with the exception of the BL roughing filters). CSE-CONT also imposes loading limits for feed units transported to the process. All approved feed units have been demonstrated to be subcritical when fully water-reflected in CSE-CONT.

Absorption

Absorption is not controlled. The process only contains materials that act as mild poisons for moderated systems and these poisons are not credited. Note that the hood stainless steel floor is explicitly modeled. In this case modeling the floor is more reactive than modeling nothing below the containers.

Geometry

Geometry is controlled in the BL process. In addition to common process containers that are controlled by CSE-CONT, the geometry of other process-specific pieces of equipment is controlled. The geometrically favorable pieces of equipment include: hoods that have drain holes limiting solution depth to three inches, wet vacuum traps that are no more than 4.625 inches in outer diameter, and hood exhaust filters that have bottom heights no less than 9.5 inches above the floor of the hood. The leachate columns have also been shown to remain subcritical based on geometry. An additional geometry control requires plastic bags to have corner slits to minimize potential accumulation of uranium solution.

Interaction

Interaction is controlled in the BL process. The hoods are equipped with dividers that passively provide a minimum of 7.375 inches of spacing between adjacent workstations; however, solution can pass under the dividers in the event of a spill or process upset. The minimum spacing between workstations in adjacent hoods is 5.5 inches Interaction between containers in a single workstation is not controlled. Spacing is provided between the stored pans and process containers by the height of the pan shelves above the hood floor. The leachate columns are installed with 23.5 inch center-to-center (c-t-c) spacing and the top of the wet vacuum trap is greater than 29 inches below the leachate columns. The c-t-c spacing between the wet vacuum trap and the safe bottle area is greater than 20 inches. One safe bottle may also be brought to each hood and placed in the fixed position holder with no additional spacing requirements. No other fixed uranium bearing equipment is located within four feet of the BL hoods.

Concentration

The concentration of uranium in solution and the density of solid forms of uranium are not controlled. Uranium solutions are modeled at optimum concentration as confirmed by KENO-V.a calculations and dry forms of uranium are evaluated at maximum credible densities.

Moderation

Moderation is controlled for some feed cans loadings per CSE-CONT but is not controlled in the hoods. The feed can is considered to be properly loaded under normal conditions. The degree of moderation is not controlled for feed added to the dissolver beaker. The uranyl nitrate produced from dissolution and collected in the dissolver beaker, 1.86-L Buchner funnel/filter combination, 4-L Erlenmeyer flask, wet vacuum traps, leachate columns, sample bottles, and safe bottles are modeled at optimum concentration (see section 5.1.5 of this module). Pans are conservatively modeled as filled with optimum concentration solution although the contents are either dry such as during feed sorting, or dry out over time, such as filtered solids from the Buchner funnel.

Enrichment

The enrichment of uranium materials processed and handled within the BL process is not controlled. For the purposes of this evaluation, the uranium within the process is no more than 97.7 weight percent enriched in 235U due to the upper limit of material available at the site. KENO-V.a calculations were performed with 100 percent U-235. Critical experiments and some handbook data were used to evaluate 93.2 weight percent enriched U. The difference between the subcritical limits for 93.2 weight percent and 97.7 weight percent is minimal.

Reflection

Reflection is limited for the purposes of NCS due to the location of process equipment used within the BL hoods but is not controlled. The hoods are located 27.125 ± 0.25 inches above the room floor limiting reflection from the concrete floor. The hoods are located near concrete walls, but the thickness of the exhaust plenums in the back of the hoods places the workstations at least 12 inches away from the rear wall. Hood B is also

located more than 12 inches from the side wall. The hood walls and floor are constructed of thin stainless steel (i.e., ≤ 0.1875 inch thick).

During normal operations, uranium-bearing equipment on the periphery of the hoods is nearly unreflected. BL process equipment within the hoods may be incidentally reflected. Incidental reflection includes operator body parts, the thin-walled stainless steel of the hood, distant room walls or floor, etc. Of these reflectors in proximity to uranium, none are infinite in effective thickness. The concrete walls and floors may be thick but are far enough from the uranium on the hood floor that their effectiveness as a reflector is diminished. Thus the nominal reflection modeled is bounding of the credible reflection that could occur. A box of water is modeled around the leachate columns and the containers in the hoods with the exception of the hood floor which is modeled as 0.1875 inch thick stainless steel. The hood floor is more than 2 feet above the concrete floor and nothing is stored under the hoods. The hood floor is dry under normal conditions based on the nature of the process; spills or other events that could result in liquid accumulation are discussed in section 5.3 of NCSET, module 12.

Volume

Volume is controlled in the BL process. The dissolution workstation is limited to one feed unit, one 4-L dissolver beaker, one 4-L "clean" beaker for 30 percent nitric acid or water, and one pan for sorting feed materials on the hood floor (additional pans may be on the shelf).

The filtration workstation is limited to the 4-L dissolver beaker that is brought over from the dissolution workstation, one 1.86-L Buchner funnel/filter combination, one 4-L Erlenmeyer flask, one non-shelved pan for the collection of filtered solids, and one sample bottle. The feed unit is not permitted at the filtration workstation.

See ANSI/ANS-8.1, appendix A for additional information.

e. Describe key personnel in preparation of criticality safety evaluations and determination of process upsets.

The following is taken from NCSET Module 12. See ANSI/ANS-8.19 and 8.26 for additional information.

The key personnel are the operations staff who actually do the operation, line management, and the criticality safety specialist, in that order.

The criticality safety professional must understand the operations that will be analyzed in the NCSE. Often the criticality safety specialist will rely on operations personnel to describe the key points of the operations. Many times the criticality safety specialist will rely on the operations people to identify potential abnormal and accident conditions. However, the criticality safety specialist must also have a first-hand understanding of the facility, operations, and processes, do his own analysis, and then discuss the results with operations personnel. Ongoing interactions with the operations staff and the operators are a necessary part of performing a CSE. It is important that the operations staff owns the safety of the operation, rather than the criticality safety specialist.

f. Describe how subcritical margins and limits are determined.

The following is taken from DOE-STD-3007-2007.

Section D of the CSE describes the methodology or methodologies used to establish limits for the operation being evaluated. Four methods may be used for the establishment of subcritical limits:

1. Reference to national consensus standards that present critical and/or subcritical limits
2. Reference to accepted handbooks of critical and/or subcritical limits
3. Reference to experiments with appropriate adjustments to ensure subcriticality when the uncertainties of parameters reported in the experiment documentation are considered
4. Use of validated calculational techniques (Note: One specific example based on validated calculations that may be used is the criticality index (CI) technique useful for setting limits on commingled arrays of fissile material containers.)

Note that standards and handbooks provide critical data and subcritical limits, which may not include applied safety margins. The analyst shall develop and document margins to be applied to the limits for the operation being evaluated when using values from referenced standards and handbooks to protect against uncertainties in process variables and against a limit being accidentally exceeded. When limits are based on reference documents such as ANSI/ANS-8 standards, criticality safety handbooks, or published experimental results, complete and specific references shall be cited. The applicability of the reference data to the operation being evaluated should be discussed.

Calculation methods may include simple hand calculation techniques (e.g., limiting surface density, density analog), deterministic computer codes (e.g., ANISN or other Sn transport theory codes), and Monte Carlo computer codes (e.g., MCNP, KENO-V.a). When applicable, nuclear cross-section data that were used should be identified (i.e., cross-section sets and release versions) along with any cross section processing codes that were used. References may be provided to allow a reviewer the opportunity to further research the methods used in the evaluation. When computer codes are used as part of the methodology, the type of computing platform along with relevant code configuration control information should be documented here. This information may be provided by reference.

ANSI/ANS-8.1, section 4.3 contains requirements for validation of methods and determination of bias. In addition, validation requirements from other ANSI/ANS-8 standards should also be followed as appropriate. Compliance with these requirements shall be demonstrated in this subsection of the evaluation. Reference may be made to more detailed validation reports; however, the results of these validation reports should be summarized.

If no benchmark experiments exist that match the system being evaluated, it may be possible to interpolate or extrapolate from existing benchmark data to that system. Sensitivity and uncertainty analysis tools may be used to assess the applicability of benchmark problems to the system being analyzed. One example of such a tool is TSUNAMI (tools for sensitivity and uncertainty analysis methodology implementation).

Proper application of TSUNAMI requires covariance data. The analyst should understand the quality of the underlying nuclear physics data utilized in the application, including the covariance data.

g. Describe when validation and bias estimates must be considered, and when they may be disregarded.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Validation and bias should be considered in any evaluation other than direct comparison to experiment.

Several recognized handbooks and national standards have done this already for certain conditions, so validation and bias are typically not discussed in the evaluation. Many hand calculation methods are also developed to be conservative.

For further guidance, see ANSI/ANS 8.1 and ANSI/ANS 8.24.

h. Describe considerations when evaluating various fissile processes, including common process upsets:
- **Aqueous**
- **Metal**
- **Recovery**
- **Fabrication/Foundry**
- **Mixed waste**

The following was provided by a team of subject matter experts at the NNSA Service Center.

For further guidance, see DOE-STD-3007-2007, NCSET modules 10, 11 and 12, and ANSI/ANS 8.1, appendix A.

Aqueous

Considerations when evaluating aqueous processes include concentration changes, precipitation, shape changes, and phase-to-phase transfer as in the 1958 Los Alamos accident and the 1970 Windscale accident.

Metal

Considerations when evaluating metal processes include shape changes, melting, a potential for a mixed moderator, mass control failures, and a potential for reflectors more effective than a foot of water.

Recovery

Particular attention must be paid to the chemistry of the process, and where separations and concentration changes occur. Considerations when evaluating recovery processes include concentration changes, precipitation, shape changes, and phase-to-phase transfer as in the 1958 Los Alamos accident and the 1970 Windscale accident. Pyrochemistry may

have its own set of problems, some related to cross sections, unless very recent data (chlorine) or when very dense reflectors (tantalum or tungsten) are used.

Fabrication/Foundry

Considerations when evaluating fabrication/foundry processes are very much the same as for metal processes, except that shape changes and phase changes are the norm. Particular attention should be paid to loss of geometry control, as many foundry operations have used masses larger than the minimum bare critical mass.

Mixed Waste

The primary considerations when evaluating mixed waste processes are where the material is located and what the potential moderators and reflectors could be. Mixed waste is more likely than non-mixed waste to have unusual reflection conditions, and moderators may be more effective than water or polyethylene. It is common to focus on large arrays for potential problems when the real criticality problem is in the potential of a few items that are close together to result in excess mass.

i. Describe considerations for evaluating the following material storage:

- **Pits**
- **Waste**
- **Fuel elements**
- **Solutions**
- **Metal parts**

The following was provided by a team of subject matter experts at the NNSA Service Center.

For further guidance, see DOE-STD-3007-2007, NCSET modules 10, 11 and 12, and ANSI/ANS 8.1, appendix A.

Pits

The primary issues concerning material storage and pits are how many, how close, and how many are handled at once. Single unit reflection is usually not a problem. ANSI/ANS 8.7 can be quite useful. This guide is applicable to the storage of fissile materials. Mass and spacing limits are tabulated for uranium containing greater than 30 weight-percent ^{235}U, for ^{233}U, and for plutonium as metals and oxides. Criteria for the range of application of these limits are provided.

Waste

The primary considerations for evaluating the storage of waste are where the material will be stored and what the potential moderators and reflectors could be. It is common to focus on large arrays when the real criticality problem is in the potential for a few items to be situated close together, perhaps with excess mass.

Fuel Elements

Proper consideration of isotopics and burnup include depletion of poisons, fuel composition and dimensions, lattice parameters, cladding materials, and potential

moderator conditions (fuel lattices especially are likely to show a K_{eff} spike at very low moderator densities).

Solutions

The biggest concern with storage of solutions is their propensity to go where they are not desired. There is a potential for slow and difficult-to-detect concentration changes, precipitation, shape changes, and phase to phase transfer as in the 1958 Los Alamos accident and the 1970 Windscale accident. If poisons are used, poison stability may be a concern. Continued integrity of containment is a concern, especially with bottles and slab tanks.

Metal parts

Geometry control is usually effective. Spacing, and susceptibility to seismic issues may be a concern, depending on the type of part (i.e., massive or bulky). Both U and Pu are pyrophoric so fire and fire suppression are a concern as is geometry change from metal to oxide. Both metals tend to oxidize in storage, so loss of geometry control must be considered.

j. Discuss elements of industry reference material:

- **LA-10860-MS, *Critical Dimensions of Systems Containing U235, Pu239, and U233*, 1986**
- **LA-12808, *Nuclear Criticality Safety Guide*, 1996**
- **BNWL-SA-4868, or successor PNL-SA-4868, *Anomalies of Criticality***
- **LA-11627-MS, *Glossary of Nuclear Criticality Terms***

LA-10860-MS, Critical Dimensions of Systems Containing U235, Pu239 and U233, 1986

This reference contains summaries of experimental data both in tabular and curve formats, and methods to compare application conditions to the conditions shown in the report.

LA-10860-MS is primarily a compilation of critical data obtained from experiments performed in a number of laboratories during the period from 1945 through 1985. It supplements the Nuclear Safety Guide, Report TID-7016 (Rev. 2) in presenting critical data on which recommendations of the guide are based.

A fundamental aim of LA-10860-MS is to illustrate relationships among critical data. The compilation and correlation of data for this purpose, from many measurements in a number of laboratories, requires a certain amount of normalization or reduction to common terms. Frequently, for example, the effects of variations in geometry or density must be removed to show trends in data. The manner in which these alterations may be made is discussed in the section (beginning on page 13 of LA-10860-MS) entitled Relations for Conversion to Standard Conditions.

LA-12808, Nuclear Criticality Safety Guide, 1996

This guide is the fifth edition in the series. It contains subcritical data in graphical format, some subcritical data which closely matches that in the ANS Standards (it was the same at the time the guide was published), discussion of how criticality controls and a CSP should work, a summary of criticality accidents, and some discussion of criticality alarms.

LA-12808 is, in spirit, revision 3 of TlD-7016, *Nuclear Safety Guide*. Due to changes in the U.S. regulatory climate since the appearance of TID-7016, *Nuclear Safety Guide*, revision 2, it was determined that a formal revision was not possible and a decision was made to change the title to *Nuclear Criticality Safety Guide* to better reflect the scope of the document. This document corrects all known errors in the previous TID-7016 series and incorporates many changes that have been suggested by the criticality safety community.

TlD-7016, *Nuclear Safety Guide*, published in 1957, allowed nuclear criticality data to be made available outside the family of the Atomic Energy Commission installations as a result of declassification. Revision 1 of TID-7016, four years later, was primarily a refinement based upon experience with the document. An accumulated wealth of experimental data and computational results led to revision 2 in 1978.

During the two decades before the publication of LA-12808, little new experimental information was reported, but abundant computational effort was made. Stimulated by the ANS Nuclear Criticality Safety Division, criticality-control problems and their resolution have been frequent topics of discussion. Consequently, this document incorporates little new experimental data, but incorporates modifications intended to extend the document's usefulness. It remains directed toward beginning criticality safety specialists who do not have the traditional background.

BNWL-SA-4868, Anomalies of Criticality

BNWL-SA-4868 (and its successor, PNL-SA-4868) provides a discussion of many of the fundamentals of criticality, a fair amount of criticality history, and a number of illustrations where conditions that might seem sub-critical are actually critical. Many of these situations are counter-intuitive unless the intuition has been honed by thousands of calculations or experiments.

LA-11627-MS, Glossary of Nuclear Criticality Terms

LA-11627-MS is a short glossary of commonly used criticality safety terms, with commonly misused terms emphasized. Terms that have not been widely accepted, such as "fissible" are not included.

k. Describe elements to consider when preparing a Safety Analysis Report for Packaging (SARP). Reference 10 CFR 71 and 49 CFR 173.

The following is taken from Lawrence Livermore National Laboratory, UCID-21218, Rev. 3.

A SARP should contain nine chapters. A summary of these chapters is provided below.

Chapter 1: General Information

- Introduction
  - Purpose of application
  - Summary information
  - Statement of compliance
  - Summary of evaluation

- Package Description
  - Packaging
  - Contents
  - Special requirements for plutonium
  - Operational features

- Appendices
  - Drawings
  - Other information

Chapter 2: Structural

- Description of structural design
  - Design features
  - Codes and standards

- Materials of construction
  - Material specifications and properties
  - Prevention of chemical, galvanic, or other reactions
  - Effects of radiation on materials

- Fabrication, assembly, and examination
  - Fabrication and assembly
  - Examination

- General considerations for structural evaluations
  - Evaluation by test
  - Evaluation by analysis

- Lifting and tie-down standards for all packages
  - Lifting devices
  - Tie-down devices

- Structural evaluation for normal conditions of transport
  - Heat
  - Cold
  - Reduced external pressure
  - Increased external pressure
  - Vibration
  - Water spray
  - Free drop
  - Corner drop
  - Compression
  - Penetration
  - Structural requirements for fissile material packages

- Structural evaluation for hypothetical accident conditions
  - Free drop
  - Crush
  - Puncture

- Thermal
- Immersion—fissile material
- Immersion—all packages

- Structural evaluation of special pressure conditions
  - Special requirement for type B packages containing more than $10^5 A_2$
  - Analysis of pressure test

- Appendices

Chapter 3: Thermal

- Description of thermal design
  - Design features
  - Decay heat of contents
  - Codes and standards
  - Summary tables of temperatures
  - Summary table of maximum pressures

- Material properties, temperature limits, and component specifications
  - Material properties
  - Temperature limits
  - Component specifications

- General considerations for thermal evaluations
  - Evaluation by test
  - Evaluation by analysis

- Thermal evaluation under normal conditions of transport
  - Initial conditions
  - Effects of tests
  - Maximum and minimum temperatures
  - Maximum normal operating pressure
  - Maximum thermal stresses

- Thermal evaluation under hypothetical accident conditions
  - Initial conditions
  - Effects of thermal tests
  - Maximum temperatures and pressures
  - Maximum thermal stresses

- Thermal evaluation of maximum accessible surface temperature
- Appendices
  - Description of test facilities and equipment
  - Test results
  - Applicable supporting documents or specifications
  - Details of analyses

Chapter 4: Containment

- Description of the containment design

    - General considerations for containment evaluations
    - Design features
    - Codes and standards
    - Special requirements for plutonium
    - Special requirements for spent fuel

- Containment under normal conditions of transport
    - Containment design criteria
    - Demonstration of compliance with containment design criteria

- Containment under hypothetical accident conditions
    - Containment design criteria
    - Demonstration of compliance with containment design criteria

- Leakage rate tests for type B packages
- Appendices

Chapter 5: Shielding

- Description of shielding design
    - Design features
    - Codes and standards
    - Summary table of maximum radiation levels

- Radiation source
    - Gamma source
    - Neutron source

- Shielding model
    - Configuration of source and shielding
    - Material properties

- Shielding evaluation
    - Methods
    - Input and output data
    - Flux-to-dose-rate conversion
    - External radiation levels

- Appendices

Chapter 6: Criticality

- Description of criticality design
    - Design features
    - Codes and standards
    - Summary table of criticality evaluation

- Fissile material and other contents
- General considerations for criticality evaluations
    - Model configuration
    - Material properties

- Demonstration of maximum reactivity
- Computer codes and cross-section libraries

- Single package evaluation
  - Configuration
  - Results

- Evaluation of undamaged-package arrays (normal conditions of transport)
  - Configuration
  - Results

- Evaluation of damaged-package arrays (hypothetical accident conditions)
  - Configuration
  - Results

- Criticality safety index for nuclear criticality control
- Benchmark evaluations
  - Applicability of benchmark experiments
  - Bias determination

- Appendices

Chapter 7: Package Operations

- Package loading
  - Preparation for loading
  - Loading of contents
  - Preparation for transport

- Package unloading
  - Receipt of package from carrier
  - Removal of contents

- Preparation of empty package for transport
- Other operations
- Appendices

Chapter 8: Acceptance Tests and Maintenance Program

- Acceptance tests
  - Visual inspections and measurement
  - Weld examinations.
  - Structural and pressure tests
  - Leakage tests
  - Component and material tests
  - Shielding tests
  - Thermal tests
  - Miscellaneous tests

- Maintenance program
  - Structural and pressure tests

    - Leakage tests
    - Component and material tests
    - Thermal tests
    - Miscellaneous tests

- Appendices

Chapter 9: Quality Assurance

- Description of QA program
    - Scope
    - Program documentation and approval
    - Summary of eighteen quality criteria
    - Cross-referencing matrix

- Package-specific QA requirements
    - Graded approach for SSCs important to safety
    - Package-specific quality criteria and package activities

- Appendices

I. Describe considerations for evaluating storage of DOE/DOT/NRC certified shipping containers and non-certified shipping containers. Potential references include 10 CFR 71, 49 CFR 173, and TID-7016, Chapter IV.

The following is taken from TID-7016, chapter IV, and is intended only as a summary. This document should be consulted for complete information.

In addressing the NCS of fissile material storage, consideration must be given to the purpose of the storage area. It may be a service area providing temporary storage for materials in process, it may be an area for transient materials in transport, or it may be an area for long-term storage. Each use presents different problems. The number of units, their mass and other properties, the necessary accessibility, and the desired margin of subcriticality help to determine the spacing of material.

ANS.N-16.5-1975, *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials*, presents mass limits for spherical units of fissile materials assembled in cubic arrays reflected by thick water. The tabulated arrays have a neutron multiplication factor not exceeding 0.95. While it does not answer all questions, this standard is directly applicable to many storage problems. The materials to which the standard is applicable are plutonium, U-233, and uranium containing more than 30 weight percent of U-233, as metals and as wet and dry oxides.

The limits are also conservatively applicable to units not spherical in shape. Each unit is considered centered in its cell, and some guidance is provided for relaxing this requirement as well as for modifying the cell shape.

The specifications for cubic arrays are applicable to arrays of any shape because of the increased neutron leakage from non-cubic arrays. The introduction of hydrogenous materials into the space between units is not provided for in the standard; if such

moderation is present, the effect must be evaluated by a validated computational technique. The effect on array reactivity due to the introduction of water, as for example from fire protection systems, is strongly dependent on the form of the fissile material and on the mass and spacing of the units. There is, however, an adequate margin in the limits to accommodate incidental moderation such as would result from enclosing the units in plastic bags that introduce no more than 10 g of polyethylene per kilogram of fissile material.

Factors for reducing the mass limits are provided for concrete-reflected arrays. The limits are reduced to 75 percent of their tabulated values if the concrete thickness is between 120 and 200 mm and to 60 percent for greater thicknesses. Criteria are presented for pairs of arrays in concrete enclosures (slight neutron coupling of arrays separated by 500 mm-thick concrete has been observed experimentally).

Each unit of an array must remain subcritical if immersed in water. The possibility of double batching of the units in a storage cell should be considered when establishing SLs and operating procedures. Administrative controls, limited capacity containers, and storage cell design may be useful for the prevention of double batching.

Consideration should be given to other normal and credible abnormal storage conditions that may affect array subcriticality. Typical examples of changes in operating conditions that should be considered are

- flooding, spraying, or otherwise supplying units or groups of units with water, oil, snow (i.e., low density water), cardboard, wood, or other moderating materials;
- the introduction of additional units or reflectors;
- improper placement of units;
- loss of moderator and neutron absorber between units;
- collapse of a framework used to space units;
- a change in the density of fissile material during storage; and
- the substitution of units containing more fissile material than permitted in operations as a result of operational error or improper labeling.

Alternate storage criteria are also provided by the standard but will not be summarized here.

13. Criticality safety personnel must demonstrate a working level knowledge of the requirements in DOE Technical Standard (STD) DOE-STD-3007-2007, *Guidelines for Preparing Criticality Safety Evaluations at Department of Energy Non-Reactor Nuclear Facilities*.

a. Describe the documentation requirements for a criticality safety evaluation conforming to this standard.

The information for KSAs a and b is taken from DOE-STD-3007-2007.

The determination of limits and controls for the safe handling, processing, and storage of fissionable materials is an essential element of a DOE nuclear CSP. The CSE is the documentation of the thought process and the activities to determine these controls. This

standard provides a framework for generating CSEs that are compliant with the ANSI/ANS-8 series of criticality safety standards.

Section II, *Content Guidance for Criticality Safety Evaluations*, contains guidance for the format and content of CSEs intended for establishing the subcriticality of fissionable material handling, processing, and storage operations. The purpose in performing the type of CSE described in this section is to analyze the criticality hazard associated with a fissionable material process or system and develop limits and controls to prevent a criticality accident. Content guidance for the CSE is provided below, arranged by specific topics and section headings. If any heading or content discussed below is optional, it will be so indicated. The addition of sections and content not discussed herein that are deemed important by the analyst may be included in the CSE.

Section A. Introduction

The purpose and scope of the evaluation should be stated in this section. Relevant background information should also be presented here. If the evaluation represents a modification or revision to an existing evaluation or system then the reason for the change should be clearly stated.

Section B. Description

The system or process is described in this section of the CSE. Illustrations and/or graphics may be provided as needed to provide clarity. Assumptions about the process and scope limitations that impact the CSE should be stated and justified. Assumptions that apply only to computer modeling should not be included here but should be presented in section D below. If the evaluation covers a specific portion of a system or process or is limited to a particular aspect of a system or process, the potential for interaction with other aspects or systems should be described as well as references to any related CSEs.

References, including drawings and operating procedures, may be provided to allow a reviewer the opportunity to further research the system being evaluated and to verify the accuracy of the descriptive information provided. References should be specific enough to identify the cited data.

The description should contain sufficient detail, clarity, and lack of ambiguity to allow a peer reviewer either to independently evaluate the system/process or to independently assess the adequacy and accuracy of the existing evaluation. Drawings and/or sketches should be provided as needed to provide clarity.

Section C. Unique or Special Requirements

This section and its content are optional and may be included to indicate any unique requirements not normally associated with DOE CSEs. If any specific technical guidance or requirement is especially pertinent to the process or CSE it may be cited here for emphasis. There is no need to document rules, DOE Orders, or ANSI/ANS standards that are routinely applicable.

Section D. Methodology and Validation

This section of the CSE describes the methodology or methodologies used to establish limits for the operation being evaluated. Four methods that may be used for the establishment of subcritical limits are

1. reference to national consensus standards that present critical and/or subcritical limits;
2. reference to accepted handbooks of critical and/or subcritical limits;
3. reference to experiments with appropriate adjustments to ensure subcriticality when the uncertainties of parameters reported in the experiment documentation are considered; and/or
4. use of validated calculational techniques. (Note: One specific example based on validated calculations that may be used is the CI technique useful for setting limits on commingled arrays of fissile material containers.)

Note that standards and handbooks provide critical data and subcritical limits, which may not include applied safety margins. The analyst shall develop and document margins to be applied to the limits for the operation being evaluated when using values from referenced standards and handbooks to protect against uncertainties in process variables and against a limit being accidentally exceeded. When limits are based on reference documents such as ANSI/ANS-8 standards, criticality safety handbooks, or published experimental results, complete and specific references shall be cited. The applicability of the reference data to the operation being evaluated should be discussed.

Section E. Process Analysis

This section and its contents shall be included in the CSE. All normal conditions and credible abnormal conditions (credible contingencies) shall be analyzed and documented. This section shall document that operations are subcritical under all normal conditions and that no credible abnormal condition can lead to an accidental criticality. This section shall identify those controls that have been developed. As part of this effort, the CSE shall document that at least two unlikely, independent, and concurrent changes in process conditions (i.e., changes in process parameters) must occur before a criticality accident is possible.

The intent of ANSI/ANS-8.1-1998 regarding application of the double-contingency principle is that two independent process parameters should be controlled. It is not always possible to control two independent parameters for every process, thus ANSI/ANS-8.1-1998 does not make the double-contingency principle a requirement. DOE O 420.1B, *Facility Safety*, also allows DOE to approve cases where the double-contingency principle cannot be met. Therefore, in the case where a criticality accident is credible and only one parameter is controlled, the process does not meet the double-contingency principle.

A CSE done for the purpose of demonstrating that a mitigated criticality accident is not a credible event cannot simply assume that application of double contingency achieves that result. If a criticality accident is not credible then the risk of a criticality accident is lower than that provided by the application of the double-contingency principle even if only one parameter is controlled. Therefore, in cases where a mitigated criticality accident is not credible, DOE O 420.1B does not require DOE approval. A CSE showing that a mitigated criticality accident is not credible should not rely on simplistic formulas for the numbers

of controls or contingencies in place. The CSE should provide justification for concluding that a criticality accident is not credible based on the application of technical practices described in section 4.2 of ANSI/ANS-8.1-1998. Such a CSE may rely on controls present in the facility. Controls that are relied upon shall be documented. Reliance should be placed explicitly on engineered features, SACs, and/or various administrative programs such as material control and accountability (MC&A), safeguards and security (S&S), on and offsite transportation requirements, and non-destructive assay to support the conclusion that a criticality accident is not a credible event. Finally, DSA and TSR level controls should be developed to ensure that the potential for a criticality accident remains not-credible. (See guidance provided in section IV of DOE-STD-3007-2007 for the selection of controls for inclusion in the DSA.)

Section F. Evaluation and Results

The use of this section heading is optional. However, the content described below shall be incorporated by reference or included somewhere in the CSE.

If calculational techniques are used, detailed descriptions of the models shall be presented. The level of detail shall be sufficient to allow an independent reviewer to reconstruct the computational model, compare the model with the descriptive information in subsection B, and determine if the overall model is accurate and appropriate. Significant assumptions and simplifications shall be stated. Pertinent calculational parameters important to the understanding of the analysis shall be specified or incorporated by reference.

All pertinent calculational results shall be reported. Where referenced calculations or reports are used to support the results of the evaluation, a summary of the referenced calculations should be included. Plots of data should be clearly labeled. Descriptions/labels of individual computer runs should indicate the physical attributes of the system being analyzed. Estimated uncertainties in the results (e.g., statistical uncertainties associated with Monte Carlo calculations) and any analyzed sensitivities to modeling simplifications should be included here as well.

Section G. Credited Controls and Assumptions

All engineered features (active and passive) and ACs identified during the performance of the process analysis shall[3] be stated in this section. The purpose is to clearly document and highlight what is being relied upon to prevent a criticality accident. These may or may not be TSRs (see section IV of DOE-STD-3007-2007).

This section should also list important assumptions relied upon by the analysis. Assumptions listed here should be closely associated with implemented controls and discussed in the process analysis section. These include but are not limited to the administrative systems relied upon by the analyst.

Section H. Summary and Conclusions

Inclusion of this section is optional. The overall criticality safety assessment of the system being analyzed may be summarized in this section. The range of applicability and special limitations in the evaluation may be documented here.

If unique requirements must be satisfied (those discussed in subsection C), a statement of compliance with these requirements may be included here.

When applicable, reference to normal and abnormal ranges of operational parameters may also be made. Portions of the evaluation that have been deferred to other documents may be summarized here.

Section I. References

To the extent practical, a CSE should stand on its own. However, references may be used so all external technical information and relevant descriptive information can be verified. Verbal communications, as references, should be avoided, and crucial conclusions of the evaluation should not depend on verbal communications. Where private communications such as emails or verbal discussions provide significant information related to the evaluation, a copy of the email or a "note to file" documenting the conversation should be included as a reference or as an attachment. References should be documented with sufficient detail to describe applicability to the process being evaluated. In any event, the evaluation shall not be so dependent on references as to prevent an independent reviewer from being able to judge the adequacy of the evaluation as a stand-alone document.

b. Discuss the role of this standard in establishing appropriate analytical techniques for criticality safety evaluations.

DOE-STD-3007-2007 provides a framework for generating CSEs supporting fissionable material operations at DOE nonreactor nuclear facilities. This standard imposes no new criticality safety analysis requirements.

Revision to the standard was undertaken for two primary reasons. First, it provides DOE expectations for the technical content and analysis practices for CSEs to ensure compliance with the required ANSI/ANS-8 series of criticality safety standards. Second, it provides the linkage between the CSE process and the DSA process required by 10 CFR 830, "Nuclear Safety Management," subpart B. Adherence to this standard ensures that requirements for CSEs in the ANSI/ANS-8 standards and DOE O 420.1B are met.

The same language is used as the ANSI/ANS-8 standards for differentiating between required (shall), recommended (should), and acceptable practices (may). To conform to this standard, criticality evaluation practices shall be performed in accordance with its requirements but not necessarily with its recommendations.

c. Discuss the relationship between DOE-STD-3007-2007 and DOE O 420.1B, *Facility Safety*.

The following is taken from DOE O 420.1B.

DOE O 420.1B states that the methodology for conducting CSEs must be approved by DOE unless the evaluations are conducted in accordance with the DOE-STD-3007-2007 and evaluated in accordance with DOE-STD-1134-1999, *Review Guide for Criticality Safety Evaluation*, or successor document.

d. Discuss the proper relationship between criticality analyses and controls, the Documented Safety Analysis (DSA), and TSR Controls, and design features documented in the DSA and TSRs. Example references include DOE-STD-3007-2007, 10 CFR 830, ANSI/ANS 8.26, § 7.7, DOE G 421.1-2, DOE G 423.1-1, and DOE-STD-3009-94 CN 3.

The following is taken from DOE-STD-3009-94.

Chapter 6 of a DSA prepared according to DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*, contains a summary description of the CSP, including a description of how CSEs are conducted.

The following is taken from DOE-STD-3007-2007.

Section IV of DOE-STD-3007-2007 provides specific guidance on the linkage between CSEs and the DSA. DOE's nuclear safety management rule, 10 CFR 830 subpart B, requires the preparation of safety bases, including DSAs and TSRs, for hazard category 1, 2, and 3 nuclear facilities. Chapter 3 of a DSA includes identification and analyses of hazards and hazard controls to provide adequate protection to the public, workers, and the environment. CSEs provide detailed information that should be reviewed when constructing DSA hazard analysis scenarios. However, while such scenario entries in the hazard analysis may summarize CSE information, they are expected to be a "stand-alone" product that does not incorporate CSEs into the DSA. The events identified in the hazard analysis should be those covered in CSE. If the CSE shows that the fissionable material process/system remains subcritical under normal and credible abnormal conditions and documents that the fissionable material process/system complies with the double-contingency principle, then there is no need to perform a separate analysis in the hazard analysis, only reference the CSE.

All controls necessary to prevent and/or mitigate criticality accidents shall be considered for inclusion in the facility DSA and TSR. The CSP shall have a mechanism to review all changes or potential changes to NCS controls for capture by the configuration control program as well as revisions and updates to the DSA and TSRs.

A process to examine the collection of controls developed in the CSEs to determine their importance in DSA space should be developed and documented as part of the CSP. The process should be agreed on by the NCS and safety analysis staff and formally documented as part of the CSP description document. Not every CSE or fissionable material process is required to have a control selected for inclusion in the DSA.

Eight evaluation criteria that should be used are listed below:

1. The selection of NCS controls for the DSA should be performed using a team of criticality safety, nuclear safety, and operations personnel.
2. The consequence of criticality should be examined for the purpose of establishing whether a particular control is safety class or safety significant, equipment important to safety, or merely provides defense-in-depth.

3. The CSEs that cover the fissionable material operations addressed by the DSA should be examined to ensure that bounding assumptions or analysis conditions are considered as potential DSA/TSR controls.
4. All passive engineered features credited in the CSE should be considered for selection as a DSA design feature.
5. All active engineered features credited in the CSE should be considered for selection as a safety class or safety significant SSC with associated SLs or limiting condition of operations as appropriate.
6. Not all engineered features need be selected for inclusion in the DSA/TSR. The minimum set of controls selected for inclusion should be those that meet the following conditions: (A) loss of the single control under consideration could directly result in a criticality accident, (B) loss of the control could result in a singly contingent condition, and (C) active controls requiring calibration. Additional engineered features beyond the minimum set may be selected as appropriate.
7. If all of the credible scenarios are shown to be subcritical by engineered features, then SACs are not required; and/or
8. Only ACs meeting at least one of the following criteria should be considered for inclusion: (A) credible violation of the control could directly lead to a criticality accident, (B) if the safety function were to be equivalent to a safety class or safety significant engineered control and (C) general references to control philosophy.

A linking document called a criticality control review (CCR) should be used to summarize the results of this evaluation process for criticality hazards and controls. The process leading to the development of the CCR should be specified as part of the CSP description document. The CSP should also document how revisions and updates to the CCR and CSEs are treated relative to the unreviewed safety question (USQ) process. The advantages to development of a CCR are

- the CCR now becomes the technical reference to the DSA for criticality safety; and
- the CCR allows summarizing several CSEs for a process or a facility in a single document so that identification of commonly important attributes and controls is possible.

14. Criticality safety personnel must demonstrate a working level knowledge of the guidance provided in DOE-STD-1134-99, *Review Guide for Criticality Safety Evaluations*.

a. Describe the purpose and general structure of the guide.

The following is taken from DOE-STD-1134-99.

The purpose of this standard is to provide a set of guidelines, not checklists, for use by DOE criticality safety personnel when reviewing CSEs produced by the contractor. It provides a consistent framework for ensuring that acceptable evaluations are produced by DOE contractors. DOE-STD-1134-99 contains guidelines that should be followed when reviewing CSEs that were developed by DOE contractors to demonstrate the safety of fissile material handling at DOE non-reactor nuclear facilities. Adherence to these guidelines will enhance the consistency and uniformity of reviews of CSEs across the DOE complex as well as compliance with DOE O 420.1B requirements. The guidance

provided is keyed to sections of DOE-STD-3007-93, but is useable with DOE-STD 3007-2007. These sections are:

- 1.0 Introduction
- 2.0 Description
- 3.0 Requirements
- 4.0 Methodology
- 5.0 Discussion of Contingencies
- 6.0 Evaluation & Results
- 7.0 Design Features (Passive & Active) and Administratively Controlled Limits & Requirements
- 8.0 Summary and Conclusions
- 9.0 References
- 10.0 Appendices

DOE-STD-1134-99 provides key review issues for each of these sections except section 1.0 Introduction.

b. Using the guide as a reference, discuss the guidelines provided for use by DOE criticality safety personnel when reviewing criticality safety evaluations produced by a contractor.

This is a performance-based competency. The Qualifying Official will evaluate its completion.

15. Criticality safety personnel must demonstrate a working level knowledge of previous criticality accidents and their causal factors.

a. Discuss common precursors to criticality accidents.

The following is taken from LA-13638.

The human element is nearly always present, and is usually the dominant cause of a criticality accident. Common failures include the following:

- Failure to use written procedures
- Failure to follow existing procedures
- Failure to understand the basis for ACs
- Incomplete understanding of abnormal conditions
- Failure to monitor holdup and accumulated errors
- Failure to implement a stop-work policy

There is also an element of supervisory, management, and regulatory failure related to criticality accidents, such as the following:

- Failure to have an effective self-reporting policy, and punishing the messengers
- Failure to train on bases for procedures
- User-unfriendly equipment
- Failure to recognize the existence of the hazard and its consequences
- Production, holiday, or schedule priorities

b. Discuss with some detail three (3) historic accidents described in LA-13638 and the lessons learned from each. Reference LA-13638, McLaughlin, Monahan, & Pruvost, *A Review of Criticality Accidents*, May 2000.

LA-13638 is available at http://www.orau.org/ptp/Library/accidents/la-13638.pdf

LA-13638 gives a description of the twenty-two process and thirty-eight experimental criticality accidents which are known. Accident conditions are discussed. Of the twenty-two process accidents, twenty-one occurred with solution and one with metal. None of the accidents resulted from an incorrect criticality calculation.

The experimental accidents show either experimenter error, or failure to understand the gross engineering dynamics of the systems (e.g., something moved that shouldn't, a two-phase flow occurred, something didn't move that should have or moved too slowly, motion did not reverse as anticipated).

Refer to LA-13638 to select three accidents.

c. Discuss generic lessons learned from criticality accident history, as described in LA-13638, Section 1C.

LA-13638 divides the lessons learned into two categories, lessons of operational importance and lessons of supervisory, managerial, and regulatory importance. The lessons learned within each of those categories are listed below. LA-13638 may be consulted for detailed explanations of each lesson learned.

Lessons of Operational Importance

- Unfavorable geometry vessels should be avoided in areas where high concentration solutions might be present.
- Important instructions, information, and procedural changes should always be in writing.
- The processes should be familiar and well understood so that abnormal conditions can be recognized.
- Criticality control should be part of an integrated program that includes fissile material accountability.
- Operations personnel should know how to respond to foreseeable equipment malfunctions or their own errors.
- Operations personnel should be trained in the importance of not taking unapproved actions after an initial evacuation.
- Readouts of radiation levels in areas where accidents may occur should be considered.
- Operations involving both organic and aqueous solutions require extra diligence in understanding possible upset conditions if mixing of the phases is credible.
- Operations personnel should be made aware of criticality hazards and be empowered to implement a stop-work policy.
- Operating personnel should be trained to understand the basis for and to adhere to the requirement for always following procedures.

- Hardware that is important to criticality control but whose failure or malfunction would not necessarily be apparent to operations personnel, should be used with caution.
- Criticality alarms and adherence to emergency procedures have saved lives and reduced exposures.

Lessons of Supervisory, Managerial, and Regulatory Importance

- Process supervisors should ensure that the operators under their supervision are knowledgeable and capable.
- Equipment should be designed and configured with ease of operation as a key goal.
- Policies and regulations should encourage self-reporting of process upsets and err on the side of learning more, not punishing more.
- Senior management should be aware of the hazard of accidental criticality and its consequences.
- Regulations should exist which promote safe and efficient operations.
- Regulators, like process supervisors, should ensure that those they regulate are knowledgeable and capable.

16. Criticality safety personnel must demonstrate a familiarity level knowledge of problem analysis principles and the techniques necessary to identify Department problems, potential causes, and corrective action(s) associated with criticality safety issues.

a. Describe and explain the application of problem analysis techniques, including the following:
- **Root cause analysis**
- **Causal factor analysis**
- **Change analysis**
- **Barrier analysis**
- **Management oversight risk tree analysis**

The following is taken from DOE Workbook: *Conducting Accident Investigations.*

Root Cause Analysis

Accidents are symptoms of larger problems within a safety management system. Although accidents generally stem from multiple causal factors, correcting only the local causes of an accident is analogous to treating only symptoms and ignoring the "disease." To identify and treat the true ailments in a system, the root causes of an accident must be identified. Root cause analysis is any technique that identifies the underlying deficiencies in a safety management system that, if corrected, would prevent the same and similar accidents from occurring.

Root cause analysis is a systematic process that uses the facts and results of the core analytic techniques to determine the most important reasons for the accident. Root cause analysis is not an exact science and therefore requires a certain amount of judgment. The intent of the analysis is to identify and address only those root causes that can be controlled within the system being investigated, excluding events or conditions that cannot be reasonably anticipated and controlled, such as some natural disasters. The core analytic

techniques—events, and causal factors analysis, barrier analysis, and change analysis—provide answers to an investigator's questions regarding what, when, where, who, and how. Root cause analysis is performed to resolve the question, "Why?"

Once several (or all) of the recommended core analytic techniques have been performed, the accident investigation board should have a broad understanding of the accident's events and conditions, along with a fairly extensive list of suspected causal factors. A root cause analysis is performed to refine the list of causal factors and categorize each according to its significance and impact on the accident.

Refining causal factors entails identifying any commonality or linkages that suggest more fundamental causal factors. The core functions and guiding principles of integrated safety management (ISM) provide a useful framework for grouping causal factors and identifying the underlying safety management deficiencies that caused the accident. For example, causal factors in an accident might include "failure to follow procedures," "failure to establish a fire watch," and "failure to stop work when unanticipated conditions arose." By reviewing the five core functions of ISM, it becomes clear that each of these causal factors reflects an underlying failure to perform work within controls, which is core function #4. Other causal factors in an accident may demonstrate similar relationships with the other core functions and guiding principles. The underlying management system deficiency, as defined by the related causal factors, is a candidate root cause.

There may be more than one root cause of a particular accident, but probably not more than three or four. If more are thought to exist at the conclusion of the analysis, the board should re-examine the list of causal factors to determine which causes can be further combined to reflect more fundamental (root) causes. Section 7 of the DOE Workbook provides some examples of root cause analysis and discusses analytical tools that can help accident investigators determine the root causes of an accident.

To initiate a root cause analysis, the facts surrounding the accident must be known. In addition, the facts must be analyzed using other analytic methods to ascertain an initial list of causal factors. A rather exhaustive list of causal factors must be developed prior to the application of root cause analysis to ensure that final root causes are accurate and comprehensive.

The board should examine the evidence collected from the accident scene, witness statements, interviews, and facility documents. It should then determine whether additional information will be needed for the particular root cause technique they are performing. It is important that the accident investigation board work together to determine the root causes of an accident. One of the board's primary responsibilities is to identify an accident's causal factors so that judgments of need can be prepared and appropriate corrective measures can be developed and implemented. Therefore, all board members must participate in the root cause analysis; it cannot be left solely to a single member of the board.

Root cause analysis can be performed using computerized or manual techniques. Regardless of the method, the intent is to use a systematic process for identifying root causes. Manual root cause analysis methods include tier diagramming and compliance/noncompliance. Each is effective as a systematic method for identifying root

causes. However, the compliance/noncompliance method reflects the limited applicability of certain techniques and underscores the need for the board to select analytic methods commensurate with the accident's scope, complexity, and severity. Computerized techniques can be somewhat more sophisticated and generally speed the process of root cause identification. It is important to note, however, that computerized techniques are dependent on the quality and quantity of data input. Moreover, at least one member of the board should be very familiar with the software package, including its limitations.

Causal Factor Analysis

The following describes the process for using the events and causal factors chart to determine the causal factors of an accident. This process is an important first step in later determining the root causes of an accident. The results of this analysis can be used with a tier diagram if desired. The quality and accuracy of root cause analysis depends on the results of the events and causal factors analysis. Therefore, the events and causal factors analysis must be complete and thorough.

Events and causal factors analysis requires deductive reasoning to determine which events and/or conditions contributed to the accident.

Before starting to analyze the events and conditions noted on the chart, the board must first ensure that the chart contains adequate detail. Both change and barrier analyses should be conducted and the results incorporated into the chart before the analysis begins. Also, the board must resolve any obvious gaps in data before this analysis begins.

By the time the board is ready to conduct a preliminary analysis of the chart, a great deal of time will have been devoted to adding, removing, and rearranging events and conditions on the chart. In all likelihood, the chart will be lengthy, possibly containing 100 events or more. Given the magnitude of data, one can become overwhelmed with where to begin identifying causal factors. It is easiest and most efficient to begin with the event on the chart that immediately precedes the accident and work backwards.

Examine the first event that immediately precedes the accident. Evaluate its significance in the accident sequence by asking, "If this event had not occurred, would the accident have occurred?" If the answer is, "The accident would have occurred whether this event happened or not" (e.g., worker punched in to work at 0700), then the event is not significant. Proceed to the next event in the chart, working backwards from the accident.

If the answer to the evaluation question is, "The accident would not have occurred without this event," then determine whether the event represented normal activities with the expected consequences. If the event was intended and had the expected outcomes, then it is not significant. However, if the event deviated from what was intended or had unwanted consequences, then it is a significant event.

Carefully examine the events and conditions associated with the significant event by asking a series of questions about this event chain, such as:

- Why did this event happen?
- What events and conditions led to the occurrence of the event?
- What went wrong that allowed the event to occur?

- Why did these conditions exist?
- How did these conditions originate?
- Who had responsibility for the conditions?
- Are there any relationships between what went wrong in this event chain and other events or conditions in the accident sequence?
- Is the significant event linked to other events or conditions that may indicate a more general or larger deficiency?

The significant events, and the events and conditions that allowed the significant events to occur, are the accident''causal factors.

Repeat this questioning process for every event in the chart. As a causal factor is identified, write a summary statement that describes the causal factor on an adhesive note of a unique color and place the note above the event chain from which it was derived when constructing the chart manually. If a computer graphics program is used to construct the chart, use a hexagon to represent causal factors.

Sometimes events and conditions from several different event chains are related and suggest a larger, more significant causal factor. For example, in two side-by-side event chains, the conditions "procedure did not address electrical hazard" and "electrical hazard not discussed in pre-job brief" may indicate that the electrical hazard was not identified in the hazard analysis for the activity. In such a case, the board can write a causal factor concerning the hazard analysis, place it on the chart, and connect it with an arrow to the two event chains from which it was derived. Alternatively, the board can record the same causal factor twice and place it above each of the applicable event chains.

After these steps have been completed for each event on the chart, the process should be repeated with all board members to ensure that nothing has been overlooked and that consensus has been reached. When the board is satisfied that all causal factors have been identified on the chart, efforts can then be focused on initiating the root cause analysis.

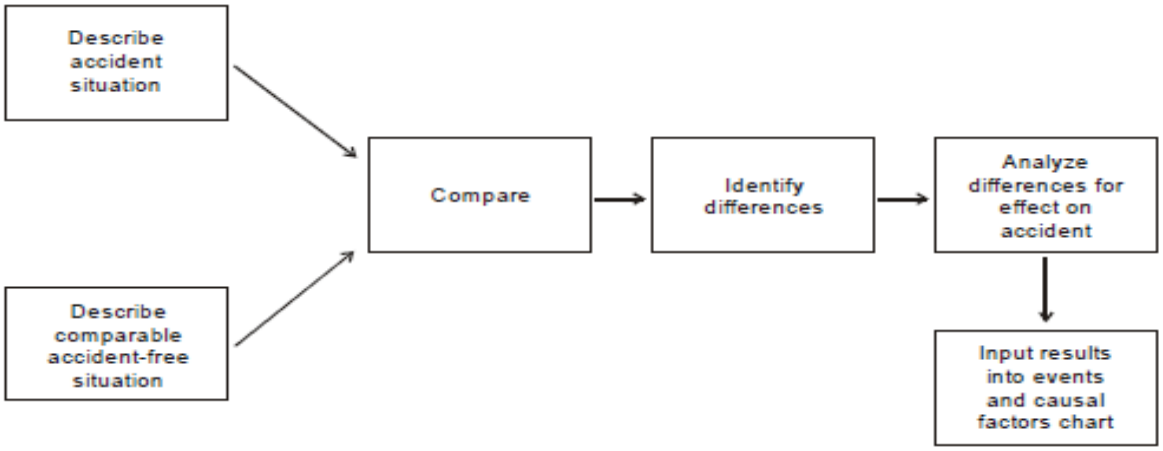
Change Analysis

Change is anything that disturbs the "balance" of a system operating as planned. Change is often the source of deviations in system operations. Change can be planned, anticipated, and desired, or it can be unintentional and unwanted. Workplace change can cause accidents, although change is an integral and necessary part of daily business. For example, changes to standards or directives may require facility policies and procedures to change, or turnover/retirement of an aging workforce will change the workers who perform certain tasks. Change can be desirable, for example, to improve equipment reliability or to enhance the efficiency and safety of operations. Uncontrolled or inadequately analyzed change can have unintended consequences, however, and result in errors or accidents.

Change analysis examines planned or unplanned changes that caused undesired outcomes. In an accident investigation, this technique is used to examine an accident by analyzing the difference between what has occurred before or was expected and the actual sequence of events. The investigator performing the change analysis identifies specific differences between the accident-free situation and the accident scenario. These differences are

evaluated to determine whether the differences caused or contributed to the accident. For example, why would a system that operates correctly 99 times out of 100 fail to operate as expected one time?

Change analysis is relatively simple to use. As illustrated in figure 17, it consists of six steps. The last step, in which investigators combine the results of the change analysis with the results from other techniques, is critical to developing a comprehensive understanding of the accident.



Source: DOE Workbook, Conducting Accident Investigations

Figure 17. The change analysis process is relatively simple

When conducting a change analysis, investigators identify changes as well as the results of those changes. The distinction is important, because identifying only the results of change may not prompt investigators to identify all causal factors of an accident.

The results of a change analysis can stand alone, but are most useful when they are combined with results from other techniques. For example, entering change analysis results into the events and causal factors chart helps to identify potential causal factors.

To conduct a change analysis, the analyst needs to have a baseline situation. This baseline situation can be

- the same situation but before the accident (e.g., previous shift, last week, or last month);
- a model or ideal situation (i.e., as designed or engineered).

Generally, it is recommended that boards compare the accident sequence to the same situation in an accident-free state—the operation prior to the accident—to determine differences and thereby identify accident causal factors. In order for the comparison to be effective, investigators must have sufficient information regarding this baseline situation.

The following data sources can be a starting point for acquiring a good working knowledge of the system, facility, or process under study prior to the accident or event; however, the list of input requirements should be tailored to fit the specific circumstances and needs of the investigation:

- Blueprints
- Equipment description documents
- Drawings
- Schematics
- Operating and maintenance procedures
- Roles and responsibilities
- Job/task descriptions
- Personnel qualifications
- Results of hazard analysis
- Performance indicators
- Personnel turnover statistics.

To develop the information needed to conduct a change analysis, it is useful for the board to list any changes they identify from their information-gathering activities on a poster board set up in the board's common meeting room. At the beginning of the investigation, the board members should simply note the changes they identify as they find them and not worry about analyzing the significance of the changes. Often, in the early stages of an investigation, there is insufficient information to determine whether a change is important or not.

As the investigation progresses, it will become clear that some of the changes noted on the poster board are insignificant and can be crossed off the list. The remaining changes that seem to be important for understanding the accident can then be organized by entering them into the change analysis worksheet.

Board members should first categorize the changes according to the questions shown in the left-hand column of the worksheet. That is, the board should determine if the change pertained to, for example, a difference in

- what events, conditions, activities, or equipment were present in the accident situation that were not present in the baseline (accident-free, prior, or ideal) situation (or vice versa);
- when an event or condition occurred or was detected in the accident situation versus the baseline situation;
- where an event or condition occurred in the accident situation versus where an event or condition occurred in the baseline situation;
- who was involved in planning, reviewing, authorizing, performing, and supervising the work activity in the accident versus the accident-free situation;
- how the work was managed and controlled in the accident versus the accident-free situation.

Reviewing the worksheet may also prompt the investigators to identify additional changes that were not originally listed.

To complete the remainder of the worksheet, first describe each event or condition of interest in the column labeled, "Accident Situation." Then describe the related event or condition that occurred (or should have occurred) in the baseline situation in the column labeled, "Prior, Ideal, or Accident-Free Situation." The difference between the events and conditions in the accident and the baseline situations should be briefly described in the

column labeled, “Difference.” As a group, the board should then discuss the effect that each change had on the accident and record the evaluation in the final column of the worksheet. The worksheet allows the user to compare the “accident situation” with the “accident-free situation” and evaluate the differences to determine each item's effect on the accident.

A change analysis summary is generally included in the accident investigation report. It contains a subset of the information listed in the change analysis worksheet. The differences or changes identified can generally be described as causal factors and should be noted on the events and causal factors chart and used in the root cause analysis, as appropriate.

Barrier Analysis

Barrier analysis is based on the premise that hazards are associated with all accidents. Barriers are developed and integrated into a system or work process to protect personnel and equipment from hazards. For an accident to occur, there must be

- a hazard, which comes into contact with
- a target, because
- barriers or controls were not in place, unused or failed.

A hazard is the potential for an energy flow to result in an accident or other adverse consequence. Energy flow is the transfer of energy from its source to another destination. This transfer of energy can be either wanted or unwanted. For example, the flow of electricity through an electrical cable to a piece of equipment is a desired energy flow. A worker coming into contact with that electricity is an undesired energy transfer. As used here, energy is defined broadly as the capacity to do work. Energy could be, for example, kinetic, biological, acoustical, chemical, electrical, mechanical, potential, electromagnetic, thermal, or radiation.

A target is a person or object that a hazard may damage, injure, or fatally harm.

A barrier is any means used to control, prevent, or impede the hazard from reaching the target.

Investigators use barrier analysis to identify hazards associated with an accident and the barriers that should have been in place to prevent it. This analysis addresses

- barriers that were in place and how they performed;
- barriers that were in place but not used;
- barriers that were not in place but were required;
- the barrier(s) that, if present or strengthened, would prevent the same or a similar accident from occurring in the future.

When analyzing barriers, investigators should first consider how the hazard and target could come together and what was in place or was required to keep them apart. Obvious physical barriers are those placed directly on the hazard (e.g., a guard on a grinding wheel); those placed between a hazard and target (e.g., a railing on a second-story platform); or those located on the target (e.g., a welding helmet). Management system barriers may be less obvious, such as the exposure limits required to minimize harm to

personnel or the role of supervision in ensuring that work is performed safely. The investigator must understand each barrier's intended function and location, and how it failed to prevent the accident.

To analyze the performance of physical barriers, investigators may need several different types of data, including:

- Plans and specifications for the equipment or system
- Procurement and vendor technical documentation
- Installation and testing records
- Photographs or drawings
- Maintenance histories

To analyze management barriers, investigators may need to obtain information about barriers at three organizational levels responsible for the work: the activity, facility, and institutional levels. For example, at the activity level, the investigator will need information about the work planning and control processes that governed the work activity, as well as the relevant safety management systems. This information could include:

- Organizational charts defining supervisory and contractor management roles and responsibilities for safety
- Training and qualification records for those involved in the accident
- Hazard analysis documentation
- Hazard control plans
- Work permits
- The work package and procedures that were used during the activity

The investigator may also need information about safety management systems at the facility level. This kind of information might include:

- The standards and requirements that applied to the work activity, such as occupational exposure limits or relevant Occupational Safety and Health Administration regulations
- The facility TSR and safety analysis report
- Safety management documentation that defines how work is to be planned and performed safely at the facility
- The status of ISM implementation

The third type of information the investigator may need would be information about the institutional-level safety management direction and oversight provided by senior line management organizations. This kind of information might include:

- Policy, orders, and directives
- Budgeting priorities
- Resource commitments

The investigator should use barrier analysis to ensure that all failed, unused, or uninstalled barriers are identified and that their impact on the accident is understood. However, the investigator must cross-validate the results with the results of other core analytic techniques to identify which barrier failures were contributory or root causes of the accident.

A barrier analysis worksheet is a useful tool in conducting a barrier analysis. A blank worksheet is provided at the end of section 7 of the DOE workbook. Steps used for completing this worksheet are:

- Identify the hazard and the target. Record them at the top of the worksheet.
- Identify each barrier. Record in column one.
- Identify how the barrier performed (What was the barrier's purpose? Was the barrier in place or not in place? Did the barrier fail? Was the barrier used if it was in place?) Record in column two.
- Identify and consider probable causes of the barrier failure. Record in column three.
- Evaluate the consequences of the failure in this accident. Record evaluation in column four.

The results of barrier analysis are first derived and portrayed in tabular form, then summarized graphically to illustrate, in a linear manner, the barriers that were unused or that failed to prevent an accident. Results from this method can also reveal what barriers should have or could have prevented an accident.

In the tabular format, individual barriers and their purposes are defined. Each is considered for its effectiveness in isolating, shielding, and controlling an undesired path of energy.

Management Oversight and Risk Tree Analysis (MORT)

MORT—a comprehensive analytical tree technique—was originally developed for DOE to help conduct nuclear criticality and hardware analysis. It was later adapted for use in accident investigations and risk assessments. Basically, MORT is a graphical checklist, but unlike the events and causal factors chart, which must be filled in by investigators, the MORT chart contains generic questions that investigators attempt to answer using available factual data. This enables the investigator to focus on potential key causal factors. The MORT chart's size can make it difficult to learn and use effectively. For complex accidents involving multiple systems, such as nuclear systems failures, MORT can be a valuable tool but may be inappropriate for relatively simple accidents. MORT requires extensive training to effectively perform an in-depth causal analysis of complex accidents. If needed, the MORT analysis is usually performed by board members with substantial previous experience in using the MORT techniques.

The benefits of MORT are that it

- uses the analytic tree method to systematically dissect an accident;
- serves as a detailed road map by requiring investigators to examine all possible causal factors (e.g., assumed risk, management controls or lack of controls, and operator error);
- looks beyond immediate causes of an accident and instead stresses close scrutiny of management systems that allowed the accident to occur;
- permits the simultaneous evaluation of multiple accident causes through the analytic tree.

In evaluating accidents, MORT provides a systematic method (analytic tree) for planning, organizing, and conducting a comprehensive accident investigation. Through MORT

analysis, investigators identify deficiencies in specific control factors and in management system factors. These factors are evaluated and analyzed to identify the causal factors of the accident.

Detailed knowledge and understanding of management and operating systems is a prerequisite to a comprehensive MORT analysis. Therefore, it is most effective if investigators have collected substantial evidence before initiating the MORT process. The management system data required include procedures, policies, implementation plans (IPs), risk assessment program, and personnel. Information about the facility, operating systems, and equipment is also needed. This information can be obtained through reviews of physical evidence, interview transcripts, management systems, and policies and procedures.

The first step of the process is to obtain the MORT charts and select the MORT chart for the safety program area of interest evaluating each event. Next, the investigators work their way down through the tree, level by level, proceeding from known to unknown. Events should be coded in a specific color relative to the significance of the event (accident). The color-coding system used in MORT analysis is shown in table 7-10 of the DOE workbook. An event that is deficient, or less than adequate in MORT terminology, is marked red. The symbol is circled if suspect or coded in red if confirmed. An event that is satisfactory is marked green in the same manner. Unknowns are marked in blue, being circled initially and colored in if sufficient data do not become available, and an assumption must be made to continue or conclude the analysis.

It is not useful to start on the first day by marking everything as needing more information (color-coded blue). Instead, start marking the first MORT chart with red and black for events where there is sufficient evidence. Ideally, all blue blocks eventually are replaced by one of the other colors; however, this may not always be possible.

When the appropriate segments of the tree have been completed, the path of cause and effect (from lack of control by management, to basic causes, contributory causes, and root causes) can easily be traced back through the tree. This becomes a matter of following the red events through the various logic gates. The tree highlights quite clearly where controls and corrective actions are needed and can be effective in preventing recurrence of the accident.

b. Describe the following types of investigations and discuss an example of the application of each of the following:
- **Type A**
- **Type B**

[Note: DOE O 225.1B, Accident Investigations, dated March 4, 2011, cancels DOE O 225.1A and DOE G 225.1A-1. The most significant change is the revised criteria for categorizing accident investigations. Please refer to DOE O 225.1B for the most current information.]

The following is taken from DOE O 225.1A.

A type A investigation is conducted for the more serious accidents and is appointed and managed by the Office of the Assistant Secretary for Environment, Safety and Health; A type B investigation is appointed and managed at the field level.

The following is taken from DOE G 225.1A-1.

Attachment 2 to DOE O 225.1A, *Accident Investigation Categorization Algorithm*, contains the criteria for determining whether an accident investigation should be categorized as type A or type B. These criteria are summarized in table 6.

The heads of DOE field elements are responsible for promptly reporting and categorizing all accidents to determine whether a type A or type B investigation is required. Not properly categorizing an accident investigation can result in wasted resources (overcategorization) or the failure to prevent similar or more serious accidents because of unresolved or unidentified causes (under-categorization). Therefore, it is important for heads of field elements to make accurate categorizations.

This categorization and subsequent initiation of a type A or type B investigation should be made expeditiously, taking into account that timeliness is crucial to conducting an accurate investigation, preserving the accident scene evidence, and identifying causal factors.

Table 6. Investigation categorization algorithm summary

Categorization Criteria	Type of Investigation	
	Type A	Type B
Human Effects	Any fatal or likely to be fatal ▪ injury (49 CFR 830.2) ▪ chemical exposure ▪ biological exposure Any one accident requiring hospitalization of three or more individuals incurring a serious injury (serious injury is defined in 49 CFR 830.2 as any injury that requires hospitalization for more than 48 hours, commencing within seven days from the date the injury was received; results in severe hemorrhages, nerve, muscle, or tendon damage; involves severe damage to an internal organ; or involves second or third-degree burns affecting more than nine percent of the body surface); or	Any accident that results in the hospitalization of one or more DOE, contractor, or subcontractor employees, or members of the public for five continuous days or longer due to ▪ serious injury ▪ chemical exposure ▪ biological exposure occupational ▪ illness (except members of the public) Any one accident resulting in five or more lost workday cases. A series of accidents involving five or more lost workday cases occurring within a one-year time period that involve identical or similar ▪ facilities ▪ systems ▪ equipment

Categorization Criteria	Type of Investigation	
	Type A	Type B
Human Effects	with a high probability of realizing a permanent total disability due to injuries, chemical exposures, or biological exposures received One individual radiation exposure (see 10 CFR 830.202) of ▪ 25 rem or more total ▪ 75 rem or more to the eye ▪ 250 rem or more to skin or extremity (shallow dose) ▪ 250 rem or more for external exposure (deep dose) or to organ or tissue (committed dose) for other than lens of the eye ▪ 2.5 rem or more dose to embryo or fetus of pregnant woman	▪ materials ▪ procedures A single radiation exposure to an individual that results in ▪ 10 rem but <25 rem total dose ▪ 30 rem but <75 rem dose to the lens of the lens of the eye ▪ 100 rem but <250 rem shallow dose to skin or an extremity ▪ 100 rem but <250 rem sum of deep dose and dose to organ or tissue (other than lens of the eye) ▪ 1 rem but <2.5 rem dose to embryo or fetus of declared pregnant woman
Environmental Effects	Any release greater than five times the reportable limits in 40 CFR 302 of a hazardous substance, material, waste, or radionuclide resulting in serious environmental damage	Any release over two times but less than five times the reportable limits in 40 CFR 302 of a hazardous substance, material, waste, or radionuclide resulting in serious environmental damage
Property Effects	Loss or damage* of > $2.5 million in property, including costs for ▪ cleaning ▪ decontaminating ▪ renovating ▪ renovating or ▪ rehabilitating structures, equipment, or property Apparent loss, explosion, or theft involving radioactive or hazardous material in quantities or circumstances likely to constitute a hazard to health, safety, or property Any unplanned nuclear criticality	Loss or damage* of over $1 million but less than $2.5 million in property, including costs for ▪ cleaning ▪ decontaminating ▪ renovating ▪ renovating or ▪ rehabilitating structures, equipment, or property The operation of a nuclear facility beyond its limits resulting in the consequences identified in columns categorization criteria of this table *When estimating loss or damage, follow the methods in DOE G 430.1-1

Categorization Criteria	Type of Investigation	
	Type A	Type B
	*When estimating loss or damage, follow the methods in DOE G 430.1-1, *Cost Estimating Guide*	
Other Effects	Any accident or series of accidents deemed appropriate by the Secretary or the Assistant Secretary for Environment, Safety and Health	Any accident or series of accidents deemed appropriate by the ▪ Secretary ▪ Assistant Secretary for Environment, Safety and Health ▪ Associate Deputy Secretary for Field Management; ▪ Cognizant Secretarial Officer ▪ Heads of field elements

Source: DOE G 225.1A-1

c. Compare and contrast immediate, short term, and long-term actions taken as the result of a problem identification or an occurrence.

The following is taken from DOE G 225.1A-1.

[Note: Corrective actions are no longer classified as short term and long term.]

Corrective actions fall into four categories:

1. Immediate corrective actions that are taken by the organization managing the site where the accident occurred to prevent a second or related accident.
2. Corrective actions required to satisfy judgments of need identified by the board in the final report. These corrective actions are developed by the heads of field elements and/or contractors responsible for the activities resulting in the accident and are designed to prevent recurrence and correct system problems.
3. Corrective actions determined by the appointing official to be appropriate for DOE-wide application. The appointing official recommends these corrective actions when the report is distributed.
4. DOE headquarters corrective actions that result from discussions with senior management. These actions usually address DOE policy.

d. Given event and/or occurrence data, apply problem analysis techniques and identify the problems and how they might have been avoided.

This is a performance-based competency. The Qualifying Official will evaluate its completion.

e. Describe various data gathering techniques and the use of trending/history when analyzing problems.

The following is taken from DOE G 225.1A-1.

Data Gathering Techniques

Reviewing Standards and Requirements

The board should identify DOE Orders and standards; Federal and state regulations; other external regulatory requirements; and site-specific policies, requirements, or guidelines applicable to the accident. This is necessary to establish the requirements governing work at the site where the accident occurred, determine what role they played in the accident, and ensure that policy issues are adequately addressed during the investigation.

Gathering Physical Evidence

Physical evidence should be gathered and a record made of all facts from all sources, including witness statements and interview transcripts, as soon as they become available.

Photographs, videotapes, and sketches should be used for recording and documenting the accident scene. The readiness team should document the accident scene initially (even though the board may wish to record the scene later as well). It is important to record the location, orientation, and subject matter for each photograph. Photographic coverage should be detailed, complete, and, if necessary, should include standard references to help establish distance, perspective, color, and date.

Gathering Documentary and Electronic Systems Evidence

Preserving documentary evidence, data, and information is an important consideration. This evidence might be on paper, videotape, magnetic tape, or computer media, either in an area of proximity to the accident or in files at other locations. Such evidence may include items such as permits, reports, analyses, logbooks, work process documentation, instrument charts, as-built drawings, entry control records, maintenance tags, and process records.

Conducting Interviews

Human evidence can be extremely delicate. Eyewitnesses can forget, overlook, or fail to recall evidence of critical value to the investigation. Individuals naturally begin to rationalize the circumstances of traumatic accidents after the event. Therefore, to preserve accuracy, the preferred approach is to obtain and record initial eyewitness statements before the participants and witnesses leave the accident site.

Examining Organizational Concerns, Management Systems, and Line Management Oversight

Accident investigations must thoroughly examine organizational concerns, management systems, and line management oversight processes to determine whether deficiencies in these areas were root causes of the accident. This examination focuses on management systems, not on individuals. To find out why management systems were not effective in preventing the conditions leading to the accident, investigators should examine the components of the Department's ISM system as defined in DOE P 450.4, *Safety Management System Policy*, DOE P 411.1, *Safety Management Functions,*

Responsibilities, and Authorities Policy, and their accompanying implementation guidance.

Use of Trending/History

The following is taken from DOE G 414.1-5.

Identified problem findings and their associated causes should also be analyzed to determine the existence of trends to identify the same or similar occurrences, generic problems, vulnerabilities, and cross-functional weaknesses at the lowest level before significant problems result. Trending typically identifies problem categories, responsible organizations, and specific activities or conditions. Benefits of trending include

- the ability to document historical data consistently in measurable, visible terms;
- the identification of changes in performance as they occur; and
- the development of leading indicators that identify degrading trends.

A consistent trend-coding system would assist in analyzing the problem findings. This trending data should be constantly analyzed, updated and summarized; and the results should be reported to management.

To assist in analyzing and trending identified problem findings and developing corrective actions, the assessing organization and/or site/organization manager should determine the applicable guiding principles and core safety management functions for ISM outlined in DOE P 450.4, for each finding. This will assist managers in identifying broader causal factors that can reduce the potential for similar problem findings. Input for determining the guiding principles and core functions may be provided by the assessing individual/organization and/or the individuals evaluating each finding and designing applicable corrective actions.

17. Criticality safety personnel must demonstrate a working level knowledge of DOE O 420.1B, *Facility Safety*, with respect to its impact on the Department's criticality safety.

[Note: DOE O 420.1B has been superseded by DOE O 420.1B, chg 1.]

The information for the all of the KSAs in this competency statement is taken from DOE O 420.1B, chg 1, unless specified otherwise.

a. Discuss the purpose and objectives of the nuclear criticality safety requirements of DOE O 420.1B, including when they are applicable.

The objectives of this Order are to establish facility and programmatic safety requirements for DOE, including the NNSA, for

- nuclear and explosives safety design criteria,
- fire protection,
- criticality safety,
- natural phenomena hazards mitigation, and
- the system engineer program.

Except for the exclusions in paragraph 3c, the requirements in this Order apply to the types of DOE facilities established in the applicability paragraphs of each chapter of this Order. The requirements in this Order are applicable to Department employees.

The contractor requirements document (CRD) of this Order sets forth requirements that are to be applied to contractors with responsibility for the design, construction, management, operation, decontamination, decommissioning, or the demolition of DOE sites or facilities.

b. Discuss the following concepts associated with the nuclear criticality safety program:
- **Criticality safety program description document**
- **Qualification requirements of nuclear criticality safety staff**
- **Acceptable preparation methodologies for nuclear criticality safety evaluations**
- **Acceptable review methodologies for nuclear criticality safety evaluations**
- **Proper treatment of the requirements and recommendations from the ANSI/ANS 8-series standards**
- **Proper treatment of the double contingency principle recommendation and deviations from the principle**
- **Single failure vulnerability**
- **Preferred order of criticality safety controls**
- **Fissionable material accumulation**
- **Firefighting guidelines**

Criticality Safety Program Description Document

The CSP description document must describe how the contractor will implement the requirements in the CRD including the standards invoked by this chapter of DOE O 420.1B. The CSP description document must be approved by DOE and implemented as approved.

Criticality safety programs must be implemented to ensure that fissionable material operations will be evaluated and documented to demonstrate that operations will be sub-critical under both normal and credible abnormal conditions.

Qualification Requirements of Nuclear Criticality Safety Staff

Nuclear criticality safety staff responsible for implementing the CSP must be trained and qualified in accordance with a qualification program approved by DOE, unless the qualification program is consistent with DOE-STD-1135-99, *Guidance for Nuclear Criticality Safety Engineering Training and Qualification.*

Acceptable Preparation Methodologies for NCSEs

The methodology for conducting CSEs must be approved by DOE, unless the evaluations are conducted in accordance with the DOE-STD-3007-2007, or successor document and evaluated according to DOE-STD-1134-99, or successor document.

Acceptable Review Methodologies for NCSEs

The methodology for conducting CSEs must be approved by DOE, unless the evaluations are conducted in accordance with the DOE-STD-3007-2007, or successor document and evaluated according to DOE-STD-1134-99, or successor document.

Proper Treatment of the Requirements and Recommendations from the ANSI/ANS 8-Series Standards

Criticality safety programs must satisfy the requirements of the revisions to consensus NCS standards of ANSI/ANS 8, unless otherwise modified or approved by DOE.

All recommendations in applicable ANSI/ANS standards must be considered and a explanation provided to DOE through the CSP description document whenever a recommendation is not implemented

Proper Treatment of the Double Contingency Principle Recommendation and Deviations from the Principle.

The double contingency principle defined in ANSI/ANS 8.1, is a requirement that must be implemented for all fissionable material processes, operations, and facility designs within the scope of this chapter unless the deviation is documented, justified, and approved by DOE.

Single Failure Vulnerability

No single credible event or failure can result in a criticality.

Preferred Order of Criticality Safety Controls

The preferred order of controls must be passive engineered controls, active engineered controls, followed by ACs.

Fissionable Material Accumulation

Facilities that conduct operations using fissionable material in a form that could inadvertently accumulate in significant quantities must include a program and procedures for detecting and characterizing accumulations.

Firefighting Guidelines

Guidelines for firefighting must be established for areas within or adjacent to moderator-controlled areas. The criteria and process for developing the guidelines must be documented in the CSP description document.

c. Discuss the contractor's responsibilities with respect to the implementation of the requirements of DOE O 420.1B.

This CRD contained in DOE O 420.1B establishes facility safety requirements for contractors responsible for design, construction, operation, management, decontamination or decommissioning of DOE sites or facilities. Contractors must comply with the CRD requirements to the extent set forth in their contracts.

Chapters of the CRD may have general and specific requirements. In complying with the CRD, contractors must determine acceptability of design and operations based on a comparison with available safety basis information.

Contractors must ensure that any work done is consistent with any other safety, design, or other analysis or requirements applicable to the affected facility. In particular, work must be performed according to the ISM requirements of 48 CFR 970.5223-1, "Integration of Environment, Safety, and Health into Work Planning and Execution", and the QA requirements of either subpart A of 10 CFR 830, or DOE O 414.1C, *Quality Assurance*, or successor document, as applicable. All new construction, as a minimum, must comply with national consensus industry standards and the model building codes applicable for the state or region supplemented in a graded manner with additional safety requirements for the associated hazards in the facility that are not addressed by the codes.

DOE implementation guidance and technical standards referenced in the CRD are not mandatory; however they must be considered in conjunction with the specific requirements. Such guidance, along with DOE and industry standards referenced therein, represent acceptable methods to satisfy the provisions of the CRD. Alternate methods that satisfy the requirements of the CRD are also acceptable. Any implementation method selected must be justified to ensure that an adequate level of safety commensurate with the identified hazards is achieved.

d. Discuss the Management and Operating (M&O) contractor responsibilities for the following in relation to criticality safety activities:
- **Criticality safety evaluations**
- **Monitoring**
- **Surveillance**
- **Transportation**
- **Storage**

Criticality Safety Evaluations

Criticality safety programs must include
- criticality safety evaluations for fissionable materials operations that document parameters, limits, and controls required to maintain sub-criticality for all normal and credible abnormal conditions; and
- provisions for implementation of limits and controls identified by the CSEs.

The methodology for conducting CSEs must be approved by DOE, unless the evaluations are conducted in accordance with the DOE-STD-3007-2007, or successor document and evaluated in accordance with DOE-STD-1134-1999, or successor document.

Monitoring

The following is taken from DOE G 421.1-1.

Monitoring for Process Accumulations

Provide monitoring or surveillance, or both, to forewarn of unacceptable or unsafe accumulations of a significant quantity of fissionable materials in process equipment, storage areas, piping, and ventilation systems, thus permitting normal corrective actions. If unacceptable or unsafe accumulations of a significant quantity of fissionable materials are

detected, corrective actions should be taken in conjunction with the area criticality safety organization.

Control Monitoring

Nuclear criticality safety controls should include provisions for periodic evaluation by an inspection program, use of corrosion specimens, or other techniques, if credible corrosion or erosion could change the geometry (or thickness) in a system that depends on geometry (or thickness) for NCS.

Soluble Poison Monitoring

Two independent methods of determining the operating concentration of a soluble poison should be provided to confirm that the poison concentration limit is satisfied. The presence of soluble poison should be monitored at a frequency that provides for automatic or operator-initiated protective action in the event of process upsets. Extraordinary care should be taken with solutions of poisons because of the difficulty of exercising intended distribution and concentration control of solutes.

Monitoring Neutron Moderation

For operations in which NCS depends upon control of neutron moderation, there should be assurance that the prescribed extent of moderation remains unchanged or that, if a credible change occurs, the reactivity of the system remains below acceptable subcritical limits. Such assurance should include consideration of all credible accidents involving any moderator or combination of moderators.

Surveillance

The following is taken from DOE-STD-3007-2007.

Major changes to equipment and processes shall be subject to the change control process. That process shall ensure that revisions to CSEs will be reflected in the appropriate document as part of the overall change package.

All operations shall be reviewed at least annually to ascertain that procedures are being followed and that process conditions have not been altered so as to fall outside the scope of currently approved CSEs. Individuals who are knowledgeable in criticality safety and who, to the extent practicable, are not immediately responsible for the operation shall conduct these reviews, in consultation with operating personnel.

As part of the annual operations review, a check should be conducted to see if revisions to the safety basis documents are required. Any needed changes should be evaluated and addressed using the established unreviewed safety question determination (USQD), potential inadequate safety analysis, and justification for continued operation processes as appropriate.

Transportation

The following is taken from DOE M 460.2-1A.

Specific requirements that must be followed include:

Transportation Planning

The contractor must prepare a transportation plan with information on shipments of spent nuclear fuel (SNF), high-level waste (HLW), tritium-bearing reactor components, and transuranic (TRU) waste shipments and submit it to the appropriate field element. TRU waste shipments shall follow the WIPP transportation plan. Transportation plans describe operational strategy and delineate steps that will be taken to meet applicable regulatory and DOE requirements. Specific contents of transportation plans are determined by the program office and/or operations office, but should include organizational roles and responsibilities, material to be shipped, projected shipping dates, estimated number of shipments, mode of transport and carriers to be used, packages to be used, preferred and/or alternative routes, shipment pre-notifications required, safe parking arrangements, tracking systems that will be used, emergency preparedness and response plans, recovery and cleanup strategy, security and escort arrangements, and a public information plan.

At the direction of the responsible program office or field element, the contractor must prepare a transportation plan for low-level and mixed low-level, or other radioactive shipments. The contractor must also assist in preparing public information materials such as fact sheets, briefing packages, press releases, etc.

Routing

For non-classified shipments of SNF, HLW, and tritium-bearing reactor components, the contractor must perform an analysis of proposed routes using transportation models such as the Transportation Routing Analysis Geographic Information System.

Security

The contractor must ensure that a security plan is developed and includes an assessment of possible transportation security risks for radioactive shipments. The plan must contain sufficient background to understand the nature of threats against the radioactive shipment, the means to identify the vulnerabilities to those threats; and an approach to address the vulnerabilities.

- For shipments of SNF and HLW, security will be provided according to the requirements set forth in 10 CFR 73, “Physical Protection of Plants and Materials.” The contractor must ensure that in-transit requirements are addressed, including developing security plans, implementing physical security access controls, training, escorts, inspections, tracking, communications, and employee background checks. Transport security must be coordinated with state and tribal law enforcement officials and the involved railroad and motor carriers.
- For shipments of low-level waste, mixed low-level waste, and other radioactive shipments, the contractor will develop a security plan when required by 49 CFR 172.800, “Purpose and Applicability.”

Carrier/Driver Requirements

Upon request from the field element or DOE, the contractor must evaluate carriers. For carriers that transport highway route controlled quantities of radioactive material in less-than-truckload or truckload (TL) quantities, any TL quantities of radioactive material will be evaluated by the contractor. A copy of the evaluation document must be provided by

the contractor to the field element within thirty days after completion of the carrier evaluation.

Shipment Prenotification

Before shipping SNF or HLW within or through a state or tribal jurisdiction, the contractor must prepare a letter, to be sent by registered mail with return receipt, to the state governor (or the governor's designee) and tribal president or governor (or tribal designee) postmarked at least seven days before the shipment. Alternatively, a notification may be delivered by messenger to these same representatives at least four days before a shipment is transported within or through the state or tribal jurisdiction. A record must be retained in the shipping records file that includes the names of the persons contacted and the dates and times of the contacts. For low-level or mixed-low level waste, or other radioactive shipments, the contractor must provide notifications to affected states based on negotiated agreements.

Transportation Operational Contingencies

For SNF, HLW, tritium-bearing reactor components, and TRU waste shipments, the contractor must concur with the carrier that travel conditions are considered acceptable before dispatch and consider information provided by states or tribes regarding weather or road conditions. In the event of any substantial unanticipated delay in transit (greater than two hours), the contractor must notify the affected states and tribes of the event through the DOE Transportation Tracking and Communications System (TRANSCOM).

Tracking

The contractor must use TRANSCOM for tracking and monitoring the following categories of shipments:

- SNF
- Tritium-bearing reactor components
- HLW
- TRU waste (unless other arrangements have been made)

Inspections

The contractor must ensure pre-shipment inspections are done by the shipper and/or carrier to ensure that shipments meet the regulatory standards. For highway shipments of SNF, HLW, tritium-bearing reactor components, and TRU waste, the contractor must ensure shipments are made available, before departure, for inspection by certified state inspectors (commercial vehicle safety alliance enhanced North American standard inspection procedures, level VI) unless other arrangements have been made with the state.

Recovery and Cleanup

The contractor must review truckload carriers' plans for recovery and cleanup or verify they have a contract with a remediation company. For shipments of SNF, HLW, tritium-bearing reactor components, and TRU waste, the contractor must ensure carriers have specific written plans and procedures for providing recovery and cleanup in the event of an accident or incident.

The following was provided by a team of subject matter experts at the NNSA Service Center.

An allowance is made for an onsite transportation manual approved by DOE. The onsite manual would be expected to implement the transportation aspects of the NCS program.

Storage

The following is taken from 10 CFR 72.124.

Design for Criticality Safety

Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to NCS. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions.

Methods of Criticality Control

When practicable, the design of an independent spent fuel storage installation or monitored retrievable storage installation must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design must provide for positive means of verifying their continued efficacy. For dry spent fuel storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of the neutron absorbing materials cannot occur over the life of the facility.

Criticality Monitoring

A criticality monitoring system shall be maintained in each area where SNM is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Underwater monitoring is not required when SNM is handled or stored beneath water shielding. Monitoring of dry storage areas where SNM is packaged in its stored configuration under a license issued under this subpart is not required.

ANSI/ANS 8.7 is the governing criticality safety document for storage requirements. This standard is applicable to the storage of fissile materials. Mass and spacing limits are tabulated for uranium containing greater than 30 wt-% 235U, for 233U, and for plutonium, as metals and oxides. Criteria for the range of application of these limits are provided.

e. Discuss the role of Department criticality safety personnel with respect to the implementation of the requirements of DOE O 420.1B.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety personnel provide review and oversight of CSPs per DOE O 226.1A, *Implementation of Department of Energy oversight Policy*, using standards DOE-STD-1158-2010, *Self-Assessment Standard for DOE Contractor Criticality Safety Programs*, and DOE-STD-1134-1999 as the principal guidelines. The requirements include the requirements of the ANSI/ANS 8 series of standards.

18. Criticality safety personnel shall demonstrate a familiarity level knowledge of historical criticality safety-related requirements.

a. Describe the history of criticality safety standards, and inconsistencies between DOE Orders and those standards, including regulatory lessons learned. Useful documents may include the following:

- **LA-3366**
- **ANSI N16.1**
- **ANSI/ANS-8.1, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, prior and current versions**
- **DOE O 6430.1A, *General Design Criteria*, § 1300-4**
- **DOE O 5480.5, *Safety of Nuclear Facilities***
- **DOE O 5480.24, *Nuclear Criticality Safety***
- **DOE O 420.1, 420.1A, and 420.1B, *Facility Safety***
- **DOE-STD-1189-2008, *Integration of Safety Into the Design Process*, § 7.5 and Table 7-2**
- **LA-2063**
- **TID-7016**

LA-3366

This discussion of criticality control is intended to encourage a working knowledge on the part of those who design and perform operations with fissile material. As background, requirements of the Atomic Energy Act are interpreted, and nuclear-safety experience is outlined. Both are shown to be compatible with reasonable principles of nuclear safety. Next, empirical criticality information is presented to help develop a feeling for conditions to be avoided during operations. Criticality control methods that are consistent with the stated principles and available criticality data are described in the final section.

ANSI N16.1

ANSI/ANS-8.1–183, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, was prepared by Subcommittee 8, Fissionable Materials Outside Reactors, of the Standards Committee of the ANS as a consolidation of revisions to ANSI N16.1-1975/ANS-8.1 and ANSI N16.9-1975/ANS-8.1 1. ANSI/ANS-8.1–1983 was approved by the ANS Committee N16, Nuclear Criticality Safety, in 1982 and by ANSI on October 7, 1983.

ANSI/ANS 8.1

ANSI/ANS-8.1 applies to handling, storing, processing, and transporting fissionable material outside nuclear reactors. The standard presents generalized basic criteria and specific limits (maximum subcritical) for some single units of simple shape containing ^{233}U, ^{235}U, ^{239}Pu, but not for multi-unit arrays. Further, the subcritical limits specified in the standard allow for uncertainties in the calculations and experimental data used in their

derivation but not for contingencies such as double-batching or failure of analytical techniques to yield accurate values.

This standard also delineates requirements for establishing the validity and area of applicability of a calculational method used in assessing NCS. However, it is concerned only with validating calculational methods and does not address important related questions such as the margin of safety to be used with the method or the qualifications of the personnel responsible for the data input.

DOE O 6430.1A, Section 1300.4

[Note: DOE O 6430.1A, § 1300 was canceled by DOE O 420.1.]

DOE O5480.5 and DOE O 5480.24

[Note: DOE Order 5480.5 was cancelled by DOE N 1321.40, and DOE Order 5480.24 was cancelled by DOE N 251.4. Current criticality requirements are contained in DOE O 420.1B, chg 1.)

DOE O 420.1B

General Requirements

Criticality safety programs must be implemented to ensure that fissionable material operations will be evaluated and documented to demonstrate that operations will be sub-critical under both normal and credible abnormal conditions.

No single credible event or failure can result in a criticality.

The CSP description document must describe how the contractor will implement the requirements in the CRD including the standards invoked by DOE O 420.1B. The CSP description document must be approved by DOE and implemented as approved.

Criticality safety programs must include the following:

- Criticality safety evaluations for fissionable materials operations that document parameters, limits, and controls required to maintain sub-criticality for all normal and credible abnormal conditions
- The preferred order of controls must be passive engineered controls, active engineered controls, followed by ACs
- Provisions for implementation of limits and controls identified by the CSEs
- Periodic reviews of operations and conditions to ensure that
  - limits and controls are effectively implemented
  - process conditions have not been altered resulting in compromise of SLs and controls

- Assessment of the need for and installation of criticality accident alarm and detection systems where appropriate to conform with chapter 3 of DOE O 420.1B

Nuclear criticality safety staff responsible for implementing the CSP must be trained and qualified according to a qualification program approved by DOE, unless the qualification program is compliant with DOE-STD-1135-99.

Specific Requirements

Criticality safety programs must apply to facilities and activities with fissionable materials operations as defined in DOE O 420.1B.

Criticality safety programs must satisfy the requirements of the revisions to consensus NCS standards of ANSI/ANS 8, unless otherwise modified or approved by DOE.

All recommendations in applicable ANSI/ANS standards must be considered, and an explanation provided to DOE through the CSP description document whenever a recommendation is not implemented.

The double contingency principle defined in ANSI/ANS 8.1, is a requirement that must be implemented for all processes, operations and facility designs within the scope of DOE O 420.1B, unless the deviation is documented, justified, and approved by DOE.

The methodology for preparing CSEs must be approved by DOE unless the evaluations are conducted in accordance with DOE-STD-3007-2007, or successor document and evaluated in accordance with DOE-STD-1134-1999, successor document.

Facilities that conduct operations using fissionable material in a form that could inadvertently accumulate in significant quantities must include a program and procedures for detecting and characterizing accumulations.

Guidelines for firefighting must be established for areas within or adjacent to moderator-controlled areas. The criteria and process for developing the guidelines must be documented in the CSP description document.

Nuclear Criticality Safety

Nuclear criticality safety represents a specialized safety discipline. Given the significance of an inadvertent nuclear criticality, the presence of quantities of fissionable materials sufficient to sustain a critical reaction can determine the facility hazard categorization. Where there is sufficient fissionable material present, NCS controls can also result in safety significant functional classification of SSCs and, potentially, TSR controls. As a result, the NCS function must be represented on the project team and closely linked to the safety analysis effort from the earliest stages of project development. Criticality safety evaluations must be integrated with the traditional safety analysis techniques to provide a comprehensive safety analysis.

DOE has promulgated guidance for performing and documenting CSEs in DOE-STD-3007-2007.

To support design development, it is important to develop fundamental design criteria to address typical criticality safety concerns (e.g., safe geometry) and to incorporate these criteria early in the design process. The purpose of these criteria is to avoid the use of cumbersome and inherently less reliable ACs. An example set of design criteria is provided in table 7-2 of DOE-STD-1189-2008, *Integration of Safety Into the Design Process*.

One of the most important criticality safety design features is to prevent, by design, natural phenomena initiators for criticality accidents (e.g., seismic and wind). In addition, the fire protection program at design will also drive criticality safety design requirements. The building code, life safety code, national fire codes, and DOE directives almost always require automatic sprinkler protection and firefighting hose capability as well as suitable drainage for nuclear facilities. Unless exemptions and variances have been approved for automatic and manual fire suppression, the criticality calculations must take this moderator into account. For example, in a facility where sprinklers are planned, the CSEs must consider the effects of introduction of water due to sprinkler activation. The presence of sprinklers will also tend to drive engineered controls for criticality safety to prevent water ingress to fissionable material containers, both in process containers and in storage. Criticality concerns could also result in a change from water-based sprinklers to an alternative gaseous suppression system, which could affect cost estimates. Therefore, there is a need for close cooperation between fire protection/fire hazards analysis and criticality safety early in design.

Criticality safety includes human interaction with the potential criticality hazard. Addressing human interaction issues typically results in ACs. Minimizing use of ACs in lieu of more reliable engineered controls should be a focal point for design. This also points to the need to identify criticality safety issues early in the design process and design the facility in such a way as to preclude criticality problems (e.g., provide storage appropriate for the types of materials; design systems that can be used by the operator in a way that ensures criticality safety; and consider criticality potential when designing sprinkler systems and other fire protection). One specific aspect of NCS operations requiring early project definition is emergency response to criticality accidents.

Designs should strive to make a criticality accident a beyond-extremely-unlikely event. If that is not practical, the double contingency principle requires control of two independent parameters. Deviations from the double contingency principle must be specifically approved by DOE and typically results in layers of ACs. A singular focus of criticality safety-in-design should be to avoid the need for single parameter control in all processes where a criticality accident is credible.

DOE-STD-1189-2008, Section 7.5 and Table 7-2

Section 7.5, Nuclear Criticality Safety

Nuclear criticality safety represents a specialized safety discipline. Given the significance of an inadvertent nuclear criticality, the presence of quantities of fissionable materials sufficient to sustain a critical reaction can determine the facility hazard categorization. Where there is sufficient fissionable material present, NCS controls can also result in safety significant functional classification of SSCs and, potentially, TSR controls. As a result, the NCS function must be represented on the project team and closely linked to the safety analysis effort from the earliest stages of project development. Criticality safety evaluations must be integrated with the traditional safety analysis techniques to provide a comprehensive safety analysis. DOE has promulgated guidance for performing and documenting CSEs in DOE-STD-3007-2007.

To support design development, it is important to develop fundamental design criteria to address typical criticality safety concerns (e.g., safe geometry) and to incorporate these

criteria early in the design process. The purpose of these criteria is to avoid the use of cumbersome and inherently less reliable ACs. An example set of design criteria is provided in table 7-2.

TID-7016 and LA-2063

The following was taken from the abstract of TID-7016, revision 2, available on the LLNL database.

TID-7016 was first issued in 1956 as classified AEC report LA-2063 and was reprinted the next year, unclassified, as TID-7016. Revision 1, published in 1961, extended the scope and refined the guiding information. The present revision of the guide differs significantly from its predecessor in that the latter was intentionally conservative in its recommendations. Firmly based on experimental evidence of criticality, the original guide and the first revision were considered to be of most value to organizations whose activities with fissionable materials were not extensive and, secondarily, that it would serve as a point of departure for members of established nuclear safety teams, experienced in the field.

The reader will find a significant change in the character of information presented in revision 2, which was published in 1978. Nuclear criticality safety has matured. The advance of calculational capability has permitted validated calculations to extend and substitute for experimental data. The broadened database has enabled better interpolation, extension, and understanding of available information, especially in areas previously addressed by undefined but adequate factors of safety. The content has been thereby enriched in qualitative guidance. The information inherently contains, and the user can recapture, the quantitative guidance.

19. Criticality safety personnel must demonstrate a working level knowledge of the following criticality safety-related ANSI/ANS standards:

- **ANSI/ANS-8.1, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors***
- **ANSI/ANS-8.3, (ANSI N-16.2), *Criticality Accident Alarm System***
- **ANSI/ANS-8.5, (ANSI N-16.4), *Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material***
- **ANSI/ANS-8.7, *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials***
- **ANS-8.14, *Use of Soluble Absorbers in Nuclear Facilities Outside of Reactors***
- **ANSI/ANS-8.15, *Nuclear Criticality Control of Special Actinide Elements***
- **ANSI/ANS-8.19, *Administrative Practices for Nuclear Criticality Safety***
- **ANSI/ANS-8.20, *Nuclear Criticality Safety Training***
- **ANSI/ANS-8.21, *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors***
- **ANSI/ANS-8.22, *Nuclear Criticality Safety Based on Limiting and Controlling Moderators***
- **ANSI/ANS-8.23, *Nuclear Criticality Accident Emergency Planning and Response***
- **ANSI/ANS-8.24, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations***

- **ANSI/ANS-8.26, *Criticality Safety Engineer Training and Qualification Program***

a. Describe the contents, requirements, and relationships among the above ANSI/ANS Standards.

ANSI/ANS 8.1

ANS 8.1 gives general safety philosophy, such as who is responsible for safety, the proper role of criticality safety staff, the requirement that processes be subcritical under normal and credible abnormal conditions, the famous double-contingency recommendation, the principle of relying on geometry control in preference to AC, and subcritical limits for some common conditions with the three most common fissile isotopes. Subcritical limits for several enrichments of ^{235}U are also given.

ANSI/ANS 8.3, (ANSI N-16.2)

This standard is applicable to all operations involving fissionable materials in which inadvertent criticality can occur and cause personnel to receive unacceptable exposure to radiation. This standard is not applicable to detection of criticality events where no excessive exposure to personnel is credible, nor to nuclear reactors or critical experiments. This standard does not include details of administrative actions or of emergency response actions that occur after alarm activation.

ANSI/ANS 8.5 (ANSI N-16.4)

This standard provides guidance for the use of borosilicate-glass Raschig rings as a neutron absorber for criticality control in ring-packed vessels containing solutions of ^{235}U, ^{239}Pu, or ^{233}U. The chemical and physical environment, properties of the rings and packed vessels, maintenance inspection procedures, and operating guidelines are specified.

ANSI/ANS 8.7

This standard is applicable to the storage of fissile materials. Mass and spacing limits are tabulated for uranium containing greater than 30 wt-% ^{235}U, for ^{233}U, and for plutonium, as metals and oxides. Criteria for the range of application of these limits are provided.

ANSI/ANS 8.14

This standard provides guidance for the use of soluble neutron absorbers for criticality control. This standard addresses neutron absorber selection, system design and modifications, safety evaluations, and quality control programs.

ANSI/ANS 8.15

This standard is applicable to operations with the following: nitrogen, plutonium, americium, curium, and californium. Subcritical mass limits are presented for isolated fissionable units. The limits are not applicable to interacting units.

ANSI/ANS 8.19

This standard provides criteria for the administration of a NCS program for outside-of-reactor operations in which there exists a potential for nuclear criticality accidents. Responsibilities of management, supervision, and the NCS staff are addressed. Objectives and characteristics of operating and emergency procedures are included.

ANSI/ANS 8.20

This standard provides criteria for NCS training for personnel associated with operations outside reactors where a potential exists for criticality accidents. It is not sufficient for the training of NCS staff.

ANSI/ANS 8.21

This standard provides guidance for the use of fixed neutron absorbers as an integral part of nuclear facilities and fissionable material process equipment outside reactors, where such absorbers provide criticality safety control.

ANSI/ANS 8.22

This standard applies to limiting and controlling moderators to achieve criticality safety in operations with fissile materials in a moderator control area. This standard does not apply to concentration control of fissile materials.

ANSI/ANS 8.23

This standard provides criteria for minimizing risks to personnel during emergency response to a nuclear criticality accident outside reactors. This standard applies to those facilities for which a CAAS, as specified in American national standard criticality accident alarm system, ANSI/ANS-8.3 1997 (R2003), is in use. This standard does not apply to nuclear power plant sites or to licensed research reactor facilities, which are addressed by other standards.

ANSI/ANS 8.24

This standard provides requirements and recommendations for validation, including establishing applicability, of neutron transport calculational methods used in determining critical or subcritical conditions for NCS analyses.

ANSI/ANS 8.26

This standard presents the fundamental content elements of a training and qualification program for individuals with responsibilities for performing the various technical aspects of criticality safety engineering. The standard presents a flexible array of competencies for use by management to develop tailored training and qualification programs applicable to site-specific job functions, facilities, and operations.

b. Discuss the applicability of the above ANSI/ANS Standards to the Department facilities and processes.

DOE O 420.1B, chg 1, states that CSPs must satisfy the requirements of the revisions to consensus NCS standards of the ANSI/ANS 8 series in effect as of the date of the Order, unless otherwise modified or approved by DOE. All recommendations in applicable ANSI/ANS standards must be considered and a explanation provided to DOE through the CSP description document whenever a recommendation is not implemented.

The Order goes on to state that the double-contingency principle defined in ANSI/ANS 8.1, is a requirement that must be implemented for all fissionable material processes, operations, and facility designs within the scope of DOE O 420.1B, unless the deviation is documented, justified, and approved by DOE.

c. Discuss the role of the Department's criticality safety personnel in implementing the requirements of these standards.

The following is taken from DOE-STD-1158-2010.

The NCS staff provides technical guidance for design of equipment, processes, and procedures; the staff reviews modifications to equipment, process, and procedures involving fissionable material; the staff maintains familiarity with criticality codes, guides, standards, and best practices; the staff interacts, both internally and externally, understands the physics of criticality and makes use of experimental data, handbook data, and bounding methods where applicable; the staff participates in training personnel; the staff participates in audits of operations; and the staff examines reports of procedural violations and criticality infractions and recommends improvements in safety practices to management.

The following was provided by a team of subject matter experts at the NNSA Service Center.

These sections require collaboration from the contractor. Collaboration cannot, by its nature, be a unilateral obligation.

The Federal criticality safety oversight includes both assessments and cooperative guidance and leadership of the site CSP. Professional collaboration with the site criticality safety management and operations is essential to a successful program.

20. Criticality safety personnel must demonstrate a familiarity level knowledge of the following criticality safety-related ANSI/ANS standards:

- **ANSI/ANS-8.6, *Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ***
- **ANSI/ANS-8.10, *Criteria for Nuclear Criticality Safety Controls in Operations With Shielding and Confinement***
- **ANSI/ANS-8.12, *Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors***
- **ANSI/ANS-8.17, *Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors***
- **ANSI/ANS 8.27, *Burnup Credit for LWR Fuel***

a. Describe the contents, requirements, and relationships between the above Orders and Technical Standards.

ANSI/ANS 8.6

This standard provides safety guidance for conducting subcritical neutron-multiplication measurements where physical protection of personnel against the consequences of a criticality accident is not provided. The objectives of in situ measurements are either to confirm an adequate safety margin or to improve an estimate of such a margin.

ANSI/ANS 8.10

This standard is applicable to operations outside of nuclear reactors with ^{235}U, ^{233}U, ^{239}Pu, and other fissile and fissionable materials in which shielding and confinement are provided for protection of personnel and the public, except the assembly of these materials

under controlled conditions, such as in critical experiments. Criteria are provided that may be used for criticality control under these conditions. This standard does not include the details of administrative procedures for control or details regarding the design of processes and equipment or descriptions of instrumentation for process control.

ANSI/ANS 8.12

This standard is applicable to operations with plutonium-uranium oxide fuel mixtures outside nuclear reactors, except the assembly of these materials under controlled conditions, such as in critical experiments. Basic criteria are presented for plutonium-uranium fuel mixtures in single units of simple shape containing no more than 30 wt% plutonium combined with uranium containing no more than 0.71 wt% ^{235}U.

ANSI/ANS 8.17

This standard provides NCS criteria for the handling, storage, and transportation of light water reactor fuel rods and units outside reactor cores.

ANSI/ANS 8.27

This standard provides criteria for accounting for reactivity effects of fuel irradiation and radioactive decay in criticality safety control of storage, transportation, and disposal of commercial light water reactor UO_2 fuel assemblies. This standard assumes the fuel and any fixed burnable absorbers are contained in an intact assembly. Additional considerations could be necessary for fuel assemblies that have been disassembled, consolidated, damaged, or reconfigured in any manner.

b. Describe the role of criticality safety personnel with respect to the requirements in these standards.

The following is taken from DOE-STD-1158-2010.

The NCS staff provides technical guidance for design of equipment, processes, and procedures; the staff reviews modifications to equipment, process, and procedures involving fissionable material; the staff maintains familiarity with criticality codes, guides, standards, and best practices; the staff interacts, both internally and externally, understands the physics of criticality and makes use of experimental data, handbook data, and bounding methods where applicable; the staff participates in training personnel; the staff participates in audits of operations; and the staff examines reports of procedural violations and criticality infractions and recommends improvements in safety practices to management.

The following was provided by a team of subject matter experts at the NNSA Service Center.

These sections require collaboration from the contractor. Collaboration cannot, by its nature, be a unilateral obligation.

The Federal criticality safety oversight includes both assessments and cooperative guidance and leadership of the site CSP. Professional collaboration with the site criticality safety management and operations is essential to a successful program.

21. Criticality safety personnel must demonstrate a familiarity level knowledge of the following criticality safety experiment related ANSI/ANS standards:

- **ANSI/ANS-1, *Conduct of Critical Experiments***
- **ANSI/ANS-14.1, *Operation of Fast Pulse Reactors***

a. Discuss when ANS 1 is applicable in a critical experiments facility and when these standards (ANS-1 or ANS 14.1) become the governing criteria as opposed to ANS 8.1.

ANS 1 provides for the safe conduct of critical experiments. Such experiments study neutron behavior in a fission device where the energy produced is insufficient to require auxiliary cooling, and the power history is such that the inventory of long-lived fission products is insignificant. ANS 8.1 is applicable to operations with fissionable materials outside nuclear reactors, except for the assembly of these materials under controlled conditions, such as in critical experiments.

b. Discuss the similarities between ANS 1 and ANS 8.6.

ANS 8.6 provides safety guidance for conducting subcritical neutron-multiplication measurements where physical protection of personnel against the consequences of a criticality accident is not provided. The objectives of in situ measurements are either to confirm an adequate safety margin or to improve an estimate of such a margin.

c. Discuss the differences in safety device actions between operations governed by ANS 1 and ANS 14.1.

ANS-1

ANSI/ANS-1 offers the following regarding safety device actions.

Each critical assembly shall be provided with at least two safety devices that are actuated automatically at a preset radiation level and that can be actuated manually. Each device shall be capable of shutting down the assembly. These devices shall also be capable of removing reactivity more rapidly than it can be added by any anticipated operation. The safety devices shall be able to perform their safety function independent of the assembly control system.

At least two neutron or gamma-ray detectors together with associated electronic and mechanical components shall be capable of independently initiating a scram of the critical assembly at a preset radiation level.

Loss of actuating power to any critical assembly safety device shall produce a scram.

ANSI/ANS-14

ANSI/ANS-14 offers the following regarding safety device actions.

At least two independent safety devices (such as safety block and control elements) shall be provided. The safety devices shall be fail-safe with respect to loss of electrical power.

The primary safety device (such as safety block movement) shall be capable of shutting down the reactor under conditions corresponding to the most reactive experiment arrangement.

d. Discuss the similarities in management practices among ANS 1, ANS 8.1, and ANS 14.1.

The following management practices are common among ANS-1, ANS 8.1, and ANS 14.1.

Management assigns responsibility and authority for the safe operation of critical experiments through the line organization.

Written general operational restrictions for each critical facility are approved by management and the cognizant regulating authority. Criteria for the safety assessment may be derived from ANSI/ANS-15.21-1996, *American National Standard Format and Content for Safety Analysis Reports for Research Reactors*. Criteria for operational restrictions may be derived from ANSI/ANS-15.1-1990, *American National Standard for the Development of Technical Specifications for Research Reactors*.

Each new program of experiments are documented, independently reviewed, and approved in a manner established by management.

Operations personnel are trained to ensure that they are capable of performing their assigned work.

At least two persons who have qualifications approved by management shall be present while a critical experiment is performed or in the reactor facility during operation of the reactor.

An emergency plan approved by management shall be in effect, and emergency equipment shall be provided.

Knowledgeable personnel, independent of the reactor operating staff, shall review the reactor's administrative, operational, and safety practices at least annually. The results of the review shall be documented.

e. State the multiplication criteria limits that should be applied to manually performed criticality experiments.

ANSI/ANS-1, *Conduct of Critical Experiments*, states that manual operations that result in reactivity additions to a critical assembly should be limited to a predicted effective multiplication factor (keff) of 0.9 for unknown configurations.

f. Give a brief explanation of the physics reason(s) for ANS-1, Section 4.4.

ANS-1 provides for the safe conduct of critical experiments. Such experiments study neutron behavior in a fission device where the energy produced is insufficient to require auxiliary cooling, and the power history is such that the inventory of long-lived fission products is insignificant.

See DOE-HDBK-1019/2-93, module 4, section 3 for an explanation of the concepts regarding reactor startup, operation, and shutdown.

22. Criticality safety personnel must demonstrate a working level knowledge of assessment techniques (such as the planning and use of observations, interviews, and document reviews) to assess facility performance, report results of assessments, and follow up on actions taken as the result of assessments.

a. Describe the role of criticality safety personnel in the assessment of Government Owned Contractor Operated (GOCO) and Government Owned Government Operated (GOGO) facilities.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety personnel should

- spend time in the field, doing oversight by wandering around
- review contractor NCS budgets and staffing plans
- establish and monitor NCS performance measures
- interface with contractor NCS personnel
- review all NCS related occurrences & incidents
- review some sampling of contractor CSEs

The following is taken from DOE-STD-1158-2010.

The standard lists NCS staff responsibilities as follows:
The NCS staff is comprised of specialists skilled in the techniques of NCS assessment and familiar with plant operations while, to the extent practicable, administratively independent of line management; the staff provides technical guidance for design of equipment, processes, and procedures; the staff reviews modifications to equipment, process, and procedures involving fissionable material; the staff maintains familiarity with criticality codes, guides, standards, and best practices; the staff interacts, both internally and externally, having access to criticality safety professionals to provide assistance as needed; the staff understands the physics of criticality and makes use of experimental data, handbook data, and bounding methods where applicable; the staff participates in training personnel; the staff participates in audits of operations; and the staff examines reports of procedural violations and criticality infractions and recommends improvements in safety practices to management.

b. Describe how DOE-STD-1158-2002, Self-Assessment Standard for DOE Contractor Criticality Safety Programs, should be used in assessments.

[Note: DOE-STD-1158-2002 has been superseded by DOE-STD-1158-2010.]

The following is taken from DOE-STD-1158-2010.

The purpose of this document is to provide an assessment guide for review of DOE contractor CSPs. Although titled as a self-assessment standard, it is often used by DOE and external review teams. This standard may be used for evaluating NCS programs for facilities and activities that involve, or potentially involve, nuclides in quantities that are

equal to or greater than the single parameter limits for fissionable materials listed in ANSI/ANS-8.1 and 8.15.

c. Describe the assessment requirements and limitations associated with the interface with contractor employees.

The following was provided by a team of subject matter experts at the NNSA Service Center.

During assessments, it is important that contractor management and employees know the scope of the assessment. Also (and it ought to go without saying) assessments are against requirements, not opinion. The Department criticality safety people have to be rather careful in interface with contractors in the following areas:

- Any comment that appears to give direction may be taken as direction whether intended or not.
- Department criticality people may be viewed by some contractor personnel as larger-than-life authorities whether they are or not.
- Some contractor personnel will try to "fix" items that may not be problems, as a result of an assessor's questions.
- Information from interviews needs to be cross checked to be sure it is factual. Employees may protect contractor management, or color facts darker than they really are due to perceived wrongs.

d. Discuss the essential elements of a performance-based assessment including the following:

- **Investigation**
- **Fact finding**
- **Exit interview**
- **Reporting**
  - **Including review for factual accuracy**
- **Follow-up**
- **Closure**

The information for KSAs d and e is taken from DOE G 414.1-1B.

Investigation/Fact Finding

Effective assessments use a combination of tools and techniques to maximize the productivity of the assessment team and resources. Such assessment techniques include interviews, document reviews, observation, inspection, and performance testing.

Exit Interview/Meeting

This meeting is used primarily by the assessment team to present the assessment summary. Reasonable time should be allowed to discuss any concerns, but this meeting should not be used to argue the assessment findings or methodology. There should be no surprises during the exit meeting since the assessment team should have taken every effort possible during the conduct of the assessment to ensure that the assessed organization was aware of the team's findings and concerns.

Reporting

Assessment reports are required for documentation of assessment results. The assessment report should be clear, concise, accurate, and easy to understand, and should include only facts that directly relate to assessment observations and results. It should include sufficient information to enable the assessed organization to develop and implement appropriate improvement plans.

Follow-up/Closure

A follow-up assessment with special focus may be performed and should be completed in accordance with applicable corrective action documents. Particularly, this follow-up assessment should evaluate the effectiveness of corrective actions. A reasonable subset of corrective actions should be reviewed for effectiveness.

e. Describe the following assessment methods and the advantages or limitations of each method:
- **Document review**
- **Observation**
- **Interview**

Document Review

Document reviews provide the objective evidence to substantiate compliance with applicable requirements. A drawback is that the accuracy of the records cannot be ascertained by review alone. This technique should be combined with interviews, observation, inspection, and/or performance testing to complete the performance picture. Records and documents should be selected carefully to ensure that they adequately characterize the program, system, or process being assessed.

Observation

Observation, the viewing of actual work activities, is often considered the most effective technique for determining whether performance is in accordance with requirements. Assessors should understand the effect their presence has on the person being observed and convey an attitude that is helpful, constructive, positive, and unbiased. The primary goal during observation is to obtain the most complete picture possible of the performance, which should then be put into perspective relative to the overall program, system, or process.

Interview

Interviews provide the means of verifying the results of observation, document review, inspection, and performance testing; allow the responsible person to explain and clarify those results; help to eliminate misunderstandings about program implementation; and provide a venue where apparent conflicts or recent changes can be discussed and organization and program expectations can be described.

f. Describe the action to be taken if the contractor challenges the assessment findings and explain how such challenges can be avoided.

The following is taken from DOE O 414.1A (archived).

Corrective action plan (CAP) dispute resolution process.

- Disputes over the CAP or its implementation (such as timeliness or adequacy) must be resolved at the lowest possible organizational level. The organization that disagrees with the disposition of a given issue may elevate the dispute for timely resolution.
- The organization that disagrees with the disposition of a given issue must elevate the dispute in a step-wise manner through the management hierarchy.
- The dispute must be raised via a deliberate and timely dispute resolution process that provides each party with equal opportunity for input and a subsequent opportunity to appeal decisions up to the Secretary of Energy, if necessary.

The following is taken from DOE G 414.1B.

Assessments should be thorough and information gathered with sufficient diligence such that accurate, detailed conclusions can be provided to the organizations that will receive the final report. Assessors should maintain good records of the assessment results. These may include personal notes or other information to support the assessment, and may be included in the checklist information. These records are useful in writing the report, and any associated findings and recommendations, and will be valuable if questions arise during the report review process.

23. Criticality safety personnel must demonstrate a working level knowledge of DOE O 231.1A Chg 1, *Environment, Safety and Health Reporting*, and DOE M 231.1-2, *Occurrence Reporting and Processing of Operations Information* with respect to their impact on Department nuclear safety.

a. State the purpose of DOE O 231.1A, and DOE M 231.1-2.

DOE O 231.1A, Environment, Safety, and Health Reporting

The purpose of this Order is to ensure timely collection, reporting, analysis, and dissemination of information on environment, safety, and health issues as required by law or regulations, or as needed to ensure that DOE and NNSA are kept fully informed on a timely basis about events that could adversely affect the health and safety of the public or the workers, the environment, the intended purpose of DOE facilities, or the credibility of the Department.

DOE M 231.1-2, Occurrence Reporting and Processing of Operations Information

This manual provides detailed requirements to supplement DOE O 231.1A, *Environment, Safety, and Health Reporting*. Information gathered in response to the requirements in this Order and manual is used for analysis of the Department's performance in environmental protection and the safety and health of its workers and the public. This information is also used to develop lessons learned and document events that impact DOE operations. This manual is approved for use by all DOE elements and their contractors.

b. Define the following terms:

- **Event**
- **Condition**
- **Facility**
- **Notification report**

- **Occurrence report**
- **Reportable occurrence**

The information for KSAs b through f is taken from DOE M 231.1-2.

Event

An event is something significant and real-time that happens (e.g., pipe break, valve failure, loss of power, environmental spill, earthquake, tornado, flood).

Condition

A condition is any as-found state, whether or not resulting from an event, that may have adverse safety, health, QA, operational, or environmental implications. A condition is usually programmatic in nature; for example, errors in analysis or calculation, anomalies associated with design or performance, or items indicating a weakness in the management process are all conditions.

Facility

A facility is defined as any equipment, structure, system, process, or activity that fulfills a specific purpose. Examples include accelerators, storage areas, fusion research devices, nuclear reactors, production or processing plants, coal conversion plants, magneto hydrodynamic experiments, windmills, radioactive waste disposal systems and burial grounds, environmental restoration activities, testing laboratories, research laboratories, transportation activities, and accommodations for analytical examinations of irradiated and unirradiated components.

Notification Report

A notification report is the initial documented report to the Department of an event or condition that meets the reporting criteria defined in DOE M 231.1-2.

Occurrence Report

An occurrence report is a documented evaluation of an event or condition that is prepared in sufficient detail to enable the reader to assess its significance, consequences, or implications, and to evaluate the actions being proposed or employed to correct the condition or to avoid recurrence.

Reportable Occurrence

A reportable occurrence is an occurrence to be reported in accordance with the criteria defined in DOE M 231.1-2.

c. Discuss the Department's policy regarding the reporting of occurrences as outlined in DOE O 231.1A, *Environment, Safety, and Health Reporting*.

[Note: The Department's policy regarding the reporting of occurrences is now outlined in DOE M 231.1-2.]

To implement the occurrence categorization, notification, reporting, and processing system, the key responsible personnel must be identified and procedures developed, approved, and implemented to ensure that all of the occurrence reporting requirements are

met. The facility manager must be available at all times to carry out the responsibilities for the categorization, notification, and reporting requirements. Facility operators are required to ensure that occurrences resulting from activities performed by subcontractors in support of facility operation are reported in accordance with the provisions of this manual.

For reportable occurrences, facility personnel are required to categorize the occurrences, notify DOE as required, and prepare and submit occurrence reports.

The documentation and distribution requirements will be satisfied through a centralized, unclassified DOE operational database, the computerized occurrence reporting and processing system (ORPS). However, under no circumstances will occurrence reports containing classified information or unclassified controlled nuclear information (UCNI) be entered into the ORPS database.

Occurrences involving foreign personnel, governments, organizations, entities, or influence must be reported by the facility manager to the Office of Counterintelligence or the Office of Defense Nuclear Counterintelligence, as appropriate. Such reporting is not intended to interfere with or delay any actions directed toward protection of personnel or property.

d. State the different categories of reportable occurrences and discuss each.

The facility manager must categorize all occurrences, except operational emergencies, within two hours of discovery by the cognizant facility staff following the site/facility-specific procedures developed in accordance with section 9 of DOE M 231.1-2. The significance categories are for those occurrences of interest for complex-wide occurrence reporting and are described very generally below.

Operational emergencies. Operational emergencies are defined in DOE O 151.1C, *Comprehensive Emergency Management*. Operational emergency occurrences are the most serious occurrences and require an increased alert status for onsite personnel and, in specified cases, for offsite authorities. The prompt notification requirements, definitions, criteria, and classifications of operational emergencies and appropriate responses are provided in DOE O 151.1C. Written occurrence reports must be completed in accordance with DOE M 231.1-2.

Significance category 1. Occurrences in this category are those that are not operational emergencies and that have a significant impact on safe facility operations, worker or public safety and health, regulatory compliance, or public/business interests.

Significance category R. Occurrences in this category are those identified as recurring, as determined from the periodic performance analysis of occurrences across a site.

Significance category 2. Occurrences in this category are those that are not operational emergencies and that have a moderate impact on safe facility operations, worker or public safety and health, regulatory compliance, or public/business interests.

Significance category 3. Occurrences in this category are those that are not operational emergencies and that have a minor impact on safe facility operations, worker or public safety and health, regulatory compliance, or public/business interests.

Significance category 4. Occurrences in this category are those that are not operational emergencies and that have some impact on safe facility operations, worker or public safety and health, or public/business interests. If the consequences are not fully determined or the event exceeds the threshold of more than one criterion, the event must be categorized at the higher criteria level being considered. The occurrence criterion must be continuously reevaluated and changed as needed when new information becomes available.

e. Discuss the categorization, notification, and timeliness requirements associated with the following:
- **Notification report**
- **Final report**
- **Closing out and verifying occurrence reports**
- **Contractor occurrence reporting procedures**

Notification Report

The written notification report must be submitted according to the following schedule:
- Reports for operational emergencies and significance category 1 occurrences must be submitted before the close of the next business day from the time of categorization (not to exceed 80 hours).
- Reports for significance categories r and 2 occurrences must be submitted before the close of the next business day from the time of categorization.
- Reports for significance category 3 occurrences must be submitted no later than close of business on the second business day from the time of categorization.
- Reports for significance category 4 occurrences must be submitted by close of business the second business day from the time of categorization. Only a short form report is required.

Final Report

The facility manager must prepare and submit the final report as soon as practical, but within 45 calendar days after initial categorization of the occurrence.

Closing Out and Verifying Occurrence Reports

The following steps describe an acceptable process for closing out the final report for all occurrences except those categorized as significance category 4:
- The final report must be prepared by the facility manager and submitted as soon as practical, but within 45 calendar days after initial categorization of the occurrence.
- If the required analysis cannot be completed within 45 calendar days after initial categorization, an update report must be submitted within the 45 days. The update report must provide a detailed explanation of the delay and provide an estimated date for submittal of the final report. This information must be reported in the "Evaluation" block of the occurrence report. It is expected that the analysis of most occurrences will be completed and the final report will be submitted within the 45 calendar days. However, for certain occurrences, such as those requiring an accident investigation, it is understood that the information required for the final

report may not be available within this time. For occurrences resulting in an accident investigation, all direct, contributing, and root causes identified in the accident investigation report, as well as the corrective actions developed in response to the judgments of need, must be included in the final report.

- For operational emergencies and significance category 1, R, and 2 final reports, the facility representative must review, approve, and add any comments, as necessary, within 14 calendar days after receipt of the report. For operational emergencies and significance category 1 final reports, after the facility representative has approved the occurrence report, the program manager must review, approve, and add any comments to the final report within 14 calendar days. If the ORPS database is being used, the facility representative and program manager's comments should be provided through ORPS. Facility representative and program manager comments are not required for their approval of the report.
- If the final report is not approved by the facility representative or the program manager, the facility representative or program manager who is rejecting the report must provide the reason for disapproval in the comment section of the report at the time the action is taken. The revised final report must be resubmitted within 21 calendar days of the disapproval. If it cannot be resubmitted within this time, an update report must be submitted within the 21 calendar days explaining the delay and providing an estimated date for re-submittal of the final report. This information must be reported in the "Evaluation" block of the occurrence report.
- All occurrence reports must be distributed as soon as practical to the following:
  - Facility representative
  - Program manager
  - Heads of all field organizations (including NNSA)
  - Office of Environment, Safety and Health (Office of Performance Assessment and Analysis) and Administrator (NNSA)
  - DOE management and operations or integrating contractors

  If the occurrence reports are entered into the ORPS database, the distribution requirement is automatically satisfied.

- As prescribed on the occurrence reporting model, and depending on the significance category, the facility manager must track all corrective actions to closure, including independent verification or sampling at the facility level, and must also evaluate the effectiveness of the corrective actions to prevent recurrence (if applicable). Site/contractor corrective action programs must include management of significance category 4 occurrences whose corrective actions are not managed through ORPS.
- The cognizant facility manager may use the ORPS database to track the status of final report corrective actions. For those facilities that do not choose to use ORPS to track the status of their corrective actions, the specific corrective action tracking number from the local corrective action tracking system must be entered into ORPS. Any changes made to the corrective actions tracked in the local corrective action system must follow the site's approved change process and should be updated in ORPS. For significance category 2 and higher reports, any text change to a corrective action previously entered in ORPS must be updated in ORPS with facility representative approval. A status report of all incomplete occurrence

reports (not final) and incomplete corrective actions (for those sites that choose to track the status of their corrective actions using ORPS) will be available at any time from the ORPS database. Retain all supporting information pertaining to each occurrence or report (e.g., graphs, analyses, and formal investigation reports) in accordance with departmental records disposition schedules.

Contractor Occurrence Reporting Procedures

The contractor must notify the DOE facility representative in a manner determined locally, and the DOE headquarters operations center (DOE HQ OC), as required, of the following reportable occurrences as soon as practical, but no later than 2 hours after categorization:

- All significance category 1 occurrences require a prompt notification to the facility representative and DOE HQ OC.
- All significance category 2 occurrences require a prompt notification to the facility representative and, if directed by the facility representative, to the DOE HQ OC.
- All significance category 3 occurrences require a prompt notification to the facility representative.
- Additionally, there are specific significance category 2, 3, and 4 occurrences that are identified as requiring prompt notification to the facility representative and DOE HQ OC.

f. Discuss the general process for preparing and submitting occurrence reports and their follow-up.

In preparing the notification report, and subsequently the final report, the following writing instructions must be followed:

- The report should enable the general reader to understand the basic what, who, when, where, and how of the event, as well as the safety issues involved and the actions taken.
- The subject/title and the first paragraph of the occurrence description should relay the essential nature of the event (i.e., a summary of the occurrence in newspaper style).
- All information should be clear and succinct. Avoid redundant and unnecessary text and lengthy log book accounts, unless a discussion of the event in chronological order is considered essential to understanding the event.
- Complex and more significant occurrences should warrant a greater level of detail. Significance category 4 occurrences would likely need only a short paragraph under occurrence description. However, all reports should present enough information so that the general reader understands why the event needs to be reported and what the effect is.
- Avoid jargon and uncommon or site/facility-specific abbreviations and acronyms. If used, acronyms should be initially spelled out.
- Unless necessary to record and explain the event (e.g., suspect/counterfeit items or material), use general descriptions of equipment, procedures, etc., rather than presenting lengthy detailed titles and the numbers and letters assigned to those items.
- Quantify the level of contamination, dose, release, and damage (e.g., estimate the acres of wild land burned) when possible, instead of merely stating a reportable limit was exceeded.

- Use the active rather than the passive voice whenever possible. For example, write, "the electrician severed the conduit" rather than "the conduit was severed."
- When appropriate for clarification, photos, sketches, and drawings must be maintained with the ORPS occurrence report record. In addition, sites are encouraged but not required to make photos, sketches, and drawings available via a webpage, with the webpage address included as a hyperlink in the ORPS report.

g. Using DOE O 231.1A, discuss the role of criticality safety personnel in nuclear safety-related reportable occurrences.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety personnel should review occurrence reports for trends that may adversely impact overall safety, specifically looking for degradation in controls and barriers that are relied on to maintain criticality safety, even though these controls may not be specifically related to criticality safety. (If ACs for other safety disciplines are lax, chances are it won't be long before criticality controls are also lax.) If the facility equipment shows a high failure rate, equipment used for criticality control may not be as robust as thought.

Criticality safety personnel should be involved in the review of occurrence reports that impact criticality or nuclear safety for cross-cutting issues and to ensure that criticality implications are correctly addressed.

h. Given an occurrence report, determine the following:
- **The adequacy of the review process used**
- **That causes were appropriately defined**
- **That corrective actions addressed causes**
- **That the lessons learned were appropriate**
- **That corrective actions have been completed**

This is a performance-based competency. The Qualifying Official will evaluate its completion.

i. Using an occurrence report involving criticality safety activities, identify and discuss the factors contributing to the occurrence.

This is a performance-based competency. The Qualifying Official will evaluate its completion.

24. Criticality safety personnel must demonstrate a familiarity level knowledge of DOE O 413.3A Chg 1, *Program and Project Management for the Acquisition of Capital Assets*, and DOE STD 1189-2008, *Integration of Safety into the Design Process*.

[Note: DOE O 413.3A Chg 1 has been superseded by DOE O 413.3B]

a. Identify the four project phases and four major decision points in an acquisition project.

The information for KSAs a and b is taken from DOE O 413.3B, which now defines projects in terms of five critical decisions rather than by phases.

Critical Decisions (CDs)

Within DOE, projects typically progress through five CDs, which serve as major milestones approved by the secretarial acquisition executive or acquisition executive. Each CD marks an authorization to increase the commitment of resources by DOE and requires successful completion of the preceding phase or CD. The amount of time between decisions will vary. The CDs are:

- CD-0, approve mission need. There is a need that cannot be met through other than material means.
- CD-1, approve alternative selection and cost range. The selected alternative and approach is the optimum solution.
- CD-2, approve performance baseline. definitive scope, schedule and cost baselines have been developed.
- CD-3, approve start of construction/execution. The project is ready for implementation.
- CD-4, approve start of operations or project completion. The project is ready for turnover or transition to operations, if applicable.

b. Identify the safety documents, and the DOE response documents [e.g., Safety Evaluation Report (SER)] associated with each critical decision.

CD-0

Perform pre-conceptual planning activities that focus on the program offices' strategic goals and objectives, safety planning, design, development of capability gaps, high-level project parameters, a rough order of magnitude estimate cost range, and schedule estimates.

Prepare a program requirements document (for NNSA only) that defines the ultimate goals which the project must satisfy.

For Hazard Category 1, 2, and 3 nuclear facilities, and to the specificity possible, document DOE expectations for safety-in-design. (Refer to DOE-STD-1189-2008.)

CD-1

Develop a risk management plan (RMP) and complete an initial risk assessment of a recommended alternative. This may be included in the project execution plan. For evaluating the safety-in-design strategy, prepare risk and opportunity assessments for input to the RMP. (Refer to DOE G 413.3-7 and DOE-STD-1189-2008.)

Document high performance and sustainable building provisions per Executive Order (EO) 13423, section 2(f), EO 13514, section 2, and sustainable environmental stewardship considerations per DOE O 450.1A, as amended, in the conceptual design report, acquisition strategy, and/or PEP, as appropriate. (Refer to DOE G 413.3-6 and DOE O 430.2B.)

Prepare a preliminary hazard analysis report (PHAR) for facilities that are below the hazard category 3 nuclear facility threshold as defined in 10 CFR 830, subpart B.

Develop and implement an ISM plan into management and work process planning at all levels per DOE M 450.4-1.

Establish a quality assurance program (QAP). (Refer to 10 CFR 830, subpart A, DOE O 414.1C, and DOE G 413.3-2.) For nuclear facilities, the applicable national consensus standard shall be NQA-1-2008 (edition) and NQA-1a-2009 (addenda).

Identify general S&S requirements for the recommended alternative. (Refer to DOE M 470.4-1 and DOE G 413.3-3.)

Complete a National Environmental Policy Act strategy by issuing a determination (e.g., environmental assessment), as required by DOE O 451.1B. Prepare an environmental compliance strategy, to include a schedule for timely acquisition of required permits and licenses.

For hazard category 1, 2, and 3 nuclear facilities, prepare a safety design strategy (SDS) for projects subject to DOE-STD-1189-2008.

For hazard category 1, 2, and 3 nuclear facilities, conduct an independent project review (IPR) to ensure early integration of safety into the design process. (Refer to DOE G 413.3-9 and DOE-STD-1189-2008.)

Prepare a conceptual safety design report (CSDR) for hazard category 1, 2, and 3 nuclear facilities, including preliminary hazard analysis. For a project involving a major modification of an existing facility, the SDS must address the need for a CSDR, as well as the required preliminary documented safety analysis (PDSA). (Refer to DOE-STD-1189-2008.)

Prepare a conceptual safety validation report (CSVR), with concurrence from the Federal project director (FPD), on the DOE review of the CSDR for Hazard Category 1, 2, and 3 nuclear facilities. (Refer to DOE-STD-1189-2008.)

For nuclear facilities, develop a checkout, testing and commissioning plan in preparation for acceptance and turnover of the structures, systems and components at CD-4. (Refer to DOE-STD-1189-2008.)

CD-2

Establish a performance baseline (PB), reflective of identified and assessed risks and uncertainties, to include the total project cost, CD-4 date, and minimum key performance parameters. The key project milestones and completion dates shall be stated no less specific than month and year. The scope will be stated in quantity, size and other parameters that give shape and form to the project. The funding assumptions upon which the PB is predicated will be clearly documented and approved. (Refer to DOE G 413.3-5.)

Develop a project management plan, if applicable. (Refer to Attachment 1 of DOE O 413.3B.)

Complete a preliminary design.

- Incorporate the guiding principles for Federal leadership in high performance and sustainable buildings per EO 13423, section 2(f), EO 13514, section 2, and sustainable environmental stewardship considerations per DOE O 450.1A into the preliminary design and design review. (Refer to DOE G 413.3-6 and DOE O 430.2B.)
- Conduct a design review of the preliminary design.
- For nuclear facilities, design reviews should include a focus on safety and security systems. Additionally, the code of record shall be placed under configuration control during preliminary design.
- Complete a preliminary design report.

For major system projects where new critical technologies are being developed, conduct a technology readiness assessment and develop a technology maturation plan, as appropriate. It is not required of a project if: 1) the technology was adequately demonstrated previously in one or more separate projects; or 2) the objective of the project is to research scientific principles. (Refer to DOE G 413.3-4.)

Prepare a hazard analysis report for facilities that are below the hazard category 3 nuclear facility threshold as defined in 10 CFR 830, subpart B by updating the PHAR based on new hazards and design information.

Determine that the QAP is acceptable and continues to apply. (Refer to 10 CFR 830, subpart A, DOE O 414.1C, and DOE G 413.3-2.)

Conduct a preliminary security vulnerability assessment, if necessary. (Refer to DOE M 470.4-1 and DOE G 413.3-3.)

Issue the final environmental impact statement (EIS) or environmental assessment and finding of no significant impact, as required by 10 CFR 1021. For an EIS, the appropriate authority shall issue the record of decision after CD-2 is granted, but prior to CD-3 approval. (Refer to DOE O 451.1B.)

For hazard category 1, 2, and 3 nuclear facilities, conduct a technical independent project review (TIPR).The TIPR is required at or near the completion of the preliminary design. The TIPR is not required for non-nuclear facilities. (Refer to DOE G 413.3-9.)

For hazard category 1, 2, and 3 nuclear facilities, update the SDS for projects subject to DOE-STD-1189-2008.

Prepare a preliminary safety design report (PSDR) that updates the CSDR for hazard category 1, 2, and 3 nuclear facilities based on updated hazard analysis and design information. For a project involving a major modification of an existing facility, the SDS must address the need for a PSDR, as well as the required PDSA. (Refer to DOE-STD-1189-2008.)

Prepare a preliminary safety validation report, with concurrence from the FPD, based on a DOE review of the PSDR for hazard category 1, 2, and 3 nuclear facilities. (Refer to DOE-STD-1189-2008.)

CD-3

Approve updated CD-2 project documentation that reflects major changes from final design, the project execution plan, the performance baseline, acquisition strategy, and project data sheet/funding documents for major items of equipment and operating expense funds.

Complete and review the final design or determine that the design is sufficiently mature to start procurement or construction. The FPD will ensure a constructability review is completed as part of the final design.

- For nuclear facilities, the code of record is controlled during final design and construction with a process for reviewing and evaluating new and revised requirements. This will determine their impact on project safety, cost, and schedule before a decision is made to revise the code of record. New or modified requirements are implemented if technical evaluations determine that there is a substantial increase in the overall protection of the worker, public or environment, and that the direct and indirect costs of implementation are justified in view of this increased protection.
- Incorporate the guiding principles for Federal leadership in high performance and sustainable buildings per EO 13423, section 2(f), EO 13514, section 2, and sustainable environmental stewardship considerations per DOE O 450.1A into the final design and the environmental impact report. (Refer to DOE G 413.3-6 and DOE O 430.2B.)

For major system projects where a significant critical technology element modification occurs subsequent to CD-2, conduct a technology readiness assessment, as appropriate. It is not required of a project if: 1) the technology was adequately demonstrated previously in one or more separate projects; or 2) the objective of the project is to research scientific principles. (Refer to DOE G 413.3-4.)

Update the hazard analysis report for facilities that are below the hazard category 3 nuclear facility threshold as defined in 10 CFR 830, subpart B, based on new hazards and design information.

Prior to start of construction, prepare a construction project safety and health plan in accordance with 10 CFR 851, appendix A, section 1(d). This plan must be kept current during construction.

Update the QAP for construction, field design changes, and procurement activities. (Refer to 10 CFR 830, subpart A, DOE O 414.1C, and DOE G 413.3-2.)

Finalize the security vulnerability assessment report, if necessary. (Refer to DOE M 470.4-1 and DOE G 413.3-3.)

For hazard category 1, 2, and 3 nuclear facilities, update the SDS for projects subject to DOE-STD-1189-2008.

Prepare the PDSA that updates the PSDR for newly planned hazard category 1, 2, and 3 nuclear facilities based on updated hazard analysis and design information; also for major

modifications of existing facilities. (Refer to 10 CFR 830, subpart B, and DOE-STD-1189-2008.)

Prepare a safety evaluation report (SER), with concurrence from the FPD, based on review of the PDSA for hazard category 1, 2, and 3 nuclear facilities. (Refer to 10 CFR 830, subpart B.)

CD-4

Verify that key performance parameters and project completion criteria have been met and that mission requirements have been achieved. The FPD will verify and document the scope accomplished, total project cost, key performance parameters met, and the completion date as it relates to the original CD-2 performance baseline and the latest approved baseline change.

Issue a project transition to operations plan that clearly defines the basis for attaining initial operating capability, full operating capability, or project closeout, as applicable. The plan will include documentation, training, interfaces, and draft schedules. (Refer to DOE G 413.3-16.)

For non-nuclear projects, conduct a formal assessment of the project's readiness to operate, as appropriate. Determine the basis for DOE acceptance of the asset and if the facility or area can be occupied from both a regulatory and a work function standpoint. Establish a beneficial occupancy/utilization date for the facility and/or equipment.

Finalize the hazard analysis report for facilities that are below the hazard category 3 threshold as defined in 10 CFR 830, subpart B.

Revise the environmental management system in accordance with DOE O 450.1A, as appropriate.

Conduct an operational readiness review (ORR) or readiness assessment (RA) for hazard category 1, 2, and 3 nuclear facilities in accordance with DOE O 425.1D and DOE-STD-3006-2010.

Prepare the DSA with TSRs for hazard category 1, 2, and 3 nuclear facilities. (Refer to 10 CFR 830, subpart B.)

Prepare an SER based on a review of the DSA and TSR for hazard category 1, 2, and 3 nuclear facilities. (Refer to 10 CFR 830, subpart B.)

For nuclear facilities, the code of record will be included as part of the turnover documentation from a design and construction-phase contractor to the operating-phase contractor; from an operating-phase contractor to the decommissioning-phase contractor; and when a change in contractor occurs during any single life-cycle phase and is maintained under configuration control.

c. Discuss the criticality guidance provided in DOE-STD-1189, Section 7.5, Table 7-2, Appendix H, and Appendix I. Also, discuss the type of evaluations that

should be provided at each critical decision point as identified in DOE-STD-1189, Table 7-1.

The following is taken from DOE-STD-1189-2008, section 7.5. The other referenced areas should be consulted for additional information.

Where there is sufficient fissionable material present, NCS controls can also result in safety significant functional classification of SSCs and, potentially, TSR controls. As a result, the NCS function must be represented on the project team and closely linked to the safety analysis effort from the earliest stages of project development. Criticality safety evaluations must be integrated with the traditional safety analysis techniques to provide a comprehensive safety analysis. DOE has promulgated guidance for performing and documenting CSEs in DOE-STD-3007-2007.

To support design development, it is important to develop fundamental design criteria to address typical criticality safety concerns (e.g., safe geometry) and to incorporate these criteria early in the design process. The purpose of these criteria is to avoid the use of cumbersome and inherently less reliable ACs. An example set of design criteria is provided in table 7.

Table 7. Example of NCS design criteria

Attribute	Criteria
Geometrically safe designs	1. Storage tanks, process piping, containers, etc. shall be designed for conservative enrichment or optimal concentration and reflection for all anticipated nuclides. 2. Designs shall be based on worst-case fire suppression actuation or local pipe breaks. 3. Leaks from solution areas should be anticipated and flooring designed to be compatible with solutions and provide collection capability (prevent long-term migration of fissionable material into sub-flooring materials). 4. Fissionable containing piping shall be spaced to preclude neutron interaction.
Layout processes to support material flow	1. Fissionable solution piping shall be arranged to minimize or eliminate manual transfers. 2. Transfers from safe to non-geometrically safe geometry shall be provided with engineered controls. 3. Avoid any favorable to unfavorable geometry solution transfers. If such need to be made, active design features should be installed to mitigate the potential for a criticality accident due to transfer of fissionable solution to an unfavorable geometry vessel.
System design for holdup minimization	1. Employ vertical tanks to facilitate particulate collection and monitoring. 2. Provide methods to facilitate holdup verification/assay. 3. Locate filtration on exhaust systems as close to main processing loop as possible.
Maximize use of passive design features	1. Utilize positive isolation techniques to prevent unmonitored backflow potential (e.g., air breaks). 2. Avoid common ties between fissionable and non-fissionable systems. 3. Designs must eliminate potential for water ingress into fissionable material processes and containers in the event of fire.
Standardize Equipment	1. Storage racks shall be modular and prevent relocation of fissionable material up to the seismic DBA. 2. Gloveboxes shall be designed to address concerns associated with spills or in-leakage of moderators. 3. All process equipment must withstand the facility seismic DBA.

Source: DOE-STD-1189-2008

One of the most important criticality safety design features is to prevent, by design, natural phenomena initiators for criticality accidents (e.g., seismic and wind). In addition, the fire protection program at design will also drive criticality safety design requirements. The building code, life safety code, national fire codes, and DOE directives almost always require automatic sprinkler protection and firefighting hose capability as well as suitable drainage for nuclear facilities. Unless exemptions and variances have been approved for automatic and manual fire suppression, the criticality calculations must take this moderator into account.

Criticality safety includes human interaction with the potential criticality hazard. Addressing human interaction issues typically results in ACs. Minimizing use of ACs in lieu of more reliable engineered controls should be a focal point for design. One specific

aspect of NCS operations requiring early project definition is emergency response to criticality accidents.

Designs should strive to make a criticality accident a beyond-extremely-unlikely event. If that is not practical, the double contingency principle requires control of two independent parameters (see DOE-STD-3007-2007). Deviations from the double contingency principle must be specifically approved by DOE and typically results in layers of ACs. A singular focus of criticality safety-in-design should be to avoid the need for single parameter control in all processes where a criticality accident is credible.

The following is taken from DOE-STD-1189-2008, appendix H.

DOE O 413.3A requires a CSDR as a part of the approval package for CD-1 approval of the conceptual design. The purpose of the CSDR is to summarize the hazards analysis efforts and safety-in-design decisions incorporated into the conceptual design along with any identified project risks associated with the selected strategies. The DOE review of the CSDR, documented in a CSVR, confirms that the preliminary safety positions adopted during conceptual design constitute an appropriately conservative basis to proceed to preliminary design.

The following is taken from DOE-STD-1189-2008, appendix I.

The PSDR should update the information in the CSDR, if needed (i.e., if the information has changed). In addition, more detailed site information of the type that can affect safety-in-design should be provided (e.g., location of nearby facilities and external hazards, meteorological information for dispersion analyses, seismic and other natural phenomena information).

The PDSA, as opposed to the hazards analysis, demonstrates the adequacy of the design from the safety prospective. As with the design, it is not necessary to show the progression of the design that led to the final choices, only those final choices, and the justification for their adequacy. The PDSA format and content discuss how this information is documented.

Demonstrating safety design adequacy for final design is focused on demonstrating that the safety design requirements specified at the end of preliminary design have been satisfied and describing the mitigated condition for hazards and accidents with the hazard controls applied.

Table 7-1 of DOE-STD-1189-2008 provides a detailed list of actions associated with project life-cycle stages and should be consulted for further information. The type of evaluations associated with each critical decision are discussed above in CS-24b and will not be repeated here. However, the section of table 7-1 that specifies criticality safety actions authorized by critical decision approval has been reproduced (in table 8) below for convenience.

Table 8. Typical actions associated with project life-cycle stages

Actions Authorized by Critical Decision Approval						
Phase / Interface	Mission Need	Conceptual Design	Preliminary Design	Detailed Design	Construction	Resource Requirements and Guidance
Criticality Safety	• Determine criticality potential • Input to Hazard Categorization	• Criticality Control Philosophy • Criticality guidance for Design	• Preliminary CSEs • Updated criticality safety design requirements	• Updated preliminary CSEs • Re-assess criticality limits and controls based on design and operating the process/facility • CSE input to PDSA (Hazard Analysis and TSR derivation)	• Update and issue CSEs • TSRs and operating procedures will incorporate criticality controls, as developed under the guidance of DOE-STD-3007 and DOE G 423.1-1. • Validate NCS controls in field\ • Prepare DSA Ch. 6	• DOE-O-420.1B • DOE-STD-3007-2007 • DOE-G-421.1-1

Source: DOE-STD-1189-2008

25. Criticality safety personnel must demonstrate a familiarity level knowledge of DOE O 425.1C, *Start-up and Restart of Nuclear Facilities*, with respect to nuclear safety issues.

[Note: DOE O 425.1C has been cancelled and replaced by DOE O 425.1D, Verification of Readiness to Start Up or Restart Nuclear Facilities.]

a. Discuss the purpose, scope, and applicability sections of DOE O 425.1C.

The information for KSAs a through c and e is taken from DOE O 425.1D.

The objectives of DOE O 425.1D are to establish the requirements for DOE, including the NNSA, for startup of new nuclear facilities and for the restart of existing nuclear facilities that have been shut down. Nuclear facilities are activities or operations that involve radioactive and/or fissionable materials in such form or quantity that a nuclear hazard potentially exists to the employees or the general public. The requirements specify a readiness review process that must, in all cases, demonstrate that it is safe to start (or restart) the applicable facility. The facility must be started (or restarted) only after documented independent reviews of readiness have been conducted and the approvals specified in this Order have been received. The readiness reviews are not intended to be tools of line management to achieve readiness. Rather, the readiness reviews provide an independent confirmation of readiness to start or restart operations.

This Order is applicable to DOE, including the NNSA. Except for the exclusions in paragraph 3c, this Order applies to all nuclear facilities classified as hazard categories 1, 2, or 3, including NNSA facilities.

Whenever "operations office" appears in this directive, it should be understood to mean "operations office or other office as explicitly stipulated by appropriate lead program secretarial office guidance."

b. Discuss the content of the requirements section of DOE O 425.1C.

Direction and control of requirements pertinent to NNSA facilities must fall under cognizant NNSA management, organizations, and activities, consistent with the NNSA Act. DOE-STD-3006-2010, *Planning and Conducting Readiness Reviews*, provides guidance on approaches and methods approved as acceptable for implementing the requirements of this Order. Other approaches and methods may be used provided they are justified, documented, and approved as being in accordance with the requirements of this Order by the authorization authority for startup or restart.

For NNSA facilities, the term "NNSA line management" is applied wherever the term "DOE line management" or a similar phrase is invoked in conjunction with a requirement or action. Direction and control of requirements pertinent to NNSA facilities, activities, or operations must fall under cognizant NNSA management, organizations, and activities, consistent with the NNSA Act. Definitions of terms used in this Order are found in the definition section of DOE-STD-3006-2010.

The readiness review process must, in all cases, demonstrate there is a reasonable assurance for adequate protection of workers, the public and the environment from adverse consequences from the start (or restart) of a hazard category 1, 2, or 3 nuclear facility, activity, or operation.

Hazard category 1, 2, and 3 nuclear facilities, activities, or operations may be started (or restarted) only after readiness reviews have been conducted and the approvals specified in this Order have been received. Requirements include the following:

- Determining the level of readiness review. DOE line management must evaluate the need to perform a readiness review prior to the startup and restart of hazard category 1, 2, and 3 nuclear facilities, activities, or operations.
- Determining the startup authorization authority (SAA). For nuclear facility, activity, or operation startup or restart actions, the SAA must be determined using the requirements of this Order.
- Startup notification report (SNR). DOE line management procedures concerning the SNR prepared by the contractor in accordance with attachment 1 must address the requirements of this Order.
- Requirements applicable to DOE ORRs. DOE line management must develop a plan of action (POA), which describes the scope of the ORR. The POA must provide a clear discussion of the physical or geographic scope of the ORR and a clear description of the SSCs, individual processes, and programs that are within the scope of the ORR. The POA must also designate the proposed ORR team leader.
- Requirements applicable to DOE RAs. The requirements for performing RAs as described below may be tailored with approval of the program secretarial officer (PSO) and concurrence from the central technical authority (CTA).
- Core requirements. Core requirements verify the readiness of personnel, procedures, programs, and equipment within the scope of the readiness review to

safely start nuclear operations. These core requirements are directly related to the seven guiding principles of ISM.

- DOE field element line management oversight of the process for verifying readiness to start up or restart nuclear facilities. DOE field element line management must review and concur with contractor procedures for implementing the requirements of the CRD.
- DOE headquarters line management oversight of the DOE field process for verifying readiness to startup and restart nuclear facilities. DOE headquarters line management must oversee DOE field processes for verifying readiness to start up and restart nuclear facilities in accordance with DOE O 226.1A, , and DOE-STD-3006.
- Records management program. Requirements for maintenance and disposition of Federal records, such as those pertaining to ORRs or RAs, are provided under the general guidance of DOE O 243.1, *Records Management Program*.
- Implementation of this Order for DOE line management must be completed in 180 days from the issuance of the Order, unless a different schedule is approved by the PSO with concurrence of the CTA.

c. Discuss the responsibilities and authorities section of DOE O 425.1C, with respect to implementation.

DOE and NNSA Line Management

- Establish procedures as necessary to manage the verification of readiness to start up or restart nuclear facilities, activities, or operations in accordance with the requirements of this Order. Forward procedures to the appropriate PSO and CTA as well as the Office of Health, Safety, and Security (HSS) for information.
- Exercise delegation of authority and document all delegations of authority made under the provisions granted by this Order.

Office of Health, Safety and Security

- In coordination with the PSO, perform independent reviews of startup and restart activities as appropriate and provide results of these reviews to the cognizant secretarial officer (CSO) for information.
- Review and comment, as appropriate, on the procedures developed by PSOs, field element offices, and contractors for verifying readiness to start up or restart nuclear facilities, activities, or operations, and provide the results of these reviews to CSOs for information.
- Review and comment as appropriate on SNRs, contractor and DOE POAs, IPs, and ORR or RA final reports for verifying readiness to start up or restart nuclear facilities, activities, or operations, and provide the results of these reviews to CSOs for information.

d. Define the following terms as they relate to DOE Order 425.1C, and nuclear safety:

- **Facility shutdown**
- **Operational Readiness Review (ORR)**
- **ORR implementation plan**
- **ORR scope**
- **Plan-of-action**

- **Prestart finding**
- **Readiness assessment**
- **Unplanned shutdown**

The following definitions are taken from DOE-STD-3006-2010. Terminology titles that have changed from the 2000 revision of this standard are shown in parentheses ( ).

Facility Shutdown (Shutdown)

1) A situation in which a reactor is taken subcritical, either manually or automatically, to a safe shutdown condition; 2) a condition in which a nonreactor nuclear facility, activity, or operation ceases; or 3) a condition in which a programmatic nuclear activity ceases (but the structure containing the activity may remain operational, i.e., not shut down). In a shutdown condition, a facility must still meet all applicable TSRs and environmental, safety, and health requirements.

Operational Readiness Review (ORR) (Readiness Review—RR)

A review conducted to determine readiness to start up or restart a nuclear facility, activity, or operation. There are two types of RRs: ORRs and RAs.

ORR Implementation Plan (Implementation Plan—IP)

The procedural document by which the RR is conducted. This document implements the scope and direction approved in the POA and defines the depth of the review.

ORR Scope (Scope)

The overall magnitude of the RR, as defined by the physical breadth and depth of the facilities and equipment to be started, the breadth of core requirements selected, and the depth of evaluation of these core requirements during the RR. The POA defines the scope of the RR when it establishes the breadth and broadly describes the desired depth.

Plan-of-Action

The document prepared by line management that describes the scope of the RR, the prerequisites to be met to begin the RR, and the proposed team leader for the RR. The contractor and DOE both prepare POAs for their respective RRs (in some instances as specified in the SNR, a DOE RA may not be required). Both contractor and DOE POAs when required are submitted to the startup approval authority for approval. The designated RR team leader(s) develops their IP from the approved POA.

Prestart Finding (Finding)

Nonconformance with a stated requirement that represents either: 1) a systematic failure to establish or implement an adequate program or control; or 2) a significant failure that could result in unacceptable impact on the safety of personnel, the facility, the general public, or the environment during nuclear operations.

Readiness Assessment

A review conducted to determine a facility's readiness to start up or restart when an ORR is not required.

Unplanned Shutdown (Shutdown)

1) A situation in which a reactor is taken subcritical, either manually or automatically, to a safe shutdown condition; 2) a condition in which a nonreactor nuclear facility, activity, or operation ceases; or 3) a condition in which a programmatic nuclear activity ceases (but the structure containing the activity may remain operational (i.e., not shut down). In a shutdown condition, a facility must still meet all applicable TSRs and environmental, safety, and health requirements.

e. Discuss M&O contractor responsibilities for implementing DOE Order 425.1C.

Contractor line management must develop procedures to define the processes that will be used to implement the responsibilities of this CRD. These procedures must be submitted to DOE line management for concurrence. Onsite transportation activities are subject to this Order unless an alternate process is identified with PSO approval and CTA concurrence.

Contractor responsibilities include the following:

- Determining the level of RR. Contractor line management must evaluate the need to perform an ORR or RA prior to the startup and restart of hazard category 1, 2, and 3 nuclear facilities, activities, or operations
- Determining the SAA. For nuclear facility, activity, or operation startup or restart actions, the SAA must be determined using the criteria specified in this Order.
- Contractor procedures must provide for SNRs.
- For ORRs, contractor line management must develop a POA, which describes the scope of the ORR and designates the proposed ORR team leader.
- For RAs, the contractor must use a graded approach to the tenets of ORR requirements specified in this CRD. An RA may be as short and simple as a check list, or may approach the breadth and depth of an ORR
- Core requirements. Core requirements verify the readiness of personnel, procedures, programs, and equipment within the scope of the RR to safely start nuclear operations. These core requirements are directly related to the seven guiding principles of ISM.
- Exemptions and equivalencies. Exemptions and equivalencies may be obtained in accordance with DOE O 251.1C, *Departmental Directives Program*, and DOE O 410.1, *Central Technical Authority Responsibilities Regarding Nuclear Safety Requirements*. CTA (or designee) concurrence is required for both exemptions and equivalencies to this Order for nuclear facilities.
- Records management program. Requirements for maintenance and disposition of Federal records, such as those pertaining to ORRs or RAs, are provided under the general guidance of DOE O 243.1.

f. Discuss the role of Department criticality safety personnel in implementing the requirements of DOE O 425.1C.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety personnel may be asked to assist in any part of the RR scope. With respect to criticality safety, criteria review and approach documents should be developed

based on DOE-STD-1158-2010 and the oversight criteria given in self-assessment for DOE CSPs.

26. Criticality safety personnel must demonstrate a familiarity level knowledge of the following DOE Orders, Technical Standards, Notice and Nuclear Regulatory Commission (NRC) Regulatory Guide:

- **DOE Order 5400.5 Chg 2, *Radiation Protection of the Public and the Environment***
- **DOE-STD-3011-2002, *Guidance for Preparation of Basis for Interim Operation (BIO) Documents***
- **Secretary of Energy Notice (SEN) SEN-35-91, *Nuclear Safety Policy***
- **DOE-STD-3009-94-CN3, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports***
- **DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities***
- **Regulatory guide 3.71, (Rev 1, October 2005) *Nuclear Criticality Safety Standards for Fuels and Material Facilities***
- **DOE STD-5506-2007, *Preparation of Safety Basis Documents for Transuranic (TRU) Waste Facilities***
- **DOE O 410.1, *Central Technical Authority Responsibilities Regarding Nuclear Safety Requirements***
- **DOE O 460.1B, *Packaging and Transportation Safety***

a. Describe the contents, requirements, and relationships between the above Orders, Technical Standards, and SEN.

The following was provided by a team of subject matter experts at the NNSA Service Center.

DOE Order 5400.5 (Chg 2), Radiation Protection of the Public and Environment

The purpose of this Order is to establish standards and requirements for operations of DOE and DOE contractors with respect to protection of members of the public and the environment against undue risk from radiation.

The structure and requirements parallel 10 CFR 835, "Radiation Protection," but extensive guidance on release of radionuclides via environmental pathways is given.

DOE-STD-3011-2002, Guidance for Preparation of Basis for Interim Operation (BIO) Documents

This standard is a parallel to DOE-STD-3009-94 with applicability to facilities that are being deactivated, or in transition to deactivation. It is a safe harbor method for 10 CFR 830, and is the end of facility life analog to the PDSA process. Full deactivation and decommissioning (D&D) may require the application of DOE-STD-3009, depending on the D&D hazard.

Secretary of Energy Notice (SEN) SEN 35-91, Nuclear Safety Policy

This notice includes the concepts now handled by 10 CFR 830, DOE policies 450.4 and 450.5, and DOE Order 5400.5. It is outdated, but not obsolete. The key concepts are:

- Line management responsibility for safety.
- Personal excellence.

- Use of consensus standards when available.
- Continuous improvement.
- DOE and contractor management are responsible for continuously pursuing enhancements to safety—not just complying with a minimal set of requirements.
- DOE activities should add no more than one tenth of one percent (0.1%) to the likelihood of prompt fatality or latent cancer to anyone within a mile of the site boundary.

DOE-STD-3009-94-CN3, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports

This standard is the safe harbor method for developing a DSA. In addition to format and content, there is guidance on how to perform accident analyses. This guidance is sufficient for a criticality analyst to use as a handbook for analysis techniques, although it is not the best or most efficient, and certainly does not address the criticality physics.

DOE-HDBK-3010-94, Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities

The purpose of this document is to provide a compendium and analysis of experimental data from which airborne release fractions and respirable fractions may be derived.

The handbook gives some data for criticality accidents, but repeatedly cautions that the criticality data should be evaluated by criticality specialists and that DOE-STD-3007 can be used. There is useful data presented regarding the fission products generated from criticality excursions in various fissile systems, and methods are given to estimate the releases from a different accident magnitude. Some methods of estimating accident yield are also given, and these may be useful for rough confirmatory calculations. In most cases, there are later and better methods, developed mostly by Barbry in France and Nomura and Okuno in Japan.

Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuels and Material Facilities

This revised regulatory guide provides licensees and applicants with updated guidance concerning criticality safety standards that NRC has endorsed for use with nuclear fuels and material facilities. As such, this guide describes methods the Nuclear Regulatory Commission (NRC) staff considers acceptable for complying with the NRC's regulations in Title 10, Parts 70 and 76, of the Code of Federal Regulations.

This regulatory guide endorses the ANSI/ANS-8 standards with a few minor exceptions, and withdraws other regulatory guides in favor of applying the ANSI/ANS-8 standards.

DOE-STD-5506-2007, Preparation of Safety Basis Documents for Transuranic (TRU) Waste Facilities

This standard provides analytical assumptions and methods, as well as hazard controls to be used when developing safety basis documents for TRU waste facilities in DOE. It also provides supplemental technical information that is specific to TRU waste operations, so that contractors can formulate, implement, and maintain safety bases for TRU waste operations in a consistent manner that is compliant with 10 CFR 830, subpart B, requirements.

DOE O 410.1, Central Technical Authority Responsibilities Regarding Nuclear Safety Requirements

The objectives of this Order are to establish CTA and chief of nuclear safety/chief of defense nuclear safety responsibilities and requirements directed by the Secretary of Energy in the development and issuance of DOE regulations and directives (includes standards) that affect nuclear safety.

- To identify CTA authorities and actions for specific regulations and directives
- To establish related responsibilities and requirements for other departmental elements
- To establish responsibilities and requirements for addressing nuclear safety regulations and directives in contracts

DOE O 460.1B, Packaging and Transportation Safety

[Note: This order has been cancelled by DOE O 460.1C, Packaging and Transportation Safety, from which the following is taken.]

The purpose of this Order is to establish safety requirements for the proper packaging and transportation of DOE, (including NNSA), offsite shipments and onsite transfers of radioactive and other hazardous materials and for modal transportation. These safety requirements include:

Offsite Safety

- Packaging and transportation safety. Each entity subject to this Order must perform packaging and transportation activities in accordance with the Department of Transportation (DOT) requirements of the hazardous materials regulations (49 CFR Parts 171-180).
- Special requirements for radioactive material packagings including
  - use of Type B or fissile materials certified packagings
  - use of DOT International Atomic Energy Agency (IAEA) certified packagings
  - application for NRC or DOT certified packagings
  - application for other Type B or fissile materials certified packagings

- Quality assurance. Each entity that participates in the design, fabrication, procurement, use, or maintenance of a hazardous materials packaging must
  - have an approved and audited QA program
  - report deviations from the applicable requirements in compliance with DOE Order 231.1A
  - report other specific deviations to the headquarters certifying official (HCO) or the NNSA CO within thirty days

- International Shipments. For use of the International Civil Aviation Organization's *Technical Instructions*, the International Maritime Organization's *International Maritime Dangerous Goods (IMDG) Code*, Transport Canada's *Transportation of Dangerous Goods Regulations*, and/or the IAEA's *Regulations for the Safe Transport of Radioactive Material (TS-R-1)* for domestic segments of international transportation by air, vessel, rail, or highway, adherence to 49 CFR Part 171 is required.

Onsite Safety

Onsite transfer of hazardous materials, substances, and wastes must be conducted in accordance with one of the following:

- 49 CFR Parts 171-180 and the Federal Motor Carrier Safety Regulations (49 CFR Parts 350-399), or
- A transportation safety document approved by the head of operations office or field office/site office manager, as appropriate.

Motor Carrier Safety

Each entity that operates a government-owned or commercial motor carrier for transporting hazardous materials must act in accordance with 49 CFR Parts 350-399.

Pipeline Safety

All transportation of hazardous materials by pipeline must be conducted in accordance with 49 CFR Parts 190-193, 195, and 199.

Railroad Safety

All railroad operations must be conducted in accordance with the applicable regulations of the Federal Railroad Administration (49 CFR Parts 200-268).

Special Permits

Any offsite hazardous materials packaging or shipment that is regulated by DOT and is not prepared in accordance with the hazardous materials regulations must be prepared in accordance with a valid DOT special permit.

DOE applications for a DOT special permit must be submitted to the HCO to review, process, and forward to DOT. NNSA applications for a DOT special permit must be submitted to the NNSA CO to review, process, and forward to DOT. Applications must be prepared in accordance with the procedures in 49 CFR 107.105.

Training

Each entity that offers for transportation, transports or transfers hazardous materials, substances and wastes must

- ensure that all personnel who support and/or perform packaging, transfer and transportation operations are appropriately trained and qualified; and
- maintain auditable training records in accordance with approved DOE or NNSA or site-specific records schedule.

Lessons Learned

DOE and NNSA will share packaging and transportation safety successes, problems, and corrective actions with other DOE and NNSA elements and the field through the use of the Department's lessons learned program.

b. Describe the role of criticality safety personnel with respect to the requirements in these Orders, Standards, and SEN.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety personnel may need to assess compliance with many of the requirements given in these orders as they relate to the criteria given in DOE-STD-1158-2010, and will need to assess overall safety performance in compliance with DOE P 450.4.

c. Discuss the DOE criticality safety interest in NRC Regulatory Guide 3.71.

This revised regulatory guide provides licensees and applicants with updated guidance concerning criticality safety standards that NRC has endorsed for use with nuclear fuels and material facilities. As such, this guide describes methods that the NRC staff considers acceptable for complying with the NRC's regulations in 10 CFR 70 and 10 CFR 76.

This regulatory guide endorses the ANSI/ANS-8 standards with a few minor exceptions, and withdraws other regulatory guides in favor of applying the ANSI/ANS-8 standards.

27. Criticality safety personnel must demonstrate a familiarity level knowledge of the following oversight related DOE Orders and Technical Standards:

- **DOE O 224.3, *Audit Resolution and Follow-up***
- **DOE O 224.2, *Auditing of Programs and Operations***
- **DOE O 226.1A, *Implementation of Department of Energy Oversight Policy***
- **DOE P 226.1A, *Department of Energy Oversight Policy***
- **DOE M 470.4-6 Chg. 1, *Nuclear Material Control and Accountability***
- **DOE P 450.4, *Safety Management System Policy***
- **DOE M 450.4-1, *Integrated Safety Management System Manual***
- **DOE-STD-3006-2000, *Planning and Conduct of Operational Readiness Reviews (ORR)***

[Note: DOE O 224.2 and been superseded by DOE O 224.2A and DOE-STD-3006-2000 has been superseded by DOE-STD-3006-2010, *Planning and Conducting Readiness Reviews*.]

a. Describe the contents, requirements, and relationships between the above Orders and Technical Standard.

DOE O 224.3, Audit Resolution and Follow-up

Purpose

- Establish requirements and responsibilities for the timely, efficient, and effective resolution of open Inspector General (IG) and Government Accountability Office (GAO) audit findings and recommendations.
- Assign accountability to senior program element managers for the management of their respective audit resolution processes.
- Establish requirements and processes for reporting status of open audit findings and recommendations.

Requirements

Audit Resolution and Follow-up Program. Audit resolution and follow-up are integral parts of good management and, therefore, are key elements of senior management responsibilities. Prompt and proper corrective actions are implemented to resolve findings, to address recommendations, and to improve the effectiveness and efficiency of DOE operations.

Audit Reports. IG audits and inspections and GAO audits resulting in public reports are management tools used to detect fraud, waste, and abuse in the Department and to promote economy and efficiency in the administration of the Department's programs and operations. Public inspection reports will be treated in accordance with the policies and procedures specified for audit reports herein.

Departmental Audit Report Tracking System (DARTS). Quarterly status reports on an organization's open audit recommendations are submitted to DARTS. This system is maintained by the Office of Management, Budget and Evaluation/Chief Financial Officer and is a tool for tracking/monitoring the Department's progress in resolving audit recommendations.

Departmental Internal Control and Audit Review Council (DICARC). DICARC is a council established to provide senior management oversight of the audit resolution and follow-up program. The council convenes semiannually or when requested by the chairperson.

DOE O 224.2A, Auditing of Programs and Operations

Purpose

To set forth audit requirements and responsibilities for the promotion of economy and efficiency in the administration of, or the prevention or detection of, fraud, waste, and abuse in programs and operations of DOE.

Requirements

The IG will perform all audits of the programs and operations of DOE. This is not intended to supplant the authority of DOE officials to make programmatic reviews of individual programs and enforce regulatory functions.

The IG will conduct audits of the activities of DOE, its contractors and financial assistance recipients that may encompass the following functions (as described by the government auditing standards):

- Financial and compliance
- Economy and efficiency
- Program results

The IG will conduct audits of contractors or financial assistance recipients as requested by other Federal agencies where DOE is the cognizant audit agency or for other reasons.

DOE O 226.1A, Implementation of Department of Energy Oversight Policy

Purpose

This Order provides direction for implementing DOE P 226.1A, which establishes DOE policy for assurance systems and processes established by DOE contractors and oversight programs performed by DOE line management and independent oversight organizations. The objective of the Order is to ensure that contractor assurance systems and DOE oversight programs are comprehensive and integrated for key aspects of operations essential to mission success.

Requirements

This Order establishes the minimum requirements for implementing DOE P 226.1A. This Order does not preclude requirements in existing or new directives from being more rigorous for operations or activities where it has been determined that the risk or hazard necessitates more rigor.

Oversight and assurance processes may identify DOE directives or site-specific requirements that conflict, are unclear, or are incomplete. Deficiencies in DOE requirements will be brought to the attention of the responsible DOE headquarters policy organization (the Office of Primary Interest) for resolution. Deficiencies in site-specific requirement will be brought to the attention of the contracting officer.

All applicable DOE organizations must establish and implement an effective oversight program consistent with DOE P 226.1A, the requirements of this Order, and the applicable attachments to this Order (Attachment 1, Contractor Requirements Document; Attachment 2, DOE Line Management Oversight Processes; and Attachment 3, Independent Oversight Processes).

For activities and programs at Government-owned and government-operated facilities and sites that are not under the cognizance of a DOE field organization, DOE headquarters program offices will establish and implement comparably effective oversight processes consistent with requirements for the contractor assurance system.

DOE P 226.1A, Department of Energy Oversight Policy

Purpose

The purpose of this policy is to establish a Department-wide oversight process to protect the public, workers, environment, and national security assets effectively through continuous improvement.

Requirements

It is DOE policy to protect the public, workers, environment, and national security assets and to perform its operations effectively. To meet this goal, all DOE organizations must implement an assurance system that ensures compliance with applicable requirements, pursues excellence through continuous improvement, provides for timely identification and correction of deficient conditions, and verifies the effectiveness of completed corrective actions. Additionally, DOE oversight programs must determine whether programs, management systems, and assurance systems comply with requirements and are effectively implemented.

It is DOE policy to implement assurance systems and oversight programs that include four essential elements:

1. A comprehensive and rigorous assurance system implemented at all sites.
2. DOE field element line management oversight processes, such as inspections, reviews, surveillances, surveys, operational awareness, and walkthroughs, that evaluate programs and management systems and the validity of the site assurance system.
3. DOE headquarters line management oversight processes that are focused primarily on the DOE field elements. To the extent necessary, DOE headquarters line

management also looks at contractor activities to evaluate the implementation and effectiveness of field element line management oversight.

4. Independent oversight processes that are performed by DOE organizations that do not have line management responsibility for the management of the activity and thus provide an independent perspective for senior management on the effectiveness of programs and activities at all organizational levels (headquarters, field, and contractor).

DOE M 470.4-6 Chg 1, Nuclear Material Control and Accountability

Purpose

To establish a program for the control and accountability of nuclear materials within the DOE, including the NNSA.

Requirements

To prescribe DOE requirements, including those for the NNSA, for nuclear MC&A for DOE-owned and DOE-leased facilities and DOE-owned nuclear materials at other facilities that are exempt from licensing by the NRC.

To effect the policy in DOE P 470.1, *Integrated Safeguards and Security Management (ISSM) Policy*, by integrating nuclear materials control and accountability into DOE operations as determined by line management, and according to sound risk management practices. (DOE P 470.1 is the Department's philosophical approach to the management of the S&S program. A principal objective of the ISSM program is to integrate S&S into management and work practices at all levels, based on program line management's risk management-based decisions, so that missions may be accomplished without security events, such as interruption, disruption or compromise. This approach includes individual responsibility and implementation of the S&S requirements found in this manual.)

DOE P 450.4, Safety Management System Policy

Purpose

Safety management systems provide a formal, organized process whereby people plan, perform, assess, and improve the safe conduct of work. The safety management system is institutionalized through DOE directives and contracts to establish the Department-wide safety management objective, guiding principles, and functions.

The system encompasses all levels of activities and documentation related to safety management throughout the DOE complex.

Requirements

The Department is committed to conducting work efficiently and in a manner that ensures protection of workers, the public and the environment. It is Department policy that safety management systems shall be used to systematically integrate safety into management and work practices at all levels so that missions are accomplished while protecting the public, the worker, and the environment. Direct involvement of workers during the development and implementation of safety management systems is essential for their success.

The DOE safety management system establishes a hierarchy of components to facilitate the orderly development and implementation of safety management throughout the DOE complex. The safety management system consists of six components: 1) the objective, 2) guiding principles, 3) core functions, 4) mechanisms, 5) responsibilities, and 6) implementation. The objective, guiding principles, and core functions of safety management are used consistently in implementing safety management throughout the DOE complex. The mechanisms, responsibilities, and implementation components are established for all work and will vary based on the nature and hazard of the work being performed.

DOE M 450.4-1, Integrated Safety Management System Manual

Purpose

The purpose of this manual is to clearly identify and institutionalize DOE requirements and responsibilities regarding development and implementation of ISM systems within DOE. This manual provides requirements and guidance for DOE and contractors to ensure development and implementation of an effective ISM system that is periodically reviewed and continuously improved.

Requirements

Integrated safety management system descriptions for DOE secretarial offices must be approved by the responsible DOE headquarters secretarial officer.

Secretarial office ISM system descriptions must be consistent with established DOE safety directives, except where exemptions are approved. These ISM system descriptions should follow applicable DOE direction and guidance.

Each ISM system description will be the primary management system description for the particular secretarial office for accomplishing work in a safe and environmentally sound manner, and must be integrated with the corresponding secretarial office QAPS and other relevant safety and management systems, such as emergency management systems. Each secretarial office ISM system must be integrated with the office business processes for work definition and planning, budgeting, authorization, execution, financial management and control, change control, performance measurement, and performance evaluation incorporating lessons learned and continuous improvement. For example, ISM accountabilities and performance should be reflected in employee performance objectives and evaluations. Secretarial office ISM system descriptions may be combined into a single document or a set of documents that also include the associated secretarial office's functions, responsibilities and authorities document, the QAP, and the line oversight program description.

Integrated safety management system descriptions must be reviewed at least annually to determine whether updates are needed. If no changes are needed to maintain ISM system description complete, accurate, and up-to-date, then no annual update is necessary. A statement to this effect should be included in the annual ISM declaration. If changes are needed, these will be approved by the secretarial officer and provided for information to the applicable CTA or applicable senior DOE official.

Secretarial offices must establish and maintain implementing mechanisms, including processes, policies, protocols, procedures, documentation, and training, to translate ISM system expectations into implementation activities and desired human behaviors. These mechanisms must address all active and applicable facility life-cycle phases including design, construction, operation, and decontamination and decommissioning.

The level of rigor in the ISM system descriptions must be consistent with the hazards and complexity of the applicable facilities and activities.

DOE-STD-3006-2000, Planning and Conduct of Operational Readiness Reviews

Purpose

This standard describes acceptable methods and approaches for meeting the RR requirements of DOE O 425.1D.

Requirements

Specifically this standard describes methods and approaches for:

- determining the type of RR that is appropriate to the specific facility startup consistent with the history, hazards, and complexity of the facility being started up or restarted;
- developing the SNR which documents the results of the process for determining the type of readiness;
- conducting the RR, including development of the POA and IP, selection of team members, and performance and documentation of the RR;
- achieving readiness; and,
- providing examples of process deliverables to include a writing guide and suggested processes for achieving readiness.

b. Describe the role of criticality safety personnel with respect to the requirements in these Orders and standard.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety personnel may need to assess compliance with many of the requirements given in these DOE documents, and will need to assess overall safety performance in compliance with DOE P 450.4 and DOE M 450.4-1.

28. Criticality safety personnel must demonstrate a familiarity level knowledge of the Price-Anderson Amendments Act of 1988 and its impact on DOE criticality safety activities.

a. Describe the purpose and scope of the Price-Anderson Amendments Act.

The following is taken from the American Nuclear Society, Price-Anderson Act—Background for Position Statement 54, November 2005.

Purpose

The main purpose of the Price-Anderson Act is to ensure the availability of a large pool of funds (currently about $10 billion) to provide prompt and orderly compensation of

members of the public who incur damages from a nuclear or radiological incident no matter who might be liable.

Scope

The scope of the Act includes

- nuclear incidents in the course of the operation of power reactors
- test and research reactors
- DOE nuclear and radiological facilities
- transportation of nuclear fuel to and from a covered facility

b. Discuss the Act's applicability to the Department criticality safety activities.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety is included in the nuclear safety rule in that it could have significant impact to the immediate worker, and the potential for criticality raises the hazard category to 2. If there is a potential for criticality, the work is by definition nuclear, and the Price-Anderson Amendments Act (PAAA) and the nuclear safety rule apply.

c. Discuss the civil and criminal penalties imposed on the Department, management and operating contractors, and subcontractors as the result of a violation of applicable rules and regulations related to criticality safety.

Civil Penalties

The following is taken from 10 CFR 820.80 and 820.81.

This policy statement sets forth the general framework through which the DOE will seek to ensure compliance with its enforceable nuclear safety regulations and Orders and, in particular, exercise the civil penalty authority provided to DOE in the PAAA of 1988. The policy set forth herein is applicable to violations of DOE nuclear safety requirements by DOE contractors who are indemnified under the Price Anderson Act, 42 U.S.C. 2210(d), and their subcontractors and suppliers.

Any person subject to a penalty under the PAAA shall be subject to a civil penalty in an amount not to exceed $150,000 for each such violation. If any violation is a continuing one, each day of such violation shall constitute a separate violation for the purpose of computing the applicable civil penalty.

Criminal Penalties

The following is taken from 10 CFR 820.70-72.

This subpart provides for the identification of criminal violations of the PAAA or DOE nuclear safety requirements and the referral of such violations to the Department of Justice.

If a person subject to the Act or the DOE nuclear safety requirements has, by act or omission, knowingly and willfully violated, caused to be violated, attempted to violate, or

conspired to violate any section of the PAAA or any applicable DOE nuclear safety requirement, the person shall be subject to criminal sanctions under the PAAA.

If there is reason to believe a criminal violation of the PAAA or the DOE nuclear safety requirements has occurred, DOE may refer the matter to the Attorney General of the United States for investigation or prosecution.

d. Discuss the requirements associated with the topics below, as they are affected by the rule-making aspect of the Price-Anderson Amendments Act:
- **Safety analysis reports**
- **Unreviewed Safety Questions (USQs)**
- **Quality assurance requirements**
- **Defect identification and reporting**
- **Conduct of operations at DOE nuclear facilities**
- **TSR**
- **Training and certification**
- **Maintenance management**
- **Categorization, notification, reporting, and processing of operational occurrences at DOE nuclear facilities**

The following was provided by a team of subject matter experts at the NNSA Service Center.

The items below relate to due diligence in defining, maintaining, and operating within the safety envelope for the facilities and operations. The rules in 10 CFR 830, DOE O 422.1, and the occurrence reporting Order and manual are the mechanisms for defining the risk and maintaining the residual risk at acceptably low levels.

Safety Analysis Reports

The safety analysis report, now the DSA, documents the hazards of a facility or operation, defines the safety and risk envelope, states the hazard controls used to maintain the safety envelope, and identifies the residual risk. Since the operating contractor is indemnified, the documentation of hazard and risk allows DOE to appropriately judge what risk the government is accepting on behalf of the public.

Unreviewed Safety Questions (USQs)

The USQ process is the mechanism to identify any changes to the safety and risk envelope that the DOE has accepted, and is part of the mechanism for keeping the safety basis current. Part of the indemnity requirement is that any substantive change in the residual risk be known and accepted by DOE.

Quality Assurance Requirements

The QA requirements in 10 CFR 830.122, "Quality Assurance Criteria," are primarily aimed at ensuring the items relied on for safety meet the expectations of the safety analysis. This goes beyond SSCs, and includes system design, safety analysis, and items contributing to correct human performance. Human performance items are focused on training and qualifications.

Defect Identification and Reporting

It is important to identify and repair or replace SSCs that, if deficient, could adversely affect public or worker safety. The regulations require that DOE be notified whenever a DOE contractor learns that a facility (including its SSCs) fails to comply with a nuclear safety requirement, contains a defect, or has been supplied with a less than adequate product or service which could result in a substantial safety hazard.

Conduct of Operations at DOE Nuclear Facilities

The purpose of conduct of operations is to ensure an acceptable level of safety by relying on rule based actions rather than expert-based actions. This includes operating procedures, periodic review of procedures and programs, and periodic assessment by the line organization of the effectiveness of directives, plans, and procedures in nuclear operations.

Technical Safety Requirements

Technical safety requirements define the documented safe operating parameters or operating envelope. They are developed as part of the DSA process. Violation of TSRs may be subject to enforcement action if egregious, repetitive, or not reported.

Training and Certification

This (partially) addresses the human element of system performance. In 10 CFR 830, it has a function analogous to operability for SSCs. Its purpose is to ensure that personnel have the skills and ability to perform the operations correctly.

Maintenance Management

This addresses two issues:

1. Maintenance as a safety management program, ensuring the SSCs remain reliable.
2. For facilities that use the method in DOE-STD-1120-2005 or successor document, and the provisions in 29 CFR 1910.120 (or 29 CFR 1926.65 for construction activities) for developing a safety and health program and a site-specific health and safety plan (including elements for emergency response plans, conduct of operations, training and qualifications, and maintenance management) are the core of operational safety.

Categorization, Notification, Reporting, and Processing of Operational Occurrences at DOE Nuclear Facilities

The occurrence reporting system serves two major information analysis functions:

1. Assess local deficiencies for trends or ineffective corrective actions
2. Assess complex-wide deficiencies for operational improvements and lessons learned

e. Discuss the role of Department criticality safety personnel with respect to implementing the requirements of the Price-Anderson Amendments Act in accordance with the following:

- **10 CFR 820, *Procedural Rules for DOE Nuclear Activities***
- **10 CFR 830, *Nuclear Safety Management***
- **10 CFR 835, *Occupational Radiation Protection***
- **DOE STD-1083, *Requesting and Granting Exemptions to Nuclear Safety Rules***

- **Office of Enforcement and Investigation procedure "Enforcement of DOE Nuclear Safety Requirements under Price-Anderson Amendments Act of 1988"**
- **Office of Enforcement and Investigation procedure "Identifying, Reporting, and Tracking Nuclear Safety Noncompliance under Price-Anderson Amendments Act of 1988"**

The following was provided by a team of subject matter experts at the NNSA Service Center.

10 CFR 820, Procedural Rules for DOE Nuclear Activities

Department criticality safety personnel may need to assist in the determination of facts and impact of potential violations. Criticality safety personnel may be involved in investigating precedents to a potential violation, such as whether corrective actions could have reasonably been expected to work, whether the contractor exercised due diligence in monitoring and discovery, and whether important information may have been missed or ignored.

If a contractor reports a potentially unsafe condition or direction, Part 820 brings Part 708 into play, and the issue must be run to ground. The contractor employee would have reasonable expectation that reporting an issue to safety oversight is reporting to a DOE official.

Note that once adjudication is requested, discussions outside the formal context of the proceedings are severely limited.

10 CFR 830, Nuclear Safety Management

Department criticality safety personnel review safety analyses, assist Department safety analysis personnel, and assist in overseeing the criticality safety management programs. With respect to PAAA, the emphasis is on compliance with the safety controls developed, and whether the safety analyses were developed in accordance with the requirements of 10 CFR 830 as the facts were understood at the time of development.

10 CFR 835, Occupational Radiation Protection

Department criticality safety personnel interface with 10 CFR 835 in two main areas. One is the assessment of dose in emergency situations, such as after a criticality accident. Department personnel will primarily be responsible to ensure that contractors have response procedures in place. The second area is dealing with dosimetry after the accident. These areas are usually the province of radiation protection personnel, but criticality personnel may have to deal with these issues, which fall into four major areas:

1. Initial screening of personnel to identify those individuals with high exposure
2. Dosimetry from biological materials such as hair phosphorous and blood sodium
3. Fixed nuclear accident dosimetry
4. Personal nuclear accident dosimetry

Department personnel will primarily be responsible to ensure that contractors have response procedures in place, although emergency assistance may be required.

DOE-STD-1083-95, Requesting and Granting Exemptions to Nuclear Safety Rules

Department criticality safety personnel may be asked to assist site office (or field or operations office) managers in assessing the exemption request(s) for recommendation to the secretarial officer. Department criticality safety personnel may also be called on to advise the secretarial officer on whether an exemption should be granted.

Office of Enforcement and Investigation procedure "Enforcement of DOE Nuclear Safety Requirements under Price-Anderson Amendments Act of 1988"

This procedure (or its replacement) is internal to the Office of Enforcement. Criticality safety personnel may be asked to assist the Office of Enforcement in determinations of fact.

Office of Enforcement and Investigation procedure "Identifying, Reporting, and Tracking Nuclear Safety Noncompliance under Price-Anderson Amendments Act of 1988"

Criticality safety personnel may be asked to assist the Office of Enforcement in determinations of fact. One of the examples of programmatic breakdown deals with lack of timely response to criticality issues.

29. Criticality safety system personnel must demonstrate a familiarity level knowledge of communications (both oral and written) when working or interacting with the contractor, stakeholders, and other internal and external organizations.

The information for the KSAs in this competency statement was provided by a team of subject matter experts at the NNSA Service Center.

a. Identify the various internal and external groups with whom criticality safety personnel must interface in the performance of their duties.

Criticality safety personnel must interface with

- DOE headquarters staff;
- DOE field office staff;
- DOE site office staff;
- ANS nuclear criticality safety division officers and members;
- ANS-8 standards working groups;
- DOE NCS program groups: (e.g., criticality safety coordinating team, criticality safety support group, nuclear data advisory group)
- Local site safety basis group members; and
- Local site contractor management.

b. Describe the media that may be utilized to communicate with these groups.

Communication with these groups may be through both written (letters, e-mail, white papers) and oral (phone and face-to-face, presentations) media.

30. Criticality safety personnel must demonstrate a familiarity level knowledge of nuclear safety-related data and information management requirements in accordance with the requirements of the following DOE Orders:

- **DOE O 200.1A, *Information Technology Management***
- **DOE O 243.1, *Records Management Program***

- **DOE O 414.1C, *Quality Assurance***
- **DOE O 241.1A, *Scientific and Technical Information Management***

a. Describe the authorized disposition requirements for criticality safety-related records in DOE O 200.1A.

As specified in DOE O 200.1A,

- 41 CFR 102-193, sets forth policies and procedures concerning the creation, collection, use, documentation, dissemination, and disposition of records maintained by Federal agencies and the National Archives and Records Administration (NARA).
- 44 U.S.C. chapters 21, 29, 31, and 33, specify the functions and responsibilities for managing Federal records and the procedures that must be followed to obtain approval for their disposition.

b. Describe the requirements for documents and records in DOE O 414.1C.

The following QAP requirements are from management criteria 4, Documents and Records, in DOE O 414.1C:

- Prepare, review, approve, issue, use, and revise documents to prescribe processes, specify requirements, or establish design.
- Specify, prepare, review, approve, and maintain records.

c. Describe the purpose, scope, contents, and requirements in these Orders.

DOE O 200.1A, Information Management Program

The objectives of this Order are:

- Ensure departmental missions and goals, information, information resources, and information technology investment decisions will be made based on programmatic need, using performance-based measures tied to the budget, using sound business practices, and complying with applicable laws and regulations.
- Treat information, information resources, and information technology as corporate assets integrated with programmatic planning and budgeting.
- Provide a framework for managing information, information resources, and information technology investment, which supports the operating elements of the Department in the accomplishment of its missions and functions in an efficient and effective manner and in accordance with departmental policy.

Requirements

Information technology (IT) will be managed consistent with all statutory, regulatory, Office of Management and Budget (OMB) and departmental requirements to fulfill the following requirements:

- Acquisition, use, and management of IT.
  - Information Technology Strategic Planning
    - Maintain the information resources management strategic plan that links IT planning and investment decisions to program mission and goals and establishes Department-wide IT performance goals, objectives, and measures.

    - Conduct IT performance management, including the development of performance measures that are quantitative and outcome-oriented.

- Capital planning and investment control (CPIC).
  - Implement CPIC processes that effectively manage the selection, control, and evaluation of departmental IT investments, ensuring prioritization and sound management, including fulfilling OMB reporting requirements for IT investments, as detailed in OMB Circular A-11 for Exhibits 300 and 53.
  - Ensure that projects and programs are utilizing a systems development lifecycle methodology that effectively manages the development and maintenance of IT systems.

- Enterprise architecture (EA). Oversee a Department-wide EA, which supports mission needs and provides business value through collaboration among departmental elements, including:
  - Compliance with OMB direction and DOE directives.
  - A governance process that promotes integrated business analysis in support of management decision-making.
  - Alignment with the CPIC process.
  - Maintenance of a target EA aligned with departmental and PSO strategic plans.

- Hardware and Software Acquisition.
  - Perform software asset management including the tracking, licensing, and utilization of DOE's software license inventory. Develop and maintain procedures to prevent illegal or inappropriate use of software licenses.
  - Ensure that DOE IT hardware acquisition and replacement practices are consistent with departmental strategic and operational plans and all statutory, regulatory, administrative, and OMB requirements. They will consist of processes which support DOE in making better hardware decisions and enhancing hardware management.
  - Oversee software QAPs to ensure that all software owned or maintained by DOE is subjected to formal QA, pursuant to DOE N 203.1 and DOE O 414.1C.

- IT operations and use.
  - Ensure software acquisition requirements adhere to the 1998 Amendment to Section 508 of the Rehabilitation Act requirements to provide access for people with disabilities.
  - Ensure that OMB and established DOE web policy and website requirements are met and all public websites are available to persons with limited English proficiency, per EO 13166, "Improving Access to Services for People with Limited English Proficiency," and corresponding secretarial memorandum, *Access to Programs and Activities by Persons with Limited English Proficiency*, dated 4-11-06.
  - Promote appropriate personal use of government equipment, consistent with departmental requirements.
  - Promote procedures to prevent illegal or inappropriate use of software licenses.

- Cyber security management. Oversee cyber security policies, procedures, and practices to ensure that they are consistent with OMB and established departmental requirements.
- Spectrum management. Manage spectrum management processes consistent with regulatory guidance, OMB requirements, and the strategic and operational plans of DOE.
- Records management. Manage records management processes consistent with regulatory guidance, OMB requirements, and the strategic and operational plans of DOE.

DOE O 243.1, Records Management Program

The objectives of this Order are to

- set forth requirements and responsibilities for implementing and maintaining a cost-effective records management program throughout DOE;
- provide for
  - awareness that records management is a part of the job of every DOE employee and contractor;
  - adequate and proper documentation of DOE activities, organization, functions, policies, decisions, procedures, and essential transactions;
  - maintenance and use of records supporting DOE activities;
  - proper records disposition; and
  - economy and efficiency in the execution of the DOE records management program.

- comply with the Federal Records Act [Public Law (P.L.) 81-574], as amended, and other legislation;
- protect the legal and financial rights of the government and individuals directly affected by government activities; and
- preserve historical information, thereby enabling DOE and DOE contractors and their successors to retrieve information needed to make informed decisions.

Requirements

The following requirements must be met to ensure a DOE records management program that meets the objectives of this Order.

- Implement a records management program that complies with the requirements for managing records in all formats, including early capture and control throughout their life cycles.
- Create and maintain current file plans/indexes that describe all categories of records created, received, and maintained by personnel in the course of their official duties.
- Preserve and disposition records in accordance with NARA-approved records disposition schedules, as posted on the DOE Office of the Chief Information Officer records management webpages (http://cio.doe.gov/RBManagement/Records/records.html).
- Preserve records placed under a destruction moratorium (freeze) as necessary to support audits, court cases, Freedom of Information Act appeals, or similar obligations.

- Request disposition authority from NARA, through the departmental records officer, for all unscheduled records.
- Store records in a manner that meets the requirements of 36 CFR 1228, subpart K. Unscheduled records are not to be sent offsite for storage at either NARA or commercial facilities.
- Review CPIC proposals and information architecture plans for electronic records management provisions.
- Conduct internal evaluations of records management practices and programs, including the economy of the operation, at least every three years.
- Ensure records management program training is provided for all personnel with records management responsibilities on a regular basis.
- Identify vital records and preserve them to ensure they are maintained, kept current and where appropriate, available in the event of a continuity of operations or catastrophic event.
- Ensure the site exit process includes a requirement for the transfer of custodianship of Federal records to another employee or a records liaison officer when employees leave on a permanent or long-term basis to prevent inadvertent loss, destruction, or alienation of Federal records.

DOE O 414.1C, Quality Assurance

The objectives of this Order are to

- ensure that DOE, including NNSA, products and services meet or exceed customers' expectations;
- achieve QA for all work based upon the principles that
  - quality is assured and maintained through a single, integrated, effective QA program (i.e., management system);
  - management support for planning, organization, resources, direction, and control is essential to QA;
  - performance and quality improvement require thorough, rigorous assessment and corrective action;
  - workers are responsible for achieving and maintaining quality; and
  - environmental, safety, and health risks and impacts associated with work processes can be minimized while maximizing reliability and performance of work products.

- establish quality process requirements to be implemented under a QAP for the control of suspect/counterfeit items, safety issue corrective actions, and safety software.

Requirements

Each DOE organization must develop and implement a QAP that

- implements QA criteria using a graded approach and describing how the criteria and graded approach are applied;
- uses national or international consensus standards where practicable and consistent with contractual or regulatory requirements and identifies the standards used;
- applies additional standards, where practicable and consistent with contractual or regulatory requirements and as necessary to address unique/specific work activities; and

- integrates, where practicable and consistent with contract or regulatory requirements, quality management system requirements.

DOE O 241.1A, Scientific and Technical Information Management

The objectives of this Order are to establish DOE requirements and responsibilities to ensure that scientific and technical information (STI) is identified, processed, disseminated, and preserved in a manner that 1) enables the scientific community and the public to locate and use the unclassified and unlimited STI resulting from DOE's research and related endeavors, and 2) ensures access to classified and sensitive unclassified STI is protected according to legal or departmental requirements.

Requirements

DOE will make STI broadly available, within applicable laws and departmental requirements, to accomplish mission objectives and strategic goals, promote scientific advancement, satisfy statutory protection and public dissemination requirements, and ensure a fair return on departmental and taxpayer investment.

DOE programs that fund activities that generate STI must emphasize it as a key outcome and use a "best business practices" approach to life-cycle information management to ensure that STI products are planned, budgeted, produced, processed, disseminated, stored, and disposed of (if appropriate) in cost-effective ways that enable full use of such information by all customer segments, including the research and development community, U.S. industry, and the general public.

Each DOE element covered under this directive must have a formal STI point of contact (e.g., a technical information officer) to participate in the development and implementation of DOE's STI program.

The STI coordinating group (STICG) will serve as an advisory body concerning cross-cutting STI issues. The STICG will be comprised of representatives of headquarters' elements that either fund research or set policies and practices affecting STI and representatives from the field and contractors.

DOE programs must ensure that STI, including scientific and technical computer software, is reviewed as appropriate for sensitivity and that appropriate announcement and availability restrictions are applied according to statutory and/or other departmental requirements.

DOE programs must conduct formal STI performance reviews as appropriate as part of their program performance review process.

DOE will use best business practices to facilitate cost-effective management and availability of DOE's STI.

DOE's central STI coordinating office will manage an international STI exchange activity to ensure U.S. access to international scientific and technical advances and market information.

All DOE elements must make the results of scientific and technical endeavors broadly available in useful and acceptable electronic formats, including announcing such STI results to the central coordinating office.

Departmental elements must share STI within the Department to optimize resources, minimize costly duplication, and ensure maximum program advancement.

DOE's central STI coordinating office will support program-specific STI management efforts by offering certain special support services upon request to DOE organizations that provide funding to cover incremental costs of the requested service(s).

d. Discuss the applicability of the above Orders to the Department criticality safety activities and processes.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality oversight procedures and directions should identify what materials from criticality oversight are records, and interface with the appropriate records management function.

e. Discuss the role of the Department criticality safety personnel in implementing the requirements of these Orders.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety personnel generate records. These records must be archived appropriately.

31. Criticality safety personnel must demonstrate a familiarity level knowledge of the following DOE safeguards, security, and nuclear material accountability Orders for nuclear safety-related issues:

- **DOE O 452.6, *Nuclear Weapon Surety Interface with the Department of Defense***
- **DOE O 470.4A, *Safeguards and Security Program***
- **DOE P 470.1, *Integrated Safeguards and Security Management (ISSM) Policy***
- **DOE M 470.4-1, Chg 1, *Safeguards and Security Program Planning and Management***
- **DOE M 470.4-2, Chg 1, *Physical Protection***
- **DOE M 470.4-6, Chg 1, *Nuclear Material Control and Accountability***
- **DOE O 471.1A, *Identification and Protection of Unclassified Controlled Nuclear Information***
- **DOE O 475.2, *Identifying Classified Information***
- **DOE M 470.4-4A, *Information Security Manual***
- **DOE Order 5660.1B, *Management of Nuclear Materials***

a. Describe the purpose, scope, contents, and requirements of these Orders.

DOE O 452.6, Nuclear Weapon Surety Interface with the Department of Defense

Objectives

To establish DOE and NNSA requirements and responsibilities for addressing joint nuclear weapon and nuclear weapon system surety activities in conjunction with the Department of Defense (DoD).

To establish and implement a systematic process to ensure that nuclear weapon surety is adequately addressed throughout all phases of each nuclear weapons life cycle.

To provide support to the DoD during the development, staffing, and implementation of safety rules that govern all nuclear weapon system operations throughout the stockpile-to-target sequence.

Requirements

NNSA in conjunction with DoD has an obligation to protect public health, safety, and the environment from potential adverse consequences of nuclear weapon operations. To ensure dual-agency judgment and responsibility, nuclear weapon system safety, security, and use control (surety) will be evaluated continually throughout the entirety of each nuclear weapon system's life cycle.

Nuclear weapon system surety will include a combination of ACs and design measures sufficient to prevent deliberate unauthorized nuclear detonation and to minimize the possibility of deliberate unauthorized acts that could lead to nuclear detonation.

Nuclear weapon system safety will include design features, safety rules, procedures, accident prevention/mitigation measures, or other controls used collectively or individually to reduce the likelihood, severity, or consequences of an accident.

DOE O 470.4A, Safeguards and Security Program

Purpose

To establish responsibilities for the DOE S&S program and the managerial framework for implementing DOE P 470.1. The requirements identified in this Order and its topical manuals are based on national policy promulgated in laws, regulations, and Eos to prevent unacceptable adverse impacts on national security, the health and safety of DOE and contractor employees, the public, or the environment.

Requirements

Safeguards and security programs must be developed and maintained that incorporate the responsibilities contained in this Order and ensure that the directives listed in the Order are effectively implemented.

All delegations of responsibilities must be documented in writing.

Interfaces and necessary interactions between S&S programs and other disciplines such as safety, emergency management, counterintelligence, facility operations, cyber system operations, and business/budget operations including property management must be

identified and clearly defined. The interfaces must be maintained throughout the life cycle of protective measures to ensure that S&S planning and operations work together effectively with these disciplines.

DOE P 470.1, Integrated Safeguards and Security Management (ISSM) Policy

Purpose

Safeguards and security management systems provide a formal, organized process for planning, performing, assessing, and improving the secure conduct of work according to risk-based protection strategies. These systems are institutionalized DOE directives and contracts. The purpose of this policy is to formalize an ISSM framework.

The ISSM system framework encompasses all levels of activities and documentation related to S&S management throughout the DOE complex.

Policy

The Department is committed to conducting work efficiently and securely. It is Department policy that the ISSM framework shall be used to systematically integrate S&S into management and work practices at all levels so that missions are accomplished securely. Direct involvement of all personnel during the development and implementation of an ISSM framework is essential for success.

The ISSM framework will be implemented through programmatic directives and other related directives.

The ISSM framework establishes a hierarchy of components. To facilitate the orderly development and implementation of S&S management throughout the DOE complex, the ISSM framework consists of six components:

1. The objective
2. Guiding principles
3. Core functions
4. Mechanisms
5. Responsibilities
6. Implementation

DOE M 470.4-1, chg 1, Safeguards and Security Program Planning and Management

Purpose

To establish program planning and management requirements for the Department's S&S program.

Requirements

The following are essential elements for planning S&S programs.

- Safeguards and security philosophy. Safeguards and security interests and activities must be protected from theft, diversion, terrorist attack, industrial sabotage, radiological sabotage, chemical sabotage, biological sabotage, espionage, unauthorized access, compromise, and other acts that may have an adverse impact on national security; the environment; or pose significant danger to the health and safety of DOE Federal and contractor employees or the public. DOE

protective forces that protect category I quantities of SNM; credible rollup of SNM to a category I quantity; or those facilities that meet or exceed the threat level 2 criteria specified in DOE O 470.3B*, Design Basis Threat Policy*, for chemical, radiological, or biological thresholds, must employ the DOE tactical doctrine contained in appendix 2 of section A.

- Safeguards and security management plan. This plan must provide a description of the implementation of S&S policy and provide detailed information on the assignment of roles, responsibilities, and authorities, as well as the development of budgets and allocation of resources.
- Safeguards and security program operations. Actions must be taken to ensure an acceptable S&S program, including curtailment or suspension of operations when such operations would result in an immediate and unacceptable impact to national security, the environment, or the health and safety of the public or employees.
- Graded protection. The Department recognizes that risks must be accepted however, an acceptable level of risk must be determined based on evaluation of a variety of facility-specific goals and considerations. By a graded approach, the Department intends that the highest level of protection be given to security interests and activities whose loss, theft, compromise, and/or unauthorized use would seriously affect the national security, the environment, departmental programs, and/or the health and safety of the public or employees. Protection of other interests and activities must be graded accordingly.
- Risk management. Safeguards and security programs must be based on the results of vulnerability and risk assessments, the results of which are used to design and provide graded protection in accordance with an asset's importance or the impact of its loss, destruction, or misuse. The results of the assessments, to include the determination of system effectiveness, are one of the key considerations the manager must evaluate when establishing the level of risk. For example, if it is determined that there is high risk that is not being mitigated by compensatory measures, reporting must be made to the Secretary of Energy or the Deputy Secretary who can accept high risk. Cognizant under secretaries can accept moderate risk.
- A performance assurance program must be developed to validate the performance of all essential S&S protection elements.
- Initial surveys must be conducted at facilities where there will be a facility clearance established for a facility with an importance rating of: A, B, C, or PP. Survey activities must be comprehensive and result in a satisfactory composite rating prior to a facility clearance being granted.
- Periodic surveys are conducted for all facilities and must cover all applicable topics to ensure survey program objectives are met. The periodic survey may be composed of multiple special survey reports, providing all the requirements of this section are met. Integration of internal and external reports including QA, property appraisals, performance assurance, and other evaluation reports may be used to augment the requirement for a periodic survey.

DOE M 470.4-2, chg 1, Physical Protection

Purpose

This manual establishes requirements for the physical protection of S&S interests.

Requirements

A facility must not possess, receive, process, transport, or store nuclear weapons or SNM until that facility has been cleared commensurate with DOE M 470.4-1.

An integrated system of positive measures must be developed and implemented to protect category I and II quantities of SNM and nuclear weapons. Protection measures must address physical protection strategies of denial and containment as well as recapture, recovery, and/or pursuit.

Physical protection for each category of SNM must consider the following factors: quantities, chemical forms, and isotopic composition purities; ease of separation, accessibility, concealment, portability; radioactivity; and self-protecting features.

The protection of nuclear material production, reactors, and fuel must be commensurate with the category of SNM.

SNM, parts, or explosives that are classified must receive the physical protection required by the highest level of classification or category of SNM, whichever is the more stringent.

Specific physical protection measures and protective force response capabilities must be described in a site S&S plan or site security plan.

The protection afforded SNM must be graded, according to the nuclear material safeguards category and attractiveness level, and reflect the specific nature of the nuclear weapons or SNM at each site.

DOE M 470.4-6, chg 1, Nuclear Material Control and Accountability

Purpose

To establish a program for the control and accountability of nuclear materials in DOE.

Requirements

Material control and accountability requirements for source and other nuclear materials.

- Separated ^{237}neptunium and separated americium (^{241}Am and ^{243}Am) must be protected, controlled, and accounted for as if they were SNM. For the purposes of this manual, separated ^{237}neptunium and separated americium, refer to the recovered or product material generated from chemical and processing operations on the target/source material.
- Tritium is a nuclear material of strategic importance; therefore, a graded safeguards programs for tritium must be implemented according to the following categorizations.
    - Category III. Weapons or test components containing reportable quantities of tritium, deuterium-tritium mixtures, or metal tritides that can be easily decomposed to tritium gas, containing greater than 50 grams of tritium (isotope) with a tritium isotopic fraction of 20 percent or greater.
    - Category IV. All other reportable quantities, isotopic fractions, types, and forms of tritium.

- Excluding tritium, separated ^{237}neptunium, and separated americium (^{241}Am and ^{243}Am), source and other nuclear materials listed in table I-1, Nuclear Materials, are exempt from the requirements of this manual except for the following:
  - An MC&A program must be established and maintained for these materials based on the strategic and monetary value of the materials.
  - Data fields used in the materials accounting system must be consistent with table I-1, Nuclear Materials.
  - Nuclear materials inventories and transactions must be documented in the nuclear materials accounting at a level specified by the DOE cognizant security authority. At a minimum, all reporting identification symbol (RIS)-level inventories and transactions must be documented by the system.
  - RIS level transactions and inventories must be reported to nuclear materials management and safeguards system (NMMSS) according to section B of this manual. Transactions and inventories for berkelium are excluded from this requirement and do not need to be reported to NMMSS; however, accounting for berkelium will be maintained at the facility level.
  - When these materials are potential substitution materials for SNM and are collocated with SNM, the requirements of section A, II, 3.a. (2), physical inventory frequencies, of this manual apply.
  - The frequency and manner of conducting physical inventories must be approved by the DOE cognizant security authority and documented in the site/facility MC&A plan.
  - Other MC&A requirements are to be determined by the DOE cognizant security authority and documented in the site/facility MC&A plan or other MC&A program documents.

Loss Detection Evaluation, Performance Testing, and Performance Requirements

- Loss detection evaluation. An assessment program for identifying and evaluating facility capability to detect the loss of category I quantities of SNM must be developed for each category I facility. Potential targets must include all category I and any other areas for which a credible scenario for unauthorized accumulation of a category I quantity of SNM have been identified. Vulnerability assessments must be approved by the DOE cognizant security authority and must be reviewed and updated when there are system changes or new information indicates a potentially significant change in the risk of unauthorized removal of SNM. Results of the reviews, including changes in the vulnerability assessments, must be reflected in the vulnerability analyses reports.
- Performance testing. MC&A performance testing programs must be developed and documented to support and verify loss detection capability and system effectiveness. The scope and intent of performance testing must be based on the graded safeguards concept (i.e., the testing program demonstrates greater testing for higher category facilities than for lower category facilities, with category I defined as highest and category IV as lowest).
- MC&A performance requirements. The performance of the selected system elements must be validated on a frequency documented in the MC&A plan. If system elements fail to meet the performance requirements, a CAP must be developed and, where necessary, compensatory measures must be taken.

DOE O 471.1A, Identification and Protection of Unclassified Controlled Nuclear Information.

Objectives

To prevent the unauthorized dissemination of UCNI.

Requirements

Any person who originates or possesses unclassified matter that may contain unclassified controlled nuclear information (UCNI) must send the unclassified matter to a reviewing official.

A reviewing official determines whether matter contains UCNI and marks or authorizes it to be marked as required in DOE M 471.1-1, chapter I, Part C.

A denying official determines whether matter requested under statute or Executive Order is exempt from release based on a reviewing official's recommendation and applicable UCNI guidelines.

Access to UCNI must be provided to only those individuals authorized for routine or special access.

Physical protection requirements for UCNI are contained in DOE M 471.1-1, chapter II.

DOE O 475.2, Identifying Classified Information

Purpose

To establish the program to identify information classified under the Atomic Energy Act or EO 12958, as amended so that it can be protected against unauthorized dissemination.

Requirements

Each field element that has access or oversees an entity that has access to classified information, documents, or material must have a classification officer. This requirement may be fulfilled by a classification officer from within the field element or from outside the field element. If this requirement is fulfilled by a classification officer from outside the field element, a written agreement for support is required, and the field element receiving the support must have a classification coordinator who monitors the work of the outside classification officer. If the size and/or complexity of a field element's program have reached the point where a classification officer from outside of the field element can no longer effectively perform the classification officer's functions, then the field element must designate a classification officer from within the organization.

If the classification officer for a field element requires assistance from another classification officer, then a written agreement for support is required that describes specifically what support is to be provided.

Each headquarters element that has employees who are authorized access to classified information, documents, or material must have either a headquarters classification representative or a headquarters classification liaison. A headquarters classification representative is required if any employees within the element are derivative classifiers. If

no employees are derivative classifiers, then a headquarters classification liaison is required.

The necessary resources and support to meet the requirements specified in this Order and DOE M 475.1-1B, *Manual for Identifying Classified Information*, must be provided by organizations that have access to classified information, documents, or material.

A sufficient number and distribution of original classifiers, derivative classifiers, and derivative declassifiers must be maintained within each DOE and NNSA headquarters and field element so as to not unduly interfere with or delay the work of the headquarters or field element.

Classification guidance must be up to date, and derivative classifiers and derivative declassifiers must have access to appropriate guidance.

Prior to the commencement of non-DOE-funded work in a classified subject area, the organization performing the work must have one of the following:

- Classification guidance covering the work to be performed or
- A written statement from the organization funding the work that the work does not involve classified activities unless the funding organization is not authorized access to classified information. In that case, no written statement is required

Information, documents, and material must be correctly classified, declassified, downgraded, or upgraded by specific individuals who have been granted such authority.

Incidents of security concern resulting from the misclassification of information, documents, or material must be reported as required in DOE M 470.4-1.

Information, documents, and material are prohibited from being classified to

- conceal violations of law, inefficiency, or administrative error;
- prevent embarrassment to a person, organization, or agency;
- restrain competition; or
- prevent or delay the release of information that does not require protection in the interest of the national security.

The performance contract or other system used to rate personnel performance must include the management of classified information as a critical element or item to be evaluated in the rating of original classifiers and all other personnel whose duties significantly involve the creation of classified information. Such performance systems include but are not limited to the annual personnel performance rating system or personnel evaluations based upon the ISSM system or an organization's internal self assessment system.

Training programs and awareness briefings must be conducted to ensure that employees serving as classification officials or working in classified subject areas are aware of their classification-related responsibilities.

The implementation of the requirements contained in this Order and DOE M 475.1-1B must be periodically evaluated to ensure that they are being met.

DOE Order 5660.1B, Management of Nuclear Materials
This Order has been canceled.

b. Discuss the applicability of the above Orders to the Department criticality safety activities and processes.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety personnel evaluate the criticality safety of a nuclear weapon in normal and abnormal environments to document the intrinsic safety of the design.

Safety is paramount when nuclear materials are involved. Conducting CSEs for process operations requires use of a credible experimental database.

The stockpile stewardship management mission involves working with nuclear materials in different concentrations.

c. Discuss the role of the Department criticality safety personnel in implementing the requirements of these Orders.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety personnel perform technical reviews and provide recommendations on CSP documents. Criticality safety personnel also review and comment on a wide variety of operating contractor documents.

32. Criticality safety personnel must demonstrate a working level knowledge of the DOE/facility contract provisions necessary to provide oversight of a contractor's operations.

The information for all of the KSAs in this competency statement—except where specified otherwise—was provided by a team of subject matter experts at the NNSA Service Center.

a. Describe the role of criticality safety personnel in contractor oversight.

The primary goal of the department criticality safety personnel is to ensure that the contractor maintains an effective CSP. Criticality safety personnel oversee contractor criticality programs in three areas to do a comprehensive performance based oversight.

1. The products observed are the process evaluations for criticality safety, often called CSEs. Criticality safety personnel review these items for technical accuracy, correct identification of hazards, and clarity, using DOE STD 1134-99 and previous experience and knowledge. This review may include accident sequence analysis and confirmatory calculations.
2. The process observed, includes the process of preparing CSEs and the process of communicating and implementing the controls developed from the CSEs in the handling of fissile material. Most of this is done from documentation.

3. The performance observed includes the implementation of CSE controls by operations, the investigation and correction of any deficiencies, the interaction of the criticality safety staff with operations, and the ownership of safety by the operating organization.

Criticality safety personnel establish performance measures with the contractor to measure and monitor the ongoing effectiveness of the site CSP.

Criticality safety personnel should also be involved at appropriate points in the budget process to ensure that the contractor has sufficient personnel and materiel to maintain and improve the program (in an ongoing facility) or maintain the program at the appropriate level in a facility being closed.

b. Compare and contrast the following:
- **DOE's expectations of a M&O contractor**
- **An M&O contractor's expectations of the DOE**

DOE's Expectations of an M&O Contractor

As stated in 10 CFR 830, subpart B, the contractor responsible for a DOE nuclear facility is required to analyze the facility, the work to be performed, and the associated hazards and to identify the conditions, safe boundaries, and hazard controls necessary to protect workers, the public and the environment from adverse consequences.

With respect to criticality safety, the contractors are expected to follow the ANSI/ANS 8 series of standards at a minimum, monitor their program, and communicate any program or safety issues to DOE.

A Management and Operating Contractor's Expectations of the Department of Energy

The following is taken from Defense Acquisition University, CON 214, Business Decisions for Contracting course.

Some companies, as a matter of corporate policy, do not do business with the government or certain government agencies. Other companies have quit doing business with the government as a result of poor government business practices. Quality contractors affected by abusive demands beyond the contract requirements, late payments, a low bid environment not based on value, etc. can find many business opportunities with other Federal agencies and in the commercial market place.

Each party can demonstrate its commitment to maintaining and strengthening the relationship by
- enhancing cooperation between the parties;
- providing opportunities and benefits that are better than those available from other sources;
- maintaining corporate values that are compatible with those of the other party;
- providing honest performance feedback, especially the negative, to the other party so that can take appropriate action quickly;
- sharing information that will enhance the other party's market knowledge; and
- upholding the implicit and explicit promises made.

c. Identify the key elements and features of an effective DOE and M&O contractor relationship.

The following is taken from Defense Acquisition University, CON 214, Business Decisions for Contracting course.

Creating and maintaining positive, mission-focused business relationships between the government and contractors plays a critical role in supporting the customer's contract requirements.

The most fundamental characteristics of a strong relationship are trust in the other party and commitment to the relationship. Trust can help both entities shape more flexible contract strategies and create a relationship based on value. When both parties are committed to the relationship they will put forth extraordinary efforts to ensure that the relationship survives.

The following are some proven strategies that can help build a successful business relationship:

- Structure the relationship in a way that maximizes the chance of a successful win-win partnership. Traditional contracting processes may lack the collaboration required to mutually achieve program goals necessary to produce a win-win partnership.
- Ensure that there is a clear understanding of the goals and motivations of all parties so they can be aligned to the maximum extent possible.
- Ensure that all parties work to create a cooperative atmosphere. This is critical in the early stages of the relationship.
- Develop a shared ability to find mutually beneficial solutions for achieving goals.
- Openly discuss differences, such as, cultural, structural, financial, contractual, programmatic, political, etc., to provide insight into each other's motivations and to develop mutual understandings.
- Balance the risks and rewards available to the parties.

All parties need to plan, execute, and deliver on their respective responsibilities in order to achieve the individual and shared goals and objectives.

d. Describe the responsibility criticality safety personnel have associated with contractor compliance under the Price-Anderson Amendments Act.

If criticality safety personnel become aware of malfeasance in the criticality safety area by the contractor, assistance from the office of enforcement may be needed. Contractor line management occasionally has to be recalibrated to forestall problem situations. Criticality safety personnel are more likely to become involved in the investigation of PAAA issues as assistants or advisors to line management or the Office of Enforcement in regards to the impact of particular events or conditions.

e. Describe the role of criticality safety personnel in the performance measure process.

Criticality safety personnel provide input to the performance measure process through the site office line management and contracting officers. The criticality safety related performance measures should be aimed at where the program needs to improve. Performance measures should be negotiated with the contractor and should be amenable to frequent measurement.

f. Explain the responsibilities of criticality safety personnel for DOE O 442.1A, *Department of Energy Employee Concerns Program*, and the identification, reporting, reviewing, and documentation of employee concerns.

If contractor or DOE personnel bring up a criticality safety concern, it may require action by the employee concerns program. If a contractor employee brings a criticality safety concern to DOE NCS personnel, all the alarm bells should go off, and the appropriate line oversight notified as quickly and accurately as possible. (This should be the site office unless they are part of the allegation.)

Nuclear criticality safety personnel are most likely to get involved in the employee concerns program through requests for assistance with an employee concern regarding some aspect of criticality safety or the implementation of criticality safety requirements. In this case the role is to advise and assist the organization or person doing the investigation. This may include review of the concern (and associated data) for actual criticality safety impact, and the documentation and reporting of the results.

33. Criticality safety personnel must demonstrate a familiarity level knowledge of terminology used in nuclear safety analysis.

a. Define the following accident related terms:
- **Accident**
- **Safety basis**
- **Beyond design basis accident**
- **Design basis**
- **Design basis accidents**
- **Evaluation guidelines**

Except where specified otherwise, the information for KSAs a-c, g, and h is taken from DOE-STD-3009-94.

Accident

An accident is an unplanned event or sequence of events that result in undesirable consequences.

Safety Basis

The DSA and hazard controls that provide reasonable assurance that a DOE nuclear facility can be operated safely in a manner that adequately protects workers, the public, and the environment.

Beyond Design Basis Accident

A beyond-design-basis accident is an accident of the same type as a design basis accident, but which is defined by parameters that exceed in severity the parameters defined for the design basis accident. The same correlation applies to beyond-derivative-design basis accidents with regard to derivative-design basis accidents.

Design Basis

The set of requirements that bound the design of SSCs within the facility. These design requirements include consideration of safety, plant availability, efficiency, reliability, and maintainability. Some aspects of the design basis are important to safety, although others are not.

Design Basis Accident

The limited set of unique and representative accidents to be assessed further in accident analysis. Unique accidents are those with sufficiently high-risk estimates that individual examination is needed (e.g., a single fire whose specific parameters result in approaching the evaluation guideline). Representative accidents bound a number of similar accidents of lesser risk (e.g., the worst fire for a number of similar fires). Representative accidents are examined to the extent they are not bounded by unique accidents. In any case, at least one bounding accident from each of the major types determined from the hazard analysis (e.g., fire, explosion, spill, etc.) should be selected unless the bounding consequences are low (minor on-site and negligible offsite impact on people or the environs.) Accidents are identified and listed by accident category (i.e., internally and externally initiated) and type.

Evaluation Guidelines

The radioactive material dose value against which the safety analysis evaluates. The evaluation guideline is established for the purpose of identifying and evaluating safety-class SSCs. Onsite evaluation guidelines are not required for adequate documentation of a safety basis utilizing the overall process DOE-STD-3009-94.

b. Define the following hazard related terms:

- **Hazard**
- **Hazard classification**
- **Hazard category 1**
- **Hazard category 2**
- **Hazard category 3**
- **Hazardous material**

Hazard

A hazard is the existence of a source of danger which could cause illness, injury, or death to personnel or damage to a facility or the environment.

Hazard Classification

DOE has established a process to classify facilities into hazard categories. DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports,* contains the flow charts which, when followed, determine the hazard classification.

Hazard Category 1

The classification of hazard category 1 is designated when a hazard analysis shows the potential for significant offsite consequences. This classification is primarily reserved for reactors capable of operating at a steady state power level greater than 20 megawatts (class A reactor). A facility may also be classified as hazard category 1 if the PSO deems it necessary. This designation would be due to the level of offsite emergency planning required to permit operation.

Hazard Category 2

The classification of hazard category 2 is designated when a hazard analysis shows the potential for significant onsite consequences. A release of materials from a hazard category 2 facility would result in exposures greater than that of 1 rem at 100 meters. Thus, emergency planning requirements are very important. The potential for a nuclear criticality accident at a facility will also earn it a hazard category 2 classification.

Hazard Category 3

The classification of hazard category 3 is designated when a hazard analysis shows the potential for only significant localized consequences. Minimum threshold values for a hazard category 3 facility would result in exposures of less than 10 rems at 30 meters from an unmitigated release of the total inventory of radionuclides. The consequences of a localized release of materials would be in excess of state and Federal reporting requirements.

Hazardous Material

Any solid, liquid, or gaseous material that is toxic, explosive, flammable, corrosive, or otherwise physically or biologically threatening to health

c. Define the following safety limit related terms:

- **Limiting conditions for operations**
- **Limiting control settings**
- **Risk**
- **Safety analysis**
- **Safety basis**
- **Safety limits**
- **Criticality safety limits**

Limiting Conditions for Operation

The limits that represent the lowest functional capability or performance level of safety SSCs required for safe operations.

Limiting Control Settings

The settings on safety systems that control process variables to prevent exceeding an SL.

Risk

Risk is a combined effect of the probability of the occurrence of an undesired event and the magnitude of the event.

Safety Analysis

A documented process: 1) to provide systematic identification of hazards within a given DOE operation; 2) to describe and analyze the adequacy of the measures taken to eliminate, control, or mitigate identified hazards; and 3) to analyze and evaluate potential accidents and their associated risks.

Safety Basis

The safety basis is comprised of the DSA and hazard controls that provide reasonable assurance that a DOE nuclear facility can be operated safely in a manner that adequately protects workers, the public, and the environment.

Safety Limits

The limits on process variables associated with those safety class physical barriers, generally passive, that are necessary for the intended facility function and that are required to guard against the uncontrolled release of radioactive materials (10 CFR 830).

Criticality Safety Limits

The controlled parameters and the associated maximum or minimum values on which NCS depends (ANSI/ANS 8.19).

d. Differentiate between the following categories of individuals who might be affected by an accident at a Department nuclear facility:

- **Off-site individual**
- **On-site individual**
- **Public**
- **Worker, including collocated worker**

The following is taken from Defense Nuclear Facilities Safety Board Technical Report number 20 unless specified otherwise.

Off-Site Individual (Off-Site Personnel)

Individuals who live or work at locations beyond the boundary of the site, or are temporarily in such places, and are seldom, if ever, on site. The site boundary is defined below.

On-Site Individual

This category consists of:

- Other onsite worker personnel—persons at work in support of operations at the site as employees of incidental contractors, such as for construction activities, or employees of privatized facilities on the site, or of leasers of DOE-owned space. These are workers for organizations not contributing directly to the missions of the site. There may be workers whose duties involve them intimately in activities with nuclear hazards at privatized or leased facilities.
- Transient onsite personnel—people engaged in activities such as delivery of equipment and supplies and collection of material to be transported offsite. They spend only a small fraction of their time onsite in any year.

Site Boundary

The following is taken from DOE-STD-3009-94.

A well-marked boundary of the property over which the owner and operator can exercise control without the aid of outside authorities. For the purpose of implementing DOE-STD-3009-94, the DOE site boundary is a geographic boundary within which public access is controlled and activities are governed by DOE and its contractors, and not by local authorities. A public road traversing a DOE site is considered to be within the DOE site boundary if, when necessary, DOE or the site contractor has the capability to control the road during accident or emergency conditions.

Public

DOE-STD-3009-94 defines the public as all individuals outside the DOE site boundary.

Worker, Including Collocated Worker

This category consists of:

- Immediate workers—workers of DOE, DOE's contractors, or subcontractors with mission-related functions, who spend a substantial part of their working day inside the facility in the conduct of their duties. Among these are immediately engaged workers, whose functions require their presence in specially hazardous rooms or areas of the facility.
- Collocated workers—individuals who are employees of DOE or of one of DOE's operating contractors or mission-related subcontractors at the nuclear site where the facility in question is located, but who spend little, if any, of their time in the facility.

e. Differentiate between the function of structures, systems, and components in the following classifications:
- **Safety-class structures systems, and components**
- **Safety-significant structures, systems and components**

The following is taken from DOE-STD-3009-94.

Safety-Class Structures, Systems, and Components

Structures, systems, and components including portions of process systems, whose preventive and mitigative function is necessary to limit radioactive hazardous material exposure to the public, as determined from the safety analyses.

Safety-Significant Structures, Systems, and Components

Structures, systems, and components which are not designated as safety-class SSCs but whose preventive or mitigative function is a major contributor to defense–in-depth and/or worker safety as determined from safety analyses.

f. Differentiate between the function and contents of the following documents:
- **TSR**
- **DSA**
- **Unreviewed Safety Question Determination (USQD)**
- **Safety design strategy**
- **Conceptual safety design report**

- **Preliminary safety design report**
- **Preliminary documented safety analysis**

TSR

The following is taken from DOE-STD-3009-94.

TSRs are the limits, controls, and related actions that establish the specific parameters and requisite actions for the safe operation of a nuclear facility and include, as appropriate for the work and the hazards identified in the DSA for the facility: SLs, operating limits (OLs), surveillance requirements, administrative and management controls, use and application provisions, and design features, as well as a bases appendix.

DSA

The following is taken from 10 CFR 830.204.

The DSA for a hazard category 1, 2, or 3 DOE nuclear facility must, as appropriate for the complexities and hazards associated with the facility

- describe the facility and the work to be performed;
- provide a systematic identification of natural and man-made hazards associated with the facility;
- evaluate normal, abnormal, and accident conditions, including consideration of natural and man-made external events, identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents which may be beyond the design basis of the facility;
- derive the hazard controls necessary to ensure adequate protection of workers, the public, and the environment, demonstrate the adequacy of these controls to eliminate, limit, or mitigate identified hazards, and define the process for maintaining the hazard controls current at all times and controlling their use;
- define the characteristics of the safety management programs necessary to ensure the safe operation of the facility, including QA, procedures, maintenance, personnel training, conduct of operations, emergency preparedness, fire protection, waste management, and radiation protection; and
- with respect to a nonreactor nuclear facility with fissionable material in a form and amount sufficient to pose a potential for criticality, define a CSP program that
    - ensures that operations with fissionable material remain subcritical under all normal and credible abnormal conditions;
    - identifies applicable NCS standards; and
    - describes how the program meets applicable NCS standards.

Unreviewed Safety Question Determination (USQD)

The following is taken from the USQD Application Guide.

The basic purpose of the USQD process is to ascertain if a change (modification, test, or experiment) to a facility can be made without a prior safety review and approval by the original approving body. The USQD process judges whether the change could result in the facility being outside its authorization basis. If the change could result in the facility being outside its authorization basis, the change involves a USQ.

Safety Design Strategy

The following is taken from DOE G 420.1-1.

The SDS is based on the premise that no one layer of protection is completely relied upon to ensure safe operation. By applying this safety strategy, the DOE O 420.1 objective of providing multiple layers of protection to prevent or mitigate an unintended release of radioactive material to the environment can be achieved.

Conceptual Safety Design Report

The following is taken from DOE-STD-1189-2008.

A CSDR is developed to

- document and establish a preliminary inventory of hazardous materials, including radioactive materials and chemicals;
- document and establish the preliminary hazard categorization of the facility;
- identify and analyze primary facility hazards and facility design basis accidents;
- provide an initial determination, based on preliminary hazard analysis, of safety class and safety significant SSCs;
- include a preliminary assessment of the appropriate seismic design category for the facility itself, as well as the safety significant SSCs;
- evaluate the security hazards that can impact the facility safety basis (if applicable); and
- include a commitment to the nuclear safety design criteria of DOE O 420.1 (or proposed alternative criteria).

Preliminary Safety Design Report

The following is taken from DOE-STD-1189-2008.

The PSDR is developed during preliminary design and updates and provides additional site and design details to those provided in the CSDR. The PSDR follows the format and content of the PDSA produced during final design.

Preliminary Documented Safety Analysis

The following is taken from DOE-STD-1189-2008.
The PDSA is documentation prepared in connection with the design and construction of a new DOE nuclear facility or a major modification to a DOE nuclear facility that provides a reasonable basis for the preliminary conclusion that the nuclear facility can be operated safely through the consideration of factors such as

- the nuclear safety design criteria to be satisfied;
- a safety analysis that derives aspects of design that are necessary to satisfy the nuclear safety design criteria; and
- an initial listing of the safety management programs that must be developed to address operational safety considerations.

g. Differentiate between the plant/facility features which have the following designations:

- **Mitigating features**
- **Preventive features**

The information for KSAs g and h is taken from DOE-STD-3009-94.

Mitigating (Mitigative) Features

Any SSC that serves to mitigate the consequences of a release of hazardous materials in an accident scenario. Mitigative features function after the accident, and affect consequence.

Preventive Features

Any SSC that serves to prevent the release of hazardous material in an accident scenario. Preventive features function to avoid the accident and affect frequency.

h. Differentiate between the following types of facilities:
- **Nuclear facility**
- **Non-reactor nuclear facility**

Nuclear Facility.

A reactor or a nonreactor nuclear facility where an activity is conducted for or on behalf of DOE and includes any related area, structure, facility, or activity to the extent necessary to ensure proper implementation of the requirements established by 10 CFR 830.

Nonreactor Nuclear Facility.

Those facilities, activities, or operations that involve, or will involve, radioactive and/or fissionable materials in such form and quantity that a nuclear or nuclear explosive hazard potentially exists to workers, the public, or the environment, but does not include accelerators and their operations and does not include activities involving only incidental use and generation of radioactive materials or radiation such as check and calibration sources, use of radioactive sources in research and experimental and analytical laboratory activities, electron microscopes, and X-ray machines.

34. Criticality safety personnel must demonstrate a familiarity level knowledge of nuclear accident analysis techniques.

a. Identify and discuss essential elements of deterministic and PRA techniques.

Deterministic Risk Assessment

The following was provided by a team of subject matter experts at the NNSA Service Center.

The deterministic risk assessment method uses the questions: “What can go wrong?” and “What are the consequences?” This is known in some circles as a vulnerability assessment, and is very similar to the preliminary hazards analysis with no preventive or mitigative controls used in the early part of a safety analysis per DOE-STD-3009-94.

Probabilistic Risk Assessment (PRA)

The following is taken from NPR 8705.5A.

A PRA characterizes risk in terms of three basic questions: 1) What can go wrong? 2) How likely is it? and 3) What are the consequences? The PRA process answers these questions by systematically identifying, modeling, and quantifying scenarios that can lead

to undesired consequences, considering uncertainties in the progression of such scenarios due to both variations of, and limited knowledge about, the system and its environment. The PRA integrates models based on systems engineering, probability and statistical theory, reliability and maintainability engineering, physical and biological sciences, decision theory, and expert elicitation. The collection of risk scenarios allows the dominant contributors to risk and areas of uncertainty about risk to be identified.

Probabilistic risk assessment generally consist of complex chains of events (or scenarios), each of which can lead to an undesired consequence or end state. Examples of such events include failures of hardware and software system elements, human actions or lack thereof, and phenomenological events such as degradation or debris impacts. Complex scenarios may include events whose implications separately appear to be slight or insignificant but collectively can combine and interact to cause high -everity consequences. The total probability from the set of scenarios modeled may also be non-negligible even though the probability of each scenario is small.

The assessment normally takes place in the context of safety, health, and mission success criteria that specify a minimum required level of confidence that loss of life and equipment will be avoided, and mission objectives will be achieved. While elements of such requirements may be allocated to other disciplines (e.g., hardware reliability), the PRA provides an integral modeling framework in which various elements can be represented.

A PRA is conducted using a systematic process to assess operational objectives, application(s), and scope; model scenarios that can lead to undesired consequences or end states; quantify scenario probabilities and consequences, as applicable, including the characterization of uncertainty; and provide and interpret results for the decision(s) being supported. Documentation and communication are also important parts of the PRA process.

b. Identify and discuss the methods used to determine and analyze failure modes.

The following is taken from DOE-STD-1027-92.

There are many analytical techniques available for evaluating the safety of the wide spectrum of chemical and nuclear DOE facilities of varying complexity. These techniques are commonly applied in the design and operation of various types of processes in many industries. A good reference for applying these techniques is *Selecting Hazard Evaluation Techniques of Guidelines for Hazard Evaluation Procedures*, Second Edition with Worked Examples (Center for Chemical Process Safety, 1992). A list of target levels of analysis sophistication for types of operations in order of increasing complexity is presented below.

Low Complexity Operations—Use Hazard Analysis

Low-complexity operations include those in which very little or no processing of materials takes place. Waste storage, vaults, tanks, cylinders, canisters, or even very simple batch laboratories are examples of such facilities. Release mechanisms are largely intuitive or straightforward and can generally be identified by simple checklists. Hazard analysis that has already been performed is considered sufficient for identifying release mechanisms.

Single-Failure Electro-Mechanical Systems—Use Failure Modes and Effects Analysis (FMEA)

These systems include relatively simple electrical and mechanical devices in which a single-failure mechanism causes a release of materials. Simple one-step processes, single glovebox operations, and small furnaces are example of such devices. Failure modes and effects analysis is a bottom-up approach that looks at the failure of each element of a system or process and identifies the consequence of each failure. Failure modes and effects analysis is most appropriate for analysis of small segments of a system or process when it is determined that failure of single components in this segment could lead to system or process failure or release of material.

Failure modes and effects analysis has some limitations which must be recognized to ensure its appropriate use. First, FMEA is not very efficient for large-scale systems analysis because, by virtue of its bottom-up approach, it examines and documents the effects of component failures having little, if any, relevance to system failure or potential release. Second, FMEA considers only one failure at a time and has no logical process for considering multiple or combined failures. Third, FMEA is strictly equipment-oriented.

Systems with Redundant Barriers or Requiring Multiple Failures—Use Event Tree Analysis (ET)

Event tree analysis is a simple approach to delineating sequences of events which could lead to an undesired event. An undesired event could be uncontrolled release of hazardous material from a facility or core damage in a reactor. In the ET analysis, for each initiating event, various systems or barriers designed to prevent the occurrence of the undesired event or to mitigate the progress of the accident are identified. At each node, the success or failure of these systems or barriers, known as ET headings, is graphically shown. The result is a pictorial representation of various combinations of systems or barriers which succeed or fail to prevent the occurrence of the undesired event or to achieve a final safe condition. Event tree analysis is most helpful for delineation of sequences of events leading to release of material when there are multiple or redundant barriers for mitigation of the progression of the accident.

Large, Moderately Complex Processes—Use Fault Tree Analysis (FT)

Large, moderately complex processes include solid handling (e.g., machining and assembly) activities which include rather simple movement of materials from one discrete step to another. Both ET and FT analysis techniques are appropriate for such facilities. Fault tree analysis is a top-down approach for systematic assessment of various ways by which an undesirable event can occur. It begins with the undesirable event and proceeds to identify the event or sequence of events leading to that event. The FT can be developed to any desired level of detail. If quantification is desired, the FT is usually developed to the lowest level where data for these basic events are available, be it the subsystem, component, or component piece, or part level.

Since FT analysis starts from the undesirable event and logically identifies basic fault conditions which can contribute to its occurrence, only those faults contributing to the occurrence of undesired event are modeled.

Complex Fluid Processes—Use Hazard and Operability Studies (HAZOP)

Complex fluid processes involve arrays of piping, tanks, and instrumentation and control systems. Hazard and operability study is a standard and widespread technique used for the analysis of chemical flow processes. The main elements of HAZOP include determining

- the hazards which exist in a unit or are associated with a process;
- the effects associated with the hazard (e.g., safety, environmental, economic);
- the occurrence of accidents; and
- the measures to prevent a hazard from occurring or to mitigate the effects of an accident or failure.

Generally, HAZOP should be used for identifying accident scenarios associated with continuous processing which involves the control of a significant number of parameters in order to maintain the process in steady-state conditions and within safe limits. Such processes generally have systems intended to monitor key parameters. Such monitoring systems may interface with automatic control and protection systems which act to maintain the process in a safe condition or may only trigger alarms to alert the operator that a parameter change requires a response.

High Complexity Facilities—Use Integrated Event Tree and Fault Tree Techniques

Facilities with a large number of interdependent components or systems and fluid flow processes are highly complex. Highly interdependent systems and components should not be taken to include basic buildings systems such as heating, ventilation, and air conditioning and electrical power distribution systems unless these systems have significant effect on the progress of the accident sequence. Highly complex facilities include multi-component transfer and control systems for which extensive instrumentation and control systems are needed. Extensive redundancy at the component, system, and safety level are also inherent in highly complex facilities. Such processes generally cannot be completely controlled through manual actions because the interactions between systems are too intricate for an operator to interpret in the time required for action. Thus, these processes are generally characterized by large-scale monitoring and automatic control systems.

For such facilities, the extensive use of ETs and FTs is needed to understand the potential release mechanisms. The specification of the use of these techniques is due to the complex system interdependencies found in such facilities. Event tree/fault tree is capable of clarifying these interdependencies. The ET/FT technique involves defining initiating events leading to process disturbance and constructing detailed ET and FT models to represent plant response to various accident conditions resulting from those disturbances. These techniques have been proven to be especially useful in evaluating processes involving very complex systems with high levels of integration and interdependency.

Connecting of the initiating event and ET and FT models in a structured fashion is a proven technique capable of handling, in an efficient and comprehensive fashion, the very complex nature of the system designs, interactions, and dependencies prevalent in these processes. A large part of the reason for selecting this technique is that the nature of the hazard is straightforward, but its possible causes are numerous. For example, insufficient cooling to the reactor core leads to the release of large quantities of radionuclides from the core, but the causes of loss of coolant are many and intricate. Thus, the emphasis on

systems is a key benefit for evaluating these processes; other techniques structured to consider the hazards themselves (such as HAZOP) are not required.

c. Discuss the methods used in the calculation of criticality safety, source term, environmental transport, and dose assessment activities, including commonly used computer models.

Criticality Safety Calculations

The following is taken from DOE-STD-3007-2007.

Section D of DOE-STD-3007-2007 describes the methodology or methodologies used to establish limits for the operation being evaluated. Four methods that may be used for the establishment of subcritical limits are

1. reference to national consensus standards that present critical and/or subcritical limits;
2. reference to accepted handbooks of critical and/or subcritical limits;
3. reference to experiments with appropriate adjustments to ensure subcriticality when the uncertainties of parameters reported in the experiment documentation are considered; and/or
4. use of validated calculational techniques. (Note: One specific example based on validated calculations that may be used is the criticality index technique useful for setting limits on commingled arrays of fissile material containers.)

Calculation methods may include simple hand calculation techniques (e.g., limiting surface density, density analog), deterministic computer codes (e.g., ANISN or other Sn transport theory codes), and Monte Carlo computer codes (e.g., MCNP, KENO-V.a). When applicable, nuclear cross section data that were used should be identified (i.e., cross-section sets and release versions) along with any cross section processing codes that were used. When computer codes are used as part of the methodology, the type of computing platform along with relevant code configuration control information should be documented here. This information may be provided by reference.

Source Term Calculation

The following is taken from DOE-HDBK-3010-94, from which detailed explanations of the terms specified in the source term equation are available.

The source term is the amount of radioactive material, in grams or curies, released to the air. The initial source term is the amount of radioactive material driven airborne at the accident source. The initial respirable source term, a subset of the initial source term, is the amount of radioactive material driven airborne at the accident source that is effectively inhalable. Lesser source terms are determined by applying filtration or deposition factors to the initial source term.

The airborne pathway is of primary interest for nonreactor nuclear facilities. DOE-STD-1027-92, quotes observations of the NRC to the effect that “for all materials of greatest interest for fuel cycle and other radioactive material licenses, the dose from the inhalation pathway will dominate the (overall) dose” (NUREG-1140). The airborne source term is typically estimated by the following five-component linear equation:

Source Term = MAR x DR x ARF x RF x LPF

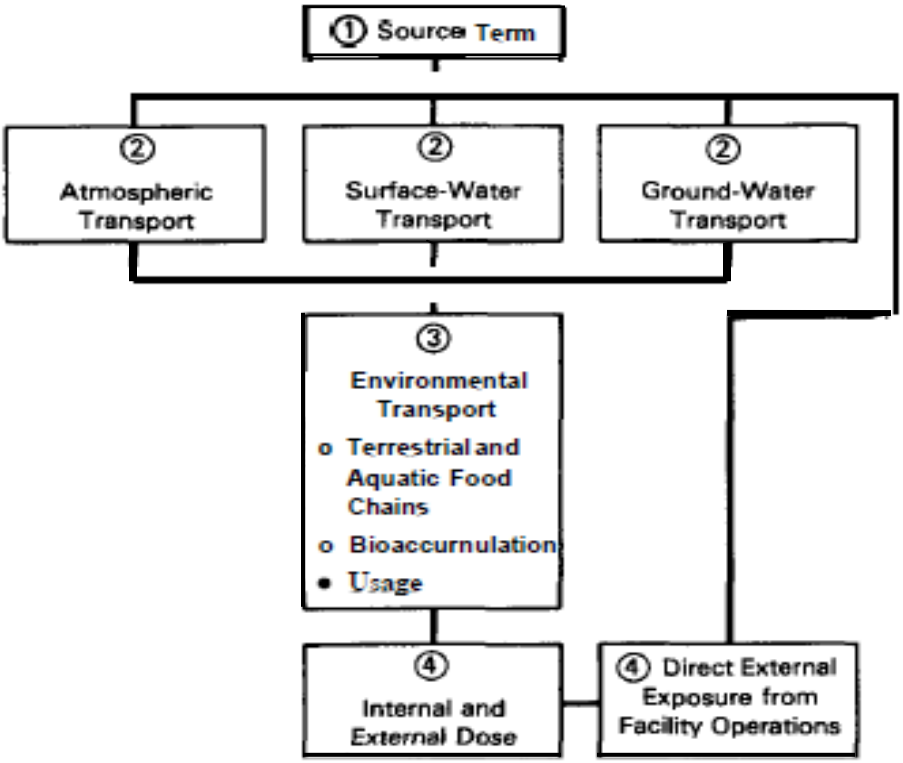
where:
MAR = material-at-risk (curies or grams)
DR = damage ratio
ARF = airborne release fraction (or airborne release rate for continuous release)
RF = respirable fraction
LPF = leakpath factor

The initial source term and initial respirable source term are products of the first three factors and first four factors respectively. A depleted source term after a subsequent stage of deposition or filtration is a product of the initial source term multiplied by the leakpath factor of the specific stage.

Environmental Transport, and Dose Assessment

The following is taken from DOE/EH-0173T

The basic considerations in performing an analysis of dose to the general public for the annual releases of radioactive materials from DOE facilities are shown in figure 18. Source-term estimates (box 1 in Figure 18) are obtained from the effluent monitoring programs established for each site, as described in chapters 2 and 3 of this Order. Models (boxes labeled 2 in figure 18) are then applied for atmospheric, surface-water, and groundwater transport. Environmental pathway analysis models (box 3 in figure 18) are then used to account for bioaccumulation in food products and the annual usage or uptake of materials by members of the public. The dose-rate factors (boxes labeled 4 in figure 18) to be used are the standard factors listed in EPA-520/l-88-020 and in DOE/EH-0071 and DOE/EH-0070.



Source: DOE/EH-0173T

Figure 18. Major steps in performing public radiation dose calculations

In applying models and computer programs for estimating public radiation doses, the following three critical assumptions should be evaluated for each application (Hoffman and Baes 1979): 1) the data available for the input parameters represent the true populations of the parameters (i.e., the data represent reality), 2) the model parameters are statistically independent (i.e., no coupled parameters), and 3) the structure of the model is an approximation of reality (i.e., the model fits the situation encountered). Although these three conditions can never be completely met, reasonable efforts should be made to evaluate these assumptions in light of the models and data sets selected for site-specific applications.

Radioactive materials released in the liquid effluents or airborne emissions from an operating DOE-controlled site or facility and transported through the environment might result in radiation exposures to members of the public. As shown in figure 18, the three major types of transport considered in evaluating the effects of radionuclides released to the environment are 1) atmospheric transport, 2) surface-water transport, and 3) groundwater transport. To estimate the concentrations of radioactive materials in the air or water at locations offsite, a number of mathematical models and computer programs are available. Examples of the methods for documenting computer programs are presented by ANSI (N413) and the Federal Information Processing Standard, publication 38. The correct operation of computer programs selected for performing the transport calculations for all environmental dose assessments should be verified on a specific computer system. This verification can be done by comparing the program results for sample problems

against either documented sample problem results or against hand calculations. Complete validation of all models (testing the computer program against actual field or laboratory data) is not feasible because of the size of some data sets and the inability to fully characterize most sites. Thus, limited comparisons against field or laboratory data are typically conducted during development of the computer program. As a result of these limited tests, modifications are often made to key parameter values to make the results compare more closely to measured conditions. This comparison process is called "model calibration" and is often used when site-specific model applications are desired. In many situations, site-specific data are not available, so default parameters or data sets are typically used in the transport calculations. These default values are often obtained from generic datasets and are designed to give conservative dose overestimates.

d. Discuss the methods used to identify and categorize the hazards associated with Department nuclear systems.

The following is taken from DOE-STD-1027-92.

DOE-STD-1027-92 outlines the process and thresholds of radionuclides that help determine the hazard category of each facility. The process includes preliminary hazard identification and a determination of the type and amount of the hazard as outlined in the chart of thresholds of radionuclides.

The preliminary assessment of hazards at a DOE nuclear facility requires only a minimal effort to identify the inventory of hazardous material in order to perform an initial hazard categorization. Reviewing basic facility information on intended facility operations and using estimates of material quantities should lead to an acceptable assessment. Whenever questions concerning appropriate facility categorization arise, provide for a margin of error by selecting the higher hazard category. This step results in the preliminary categorization of a DOE nuclear facility in a hazard category 1, 2, or 3 or below category 3, (radiological facility).
The objectives of hazard analysis are to

- identify the hazards contained in a facility;
- perform final hazard categorization in accordance with criteria in DOE-STD-1027-92, based on hazardous material quantity and energy sources and initiating events (preventive and mitigative features are not to be considered in hazard categorization that comes later);
- provide an overall assessment of the importance of the various hazards,
- identify occupational hazards and related DOE-prescribed standards, and
- characterize and analyze the remaining non-routine hazards that are unique and representative hazards to be analyzed.

e. Identify and discuss the role and use of human factors techniques in hazard and accident analysis.

The following is taken from *Outlook for Human Factors and Impact on Inherent Safety for the Process Industries*.

In evaluating process safety management (PSM) for compliance, the Occupational Safety and Health Administration (OSHA's) PSM compliance directive, briefly describes that a

human factors review may include a review of operator/process and operator/equipment interface, the number of tasks operators must perform and the frequency, the evaluation of extended or unusual work schedules, the clarity and simplicity of control displays, automatic instrumentation versus manual procedures, operator feedback and the clarity of signs and codes.

Components of an effective program that incorporates human factors issues into existing process safety programs should include:

- Management knowledge and commitment
- Written human factors policy
- Management system for implementing the human factors program
- Employee knowledge and involvement on human factors
- Training on human factors issues
- Incorporating human factors into hazards analysis and risk assessment
- Human factors in process design and process change
- Incident investigation and human factors root cause assessment
- Consideration of human factors in written work procedures
- Measurement and auditing of the human factors program performance

Probabilistic risk assessment (PRA) is difficult in criticality safety. The facility should use PRA techniques to evaluate the reliability of a control. Identifying the failure rate of equipment is useful in PRA. However, the facility should never use PRA to determine if a control is needed.

35. Criticality safety personnel must demonstrate a familiarity level knowledge of terminology associated with PRA techniques.

a. Define the following terms with respect to PRA:

- **Probability**
- **Reliability**
- **Availability**
- **Unavailability**
- **Risk**
- **Safety**
- **Accident sequence**
- **Dominant contributors**
- **Minimal cut set**

The following definitions are taken from NUREG-0492 unless specified otherwise.

Probability

The relative frequency definition of probability states that:

$$P(E_1) = \lim_{N \to \infty} \left(\frac{N_1}{N} \right)$$

Where:

$P(E_1)$ = the probability of a specific outcome (E_1) occurring
N_1 = the number of times that outcome E_1 has occurred
N = the number of repetitions or trials

Some obvious properties of $P(E_1)$ arise from this equation:
$0 < P(E_1) < 1$
If $P(E_1) = 1$, E_1 is certain to occur
If $P(E_1) = 0$, E_1 is impossible

Reliability

The reliability of a system (Rt) is the probability of continuous successful operation for time t. Expressed mathematically:

$$R(t) = e^{-t/\theta} = e^{-\lambda t}$$

The expected number of failures in time $t = t/\theta = \lambda t$ if $\lambda = 1/\theta$
But m = expected number of failures. Therefore,
$P\{0 \text{ occurrences of system failure}\} = e^{-m} = e^{t/\theta} = e^{-\lambda t}$

Availability

If q(t) represents the unavailability of a system or component (defined as the probability that the system or component is down at time t and unable to operate if called on), then:

1 – q(t) is the system or component availability (defined as the probability that the system or component is up and able to operate if called on.

Unavailability

Unavailability is the probability that the system or component is down at time t and unable to operate if called on (see availability above).

Risk

The following is taken from NUREG-1500, vol. 1.

Overall risk is the product of accident frequencies and consequences.

Safety

The following is taken from 51 FR 28044.

The following quantitative objectives are used to determine achievement of safety goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent of the sum of the cancer fatality risks resulting from all other causes.

Accident sequence

The following is taken from NUREG-1150, vol. 2.

In accident sequence ET analysis, accident sequences leading to core damage were defined by constructing ETs for each initiating event group. In general, separate ETs were constructed for each group.

System ETs that included the systems responding to each initiating event group as defined in the accident sequence initiating event analysis were constructed. The ET structure reflected system interrelationships and aspects of accident phenomenology that determined whether or not the sequences led to core damage. Phenomenological information, such as containment failure effects that potentially impact core cooling or other systems, was obtained from the staff involved in the accident progression and containment loadings analysis.

Dominant contributors

The following is taken from NPR 8705.5A.

The PRA integrates models based on systems engineering, probability and statistical theory, reliability and maintainability engineering, physical and biological sciences, decision theory, and expert elicitation. The collection of risk scenarios allows the critical components (dominant contributors) to risk and areas of uncertainty about risk that may require special attention during fabrication and assembly to be identified.

Minimal cut set

A minimal cut set is a smallest combination of component failures which, if they all occur, will cause the top event to occur. By this definition, a minimal cut set is thus a combination (intersection) of primary events sufficient for the top event. The combination is a "smallest" combination in that all the failures are needed for the top event to occur; if one of the failures in the cut set does not occur, then the top event will not occur (by this combination).

b. Define the following terms and differentiate between the associated processes: Note: NUREG-0492, *Fault Tree Handbook*, may be useful for this competency:
- **Event tree**
- **Fault tree**

Event Tree

The following is taken from NUREG 1150.

An ET is a pictorial representation of various combinations of systems or barriers which succeed or fail to prevent the occurrence of the undesired event or to achieve a final safe condition.

The following is taken from Clemens, P. L., *Event Tree Analysis*.

Event tree analysis is a bottom-up, deductive system safety analytical technique that is applicable to physical systems (with or without human operators) and decision-making/ managerial systems. It is complementary to other techniques such as FT analysis and FMEA.

Event tree analysis explores system responses to initiating "challenges" and enables probability assessment of success/failure. Examples of "challenges" are pipe or vessel burst, utility system failure, ignition of stored combustibles, etc.

Fault Tree

The following is taken from NUREG-0492.

The FT itself is a graphic model of the various parallel and sequential combinations of faults that will result in the occurrence of the predefined undesired event. The faults can be events that are associated with the component hardware failures, human errors, or any other pertinent events that can lead to the undesired event. An FT thus depicts the logical interrelationships of basic events that lead to the undesired event—which is the top event of the FT. Figure 19 depicts a typical FT diagram.

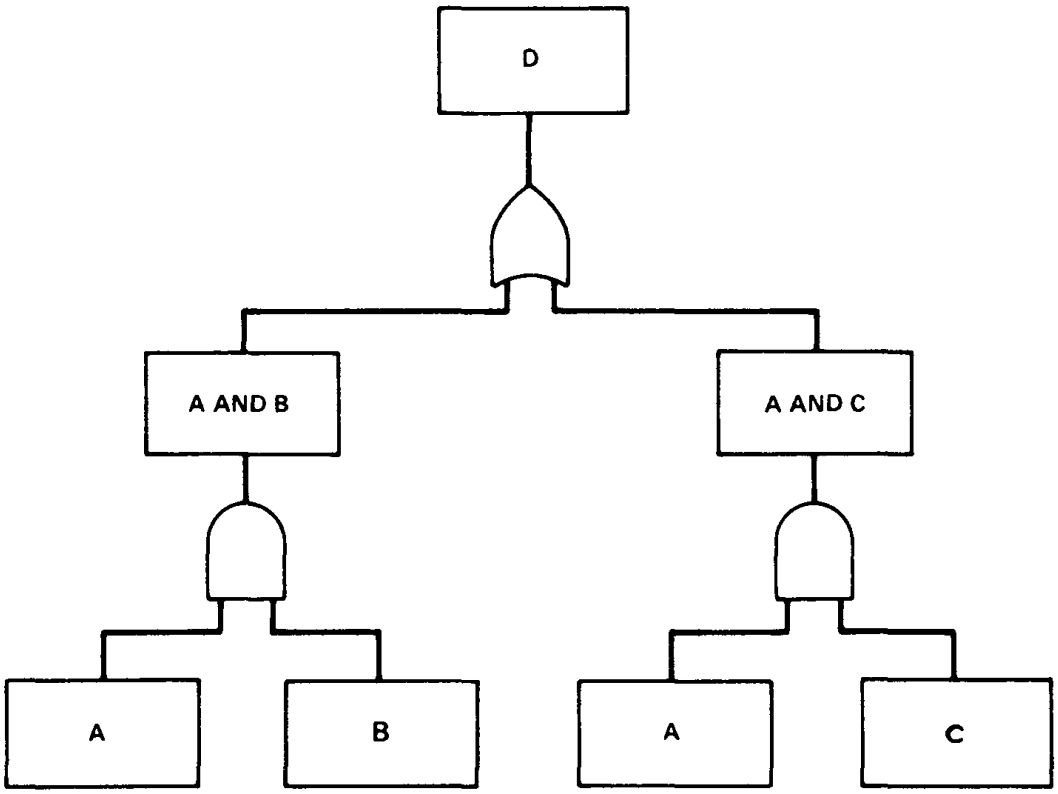


Figure 19. Fault tree diagram

Source: NUREG-0492.

An FT analysis can be simply described as an analytical technique, whereby an undesired state of the system is specified (usually a state that is critical from a safety standpoint), and the system is then analyzed in the context of its environment and operation to find all credible ways in which the undesired event can occur.

c. Discuss the concept of "credible" as used in criticality safety process evaluations as compared to the terms "not credible" or "beyond extremely unlikely" as used in DSAs.

The following is taken from DOE-STD-3007-2007.

Credible is the attribute of being believable on the basis of commonly acceptable engineering judgment. Due to the general lack of statistically reliable data, assigning numerical probabilities to events is not usually justifiable and when used should be backed up with references.

DOE-STD-3009-94 and DOE-STD-3011-2002 caution that a frequency cutoff such as less than 10^{-6}/yr (beyond extremely unlikely) is not to be used as an absolute cutoff for dismissing physically credible operational accidents without any evaluation of preventive and mitigative features in hazard analysis.

36. Criticality safety personnel must demonstrate a working level knowledge of the 10 CFR 830, *Nuclear Safety Management*, requirements related to USQs and the associated DOE Guide 424.1-1, *Implementation Guide for Use in Addressing Unreviewed Safety Question Requirements*.

[Note: DOE G 424.1-1 was replaced by DOE G 424.1-1A, which has been replaced by DOE G 424.1-1B]

a. Discuss the reasons for performing an USQD.

The following is taken from 10 CFR 830.203.

The contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must implement the DOE-approved USQ procedure in situations where there is a

- temporary or permanent change in the facility as described in the existing DSA;
- temporary or permanent change in the procedures as described in the existing DSA;
- test or experiment not described in the existing DSA;
- potential inadequacy of the DSA because the analysis potentially may not be bounding, or may be otherwise inadequate.

b. Define the following terms:

- **Accident analyses**
- **Safety analysis**
- **TSR**

The following definitions are taken from DOE-STD-3009-94.

Accident Analyses

Accident analysis has historically consisted of the formal development of numerical estimates of the expected consequence and probability of potential accidents associated with a facility. Accident analysis is a follow-on effort to the hazard analysis, not a fundamentally new examination requiring extensive original work. As such, it requires documentation of the basis for assignment to a given likelihood of occurrence range in

hazard analysis and performance of a formally documented consequence analysis. Consequences are compared with the evaluation guideline to identify safety-class SSCs.

Safety Analysis

A documented process: 1) to provide systematic identification of hazards within a given DOE operation; 2) to describe and analyze the adequacy of the measures taken to eliminate, control, or mitigate identified hazards; and 3) to analyze and evaluate potential accidents and their associated risks.

Technical Safety Requirements

Technical safety requirements mean the limits, controls, and related actions that establish the specific parameters and requisite actions for the safe operation of a nuclear facility and include, as appropriate for the work and the hazards identified in the DSA for the facility: safety limits, operating limits, surveillance requirements, administrative and management controls, use and application provisions, and design features, as well as a bases appendix.

c. Describe the situations for which a criticality safety evaluation is required to be performed.

The following is taken from DOE-STD-3007-2007.

Criticality safety evaluations must be performed to support production, operations, storage, and D&D activities at DOE facilities, document and analyze existing conditions and controls or develop new controls and limits for special circumstances such as shipping containers, justification for installation or removal of CAASs, or for operations with shielding and confinement.

Before a new operation with fissionable material is begun, or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions (ANSI/ANS 8.1).

d. Define the conditions for an USQ.

The following is taken from 10 CFR 830.203.

Four criteria define a USQ. They are a

1. temporary or permanent change in the facility as described in the existing DSA;
2. temporary or permanent change in the procedures as described in the existing DSA
3. test or experiment not described in the existing DSA; or
4. potential inadequacy of the DSA because the analysis potentially may not be bounding or may be otherwise inadequate.

e. Describe the responsibilities of contractors authorized to operate DOE nuclear facilities for the performance of safety evaluations.

The following is taken from 10 CFR 830.202.

The contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must establish and maintain the safety basis for the facility. The contractor must

- define the scope of the work
- identify and analyze the hazards
- categorize per DOE-STD-1027-92
- prepare a DSA
- update the safety basis to keep current, reflect changes in the facility, work, and hazards
- submit annual updates to DOE
- incorporate any changes, conditions, or hazard controls directed by DOE

f. Describe the actions to be taken by a contractor upon identifying information that indicates a potential inadequacy of previous safety analyses or, a possible reduction in the margin of safety as defined in the TSR.

The information for KSAs f, g, and h is taken from DOE G 424.1-1B.

Because a safety analysis inadequacy has potential to call into question information on which authorization of operations is based, per 10 CFR 830.203(g) the contractor is to

- take action, as appropriate, to place or maintain the facility in a safe condition until an evaluation of the safety of the situation is completed;
- notify DOE of the situation;
- perform a USQD and notify DOE promptly of the results; and
- submit the evaluation of the safety of the situation to DOE prior to removing any operational restrictions that were initiated.

g. Discuss the actions to be taken if it is determined that an USQ is involved.

When the USQD is positive, indicating the need for DOE review and approval of the change, the safety analyses and controls associated with the approved action become part of the safety basis for the facility. Any changes necessary to the DSA and TSR documents as a result of the change should be incorporated at the next annual update. The results of the USQD define the need for DOE approvals of the supporting CSEs and explicit updates of the DSA and TSRs.

h. Discuss the following terms as they apply to USQs:

- **Margin of safety**
- **Important to safety**
- **Safety basis**

Margin of Safety

A proposed change or test involves a USQ if a margin of safety could be reduced. For purposes of performing the USQ determination, a margin of safety is defined by the range between two conditions. The first is the most adverse condition estimated or calculated in safety analyses to occur from an operational upset or family of related upsets. The second condition is the worst-case value known to be safe, from an engineering perspective. This value should be related to the condition at which some accident prevention or mitigation action must be taken in response to the upset or accident, as required by a DOE-approved TSR, and not the actual predicted failure point of some component.

The bases for a hazard control should define the margin of safety. If the bases of a hazard control do not specifically identify a margin of safety, the DSA and other appropriate safety basis documents should be reviewed to determine whether the proposed change, test or experiment, or new information has or would result in a reduction in the margin of safety. The judgment on whether the margin is reduced should be based on physical parameters or conditions that can be observed or calculated.

The safety margin is sometimes implicitly described. A margin of safety can depend on a parameter other than one of the process variables. Therefore, the precise determination of a numerical value associated with a change is not always possible. Implicit margins are, for example, conditions for acceptance for a computer code, method, or industry-accepted practice. It may be sufficient to determine only the direction of the margin change (i.e., increasing or decreasing) due to the proposed change.

Safety margins generally include worst-case assumptions of initial conditions, conservative assumptions in computer modeling and codes, allowance for instrument drift and system response time, redundancy and independence of components in safety trains, and plant response during operating transient and accident conditions. A change that affects initial conditions, a system response time, or some other parameter that can affect the course of an accident analysis supporting the bases of hazard controls must be evaluated to determine whether the change would reduce the margin of safety.

Important to Safety

A proposed change or test involves a USQ if the probability of the occurrence or the consequences of an accident or the malfunction of equipment important to safety previously evaluated in the DSA could be increased,

Equipment important to safety should be understood to include any equipment whose function can affect safety either directly or indirectly. This includes safety class and safety significant SSCs, other systems that perform an important defense-in-depth function, equipment relied on for safe shutdown, and in some cases, process equipment. Support systems to safety systems that are required for the safety function are also safety systems, and should be included.

The safety analyses for the facility assume the proper functioning of equipment important to safety in demonstrating the adequacy of design. The proper functioning of other systems, including support systems, is generally assumed. The scope of the USQD should include these other systems. For example, a change that does either of the following is a change that increases the probability of a malfunction of equipment important to safety:

- Degrades the performance of equipment important to safety, assumed to function in the accident analysis, to below the performance level assumed in the existing safety analyses
- Increases the challenge to equipment important to safety assumed to function in the accident analysis (for example, more rapid pressure rise), degrading performance to a level below that assumed in the existing safety analyses

In answering this question, the first step is to determine what SSCs could be affected by the proposed change. Then the effects of this change on equipment important to safety are

evaluated, including both direct and indirect effects. Direct effects are those in which the change affects the equipment (for example, a motor change on a pump). Indirect effects are those in which the change affects one piece of equipment, which in turn can affect equipment important to safety. An example of indirect effects would be one piece of equipment falling on safety equipment.

After the impact of the change on equipment important to safety is identified, a determination is made whether an increase in the probability of a malfunction of the SSCs has occurred.

Safety Basis

According to 10 CFR 830, the safety basis is the DSA and hazard controls that provide reasonable assurance that a DOE nuclear facility can be operated safely in a manner that adequately protects workers, the public, and the environment.

The USQ process is applicable when the project identifies situations where it is apparent that the existing safety basis may not be bounding or may be otherwise inadequate. The existence of a USQ does not mean that a facility or operation is unsafe. The purpose of the USQ process is to alert DOE of events, conditions, or actions that affect the DOE approved safety basis of the facility or operation and ensure appropriate DOE line management action. If a change is proposed or a condition is discovered that could increase the risk of operating a facility beyond that established in the current safety basis, DOE, including NNSA, line management reviews and determines the acceptability of the change through the process of approving a revised safety basis that would be developed and submitted by the contractor.

37. Criticality safety personnel must demonstrate a working level knowledge of the 10 CFR 830, *Nuclear Safety Management*, requirements related to TSRs and the associated DOE Guide 423.1-1, *Implementation Guide for Use in Developing Technical Safety Requirements*.

[Note: DOE G 423.1-1 was replaced by DOE G 423.1-1A]

The information for all of the KSAs in this competency statement is taken from DOE G 423.1-1A.

a. Discuss the purpose of TSRs.

Technical safety requirements define the performance requirements of SSCs and identify the safety management programs used by personnel to ensure safety. Technical safety requirements are aimed at confirming the ability of the SSCs and personnel to perform their intended safety functions under normal, abnormal, and accident conditions. These requirements are identified through hazard analysis of the activities to be performed and identification of the potential sources of safety issues. Safety analysis to identify and analyze a set of bounding accidents that take into account all potential causes of releases of radioactivity also contribute to the development of TSRs.

b. Describe the responsibilities of contractors authorized to operate DOE nuclear facilities for TSRs.

10 CFR 830.205, "Technical Safety Requirements," requires DOE contractors responsible for hazard category 1, 2, and 3 DOE nuclear facilities to develop TSRs that identify the limitations to each DOE owned-contractor operated nuclear facility based on the DSA and any additional safety requirements established for the facility. Although not required by 10 CFR 830.205, there also may be a need to establish TSRs for safe operation of radiological facilities.

c. Define the following terms and discuss the purpose of each:

- **Safety limit**
- **Limiting control settings**
- **Limiting conditions for operation**
- **Surveillance requirements**

Safety Limits (SLs)

Safety limits are the limits on important process variables needed for the facility function that, if exceeded, could directly cause the failure of one or more of the passive barriers that prevent the uncontrolled release of radioactive materials, with the potential of consequences to the public above specified evaluation guidelines.

"Needed for the facility function" means the process variable is operator-controlled to accomplish the facility mission and, if the variable were left unchecked, would initiate an event that challenges the passive safety boundary. Safety-limit designation is distinct to process events because other events, such as external or natural phenomena events, that may also challenge the passive safety boundary have no SLs because they are not under operator control.

Limiting Control Settings (LCSs)

LCSs define the settings on safety systems that control process variables to prevent exceeding an SL.

Limiting control settings for reactors should include reactor trip system instrumentation setpoints. The reactor trip set-point limits are the nominal values at which the reactor trips are set and should be selected to provide sufficient allowances between the trip setpoint and the SL. This allowance will ensure the core and the reactor coolant system are prevented from exceeding SLs during normal operation and anticipated operational occurrences.

Limiting control settings of instruments that monitor process variables at nonreactor nuclear facilities are the settings that either initiate protective devices themselves or sound an alarm to alert facility personnel to take action to protect barriers that prevent the uncontrolled release of radioactive materials. An LCS is only specified for a variable that also protects an SL. Limiting control settings should be chosen so that there is adequate time after exceeding the setting to correct the abnormal situation, automatically or manually, before an SL is exceeded.

Limiting Conditions for Operation (LCOs)

Limiting conditions for operation define the limits that represent the lowest functional capability or performance level of safety SSCs required to perform an activity safely.

Limiting conditions for operation should include the initial conditions for those design basis accidents or transient analyses that involve the assumed failure of, or present a challenge to, the integrity of the primary radioactive material barrier. Identification of these variables should come from a search of each transient and accident analysis documented in the DSA. The LCO should be established at a level that will ensure the process variable is not less conservative during actual operation than was assumed in the safety analyses.

Limiting conditions for operation should also include those SSCs that are part of the primary success path of a safety sequence analysis, and those support and actuation systems necessary for them to function successfully. Support equipment for these SSCs would normally be considered to be part of the LCO if relied upon to support the SSCs function.

Surveillance Requirements (SRs)

Surveillance requirements are used to ensure operability or availability of the safety SSCs identified in the OLs, and are most often used with LCOs to periodically validate the operability of active systems or components that are subject to a limiting condition.

Surveillance requirements consist of short descriptions of the type of surveillance required and its frequency of performance. These statements should be as brief as possible but should identify those requirements needed to ensure compliance with the related OLs. Each SR should begin with a verb. Use of terms and sentence structure among requirements should be consistent.

Failure to perform a surveillance within the required time interval or failure of a surveillance to meet its acceptance criteria should result in the equipment/component/ condition being declared inoperable and should be considered a failure to meet the LCO. When equipment or a component fails the SR, the action required by the TSR for the inoperable equipment or component should be taken. Failure to take the required action is a TSR violation.

d. Describe the general content of each of the following sections of the TSR:

- **Use and application**
- **Safety limits**
- **Operating limits**
- **Surveillance requirements**
- **Administrative controls**
- **Basis**
- **Design features**

Use and Application

This section should contain basic information and instructions for using and applying the TSR. The following elements should be addressed under separate headings in this section:

- Definitions. Provide an alphabetical list of terms used throughout the TSR and their corresponding definitions.
- Operational modes (reactors). The number of modes should be held to a minimum based upon the minimum number required to be able to distinguish between different facility conditions and to ensure the provision of an adequate level of safety while in each condition.
- Operational modes (nonreactor nuclear facilities). The number of modes should be established based on the minimum number required to distinguish between different facility conditions as dictated by required equipment operability and needed parameter limits.
- Frequency notation. The frequency notations, as used in the surveillances and elsewhere, should be defined as specified in the guide when included in the TSR.

Safety Limits

Safety limits should describe as precisely as possible the process variables or the parameters being limited, and should state the limit in measurable units (pressure, temperature, flow, etc.). In general, SLs should be monitored continuously. They should be based on, and specified in terms of, three basic rules:

- Exceeding an SL is a TSR violation for each applicable mode.
- Each SL should have a mode applicability statement. This statement should consist of a simple list of modes or other conditions for which the SL is applicable.
- Action statements should describe the actions to be taken in the event that the SL is not met. These actions should first place the facility in a safe, stable condition, or should verify that the facility already is safe and stable and will remain so. Secondly, an action statement should establish the steps and time limits to correct the out-of-specification condition.

Operating Limits (OLs)

Technical safety requirements are defined by 10 CFR 830.3, "Definitions," as the limits, controls, and related actions that establish the specific parameters and requisite actions for the safe operation of a nuclear facility. Operating limits are one of the components of the TSRs, which also include, as appropriate for the work and the hazards identified in the DSA for the facility, SLs, SRs, administrative and management controls, use and application provisions, and design features, and a bases appendix.

Surveillance Requirements (SRs)

Surveillance requirement statements consist of short descriptions of the type of surveillance required and its frequency of performance. These statements should be as brief as possible but should identify those requirements needed to ensure compliance with the LCS or LCO. Begin each SR with a verb. Be consistent in use of terms and sentence structure among requirements. Describe the purpose of SRs; that is, SRs are requirements relating to test, calibration, or inspection that ensure the necessary operability and quality of safety-related systems and components required for the safe operation of a facility. Surveillance should be based on the following rules:

- Surveillance requirements must be met for all equipment, components, and conditions for the facility to be considered operable.
- Each SR should be performed at the specified frequency, with a maximum extension of 25 percent of the interval between any two consecutive surveillances.

(This extension is intended to provide operational flexibility both for scheduling and for performing surveillances. It should not be relied upon as a routine extension of the specified interval.)

- Special test exceptions to TSRs may be allowed under controlled conditions. These test exceptions should be placed in Section 3 (LCO). Any test exception should be clearly written to state which LCOs are being excepted, for how long, and under what conditions.

Administrative Controls (ACs)

The AC section should impose administrative requirements necessary to control operation of the facility such that it meets the TSR. The following topics should be included:

- Contractor responsibility. The facility or plant manager is responsible for overall operation of the nuclear facility and should delegate in writing the succession to this responsibility during his or her absence.
- Contractor organization. Onsite and offsite organizations should be described for facility operation and contractor management.
- Procedures. Operations procedures should provide sufficient direction to ensure that the facility is operated within its design basis and supports safe operation of the facility.
- Programs. Programs developed to ensure the safe operation of the facility should be discussed here and thereby committed to by reference.
- Minimum operations shift complement. This section of the ACs should include the maximum daily working hours and maximum number of consecutive days on duty.
- Operating support. A list of facility support personnel that includes name, title, and work and home telephone numbers should be maintained.
- Facility staff qualifications and training. Minimum qualifications for members of the facility staff in positions affecting safety should conform to the requirements of DOE Order 5480.20A, *Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities*, or a successor document, and should be provided in the AC section.
- Record-keeping. Records need to be kept of all information supporting the implementation of the TSR, including operational logs of modes changes, entering actions, surveillances, deviations, procedures, programs, meetings, recommendations, etc.
- Reviews and audits. The methods established to conduct independent reviews and audits should be described.
- Deviations from TSRs. The actions and reporting required when there are deviations from TSRs should be explained.

Basis

The TSR bases appendix provides summary statements of the reasons for the selection of each specific SL, OL, and SR. The bases show how the numerical values, conditions, surveillances, and action statements fulfill the purpose derived from the safety documentation. Included in the bases should also be a description of the safety functions that each safety system provides and identification of what is included in each safety system. The level of detail in the description should be sufficient for the operations staff to confirm that the system is operable. This description is provided so that the operations staff knows exactly what must be operable to consider the entire safety system operable.

The bases appendix references the basis for specific parts of the TSR given in the DSA and other safety documentation.

The bases appendix should present all conditions of operation, including limiting accident conditions. All systems, subsystems, components, structures, and equipment that are to be included in the TSR should be presented or referenced to other DSA chapters and discussed in this appendix.

The TSR bases should include the following:

- Identification of any requirements relevant to the safety basis that have been selected by the facility or imposed on the facility by DOE
- Identification of specific information from the DSA used in the derivation of individual TSRs, including operating conditions limiting accident initial conditions, relevant parameters of safety class or safety significant SSCs, instrumentation, operator actions, assumed limits, and design features

Design Features

The purpose of the design features section is to describe in detail those features not covered elsewhere in the TSRs that, if altered or modified, would have a significant effect on safety. The following two areas should be addressed in this section:

- Vital passive safety SSCs such as piping, vessels, supports, structures (such as confinement), and containers.
- Configuration or physical arrangement including dimensions, the parameter(s) being controlled, and the reasoning behind the design should be provided as identified in the safety analysis. Examples of such situations are where criticality avoidance is dependent on physical separation and where equipment configuration is used to minimize radiation levels.

e. Discuss the conditions that constitute a violation of the TSRs and state the reporting requirements should a violation occur.

Violations of a TSR occur as a result of the following four circumstances.

1. Exceeding a SL
2. Failure to complete an action statement within the required time limit following exceeding an LCS or failing to comply with an LCO
3. Failure to perform a surveillance within the required time limit
4. Failure to comply with an AC statement

Reporting of all TSR violations should be made in accordance with the provisions of DOE O 231.1A Chg 1.. The reporting of violations on ACs can involve judgment since the details of programs such as a program for criticality control do not appear directly as a TSR, and some program requirements are more important than others. Violations of controls identified in the accident or criticality scenarios in the DSA should be reported as if they were TSR violations. To ensure consideration for mitigation in potential enforcement actions, identified TSR violations should be evaluated for voluntary reporting to the DOE Noncompliance Tracking System.

f. Discuss the requirements for administrative control of the TSRs.

The facility must be operated in accordance with the provisions of a DOE-approved TSR. To ensure this is the case, the TSR and its appendixes must be an administratively controlled document so that only current copies of the DOE-approved TSR are used for operation of the facility. Making the TSR controlled involves establishing a list of the copies of the TSR that serve as official copies and instituting a formal process for issuing and distributing these copies and incorporating DOE-approved changes into them.

g. Discuss the possible source documents that may be used in developing TSRs.

The DSA required by 10 CFR 830.204, "Documented Safety Analysis," furnishes the technical basis for TSRs. For some facilities, other documentation such as the SER may provide additional safety controls or operating restrictions that should be reflected in the TSRs. The TSR derivation section in the DSA is intended to provide a link between the safety analysis and the list of variables, systems, components, equipment, and administrative procedures that must be controlled or limited in some way to ensure safety.

For existing facilities that have neither a DOE-approved DSA, nor DOE-approved technical specifications (TSs)/operational safety requirements, the schedule for developing the TSR should be coordinated with the DSA upgrade so that the TSR will reflect the DOE-approved DSA.

In areas for which the DSA does not directly supply all of the input for the TSR (e.g., surveillance frequencies and acceptance criteria), national and international codes, standards, and guides should be used wherever possible. Where no code, standard, or guide is applicable, other documents (e.g., reliability analyses, failure modes and effects analyses, manufacturer documentation, information from operating history, or engineering judgment) may provide the basis.

h. Discuss the requirements for emergency actions that depart from the approved TSRs.

In an emergency, if a situation develops that is not addressed by the TSR, site personnel are expected to use their training and expertise to take actions to correct or mitigate the situation. Also, site personnel may take actions that depart from the requirements of a TSR provided 1) an emergency situation exists; 2) these actions are needed immediately to protect workers, the public, or the environment from imminent and significant harm; and 3) no action consistent with the TSR is immediately apparent. Such action must be approved by a certified operator for reactor facilities or by a person in authority as designated in the TSRs for nonreactor nuclear facilities. (The designation of the person or persons should be indicated with their job title.) If emergency action is taken, a verbal notification should be made to the responsible head of the field element and a written report should be made to the CSO within 24 hours.

[Note: DOE O 151.1C requires that the headquarters and the field element manager be notified within 30 minutes of declaration of an operational emergency.]

38. Criticality safety personnel must demonstrate a familiarity level knowledge of DOE-STD-1186-2004, *Specific Administrative Controls,* (SACs) with respect to its impact on criticality safety.

The information for all of the KSAs in this competency statement is taken from DOE-STD-1186-2004.

a. Discuss how SACs are identified.

Specific administrative controls are ACs relied on to perform specific safety functions of importance similar to those of safety SSCs. An SAC exists when an AC is identified in the DSA as a control needed to prevent or mitigate an accident scenario and has a safety function that would be safety significant or safety class if the function were provided by an SSC.

b. Discuss the position of SACs in the preferred hierarchy of hazard controls.

When selecting hazard controls, it is preferable to choose engineering controls over ACs due to the inherent uncertainty of human performance. When choosing engineering controls, it is preferable to choose passive SSCs over active SSCs. When ACs are selected over engineering controls, and the AC meets the criteria for an SAC as provided in DOE-STD-1186-2004, the AC shall be designated as an SAC.

c. Describe how SACs are treated in DSAs and TSRs.

Specific administrative controls have elevated safety significance, and have more stringent implementation and verification requirements to ensure their effectiveness and dependability.

Controls identified as part of a safety management program may or may not end up as controls that need to have enhanced dependability, as is the case with SACs, based on the designations derived from the hazards and accident analyses in the DSA.

For site-wide safety management programs the DSA should explain the features of those programs that are important to the facility safety basis and can refer to the site-wide program documentation for the details. As appropriate to the hazard, the DSA may identify specific controls that are required for safety. These controls should be considered for designation as an SAC.

A clear distinction is made between programmatic ACs and SACs. Most ACs in the TSRs are designed to provide broad programmatic support for safety management programs supporting defense-in-depth, or worker safety. These ACs are classified as programmatic ACs. Programmatic ACs should not be used to provide specific or mitigative functions for accident scenarios identified in DSAs where the safety function has importance similar to, or the same as, the safety function of safety class or safety significant SSCs. The classification of SAC was specifically created for this safety function. ACs meeting the criteria in DOE-STD-1186-2004 for selection as SACs should be formulated, implemented, and maintained.

d. Discuss how SACs are implemented and maintained.

SACs are generally procedures. These procedures should include specifications for implementation such as qualifications of involved personnel, steps involved, verification of identified limits, frequency of verification, requirements for any independent verifications, interfaces with measuring equipment, and the required accuracy of the equipment, etc. TSRs are the formal requirements that implement those procedures and recovery actions in the case of breakdown of the control. SACs are addressed through the TSRs generally by two forms as identified below.

- Limiting condition for operation/surveillance requirement. Specific AC TSRs can often be written in the format of an LCO.
- Specific "directive action" AC. A specific directive action AC TSR can be in the ACs section of the TSRs.

DOE O 420.1B, section 4.5.1.2 states:

> Configuration management shall integrate the elements of system requirements and performance criteria, system assessments, change control/work control, and documentation control. Documents that define the system design basis (or when the design basis is not clearly defined, the identification of system requirements and performance criteria essential to the system's performance of its safety function, the basis for the requirements, and how the current system configuration satisfies the requirements and criteria) and supporting documents shall be compiled and kept current using a formal change control/work control program.

It is important that these requirements are applied to SACs to ensure the continuing ability of SACs to perform their safety function when called upon.

e. Describe measures used to ensure the dependability of SACs.

Lessons Learned on Human Actions Used for Safety Controls in Accident Scenarios

Each of the attributes listed in DOE-STD-1186-2004 to improve worker performance in using ACs should be carefully evaluated for improving the dependability of SACs.

Conduct of Operations

The dependability of all hazard controls, including SACs, is improved by implementing the facility-appropriate sections of the guidelines for conduct of operations provided in DOE Order 5480.19, *Conduct of Operations for DOE Facilities*. Two key elements of a proper conduct of operations program that can improve the dependability of SACs are independent verification and lockouts/tagouts. Detailed guidance is provided in DOE Order 5480.19 for each of these program elements.

Instrumentation, Controls and Support Equipment for SACs

Operators often must rely on effective instrumentation and controls and support equipment to implement SACs. For this reason, instrumentation and controls and equipment that support a SAC should meet performance requirements consistent with the importance of the safety function of the specific AC.

Training and Qualification for SACs

As a minimum, hazard analysts, personnel involved with formulation of SACs, and TSR writers should receive training on the guidance of DOE Order 5480.20A, chg. 1, *Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities*. Training on TSRs for operations personnel should include specific training on attributes of the SACs as identified in the safety basis. Training should also include training on the implementing procedures for SACs.

Establishing a Safety Culture

The Institute of Nuclear Power Operations *Excellence in Human Performance Initiative 2001* identified the key principles in developing an appropriate safety culture to improve human performance.

39. Criticality safety specialists must demonstrate a working level knowledge of DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, with respect to their impact on the Department's criticality safety.

a. Discuss the differences between the hazard categorization for dose to the public, and hazard categorization when criticality risks exist.

The following is taken from DOE-STD-1027-92.

Attachment 1 to DOE-STD-1027-92 classifies a facility as either hazard category 1, 2, or 3, depending only on the quantities of radioactive material in the facility, and gives the threshold quantities as well as the appropriate ground rules for evaluating the facility.

DOE Order 5480.23, *Nuclear Safety Analysis Report*, states that category 1 hazards have the potential for "significant offsite consequences." Based on total curie content, potential material forms, and maximum energy for dispersion available, one class of facilities which possess this hazard potential is the class A nuclear reactors (class A reactors are those that have a steady-state power level greater than 20 megawatts). In addition, the PSO may designate other facilities as category 1 if he feels there exists the potential for significant off-site consequences. Dose to the public would be considered as a significant offsite consequence.

Hazard category 2 is assigned when the hazard analysis shows the potential for significant onsite consequences. It includes facilities with the potential for nuclear criticality events or with sufficient quantities of hazardous material and energy, which would require onsite emergency planning activities.

b. Describe the requirements for a facility to remain hazard category 3 or below if a criticality program or criticality controls exist.

The following was derived from DOE-STD-1027-92 by a team of subject matter experts at the NNSA Service Center.

If the facility fissile inventory could exceed the minimum values specified in table A.1 of DOE-STD-1027-92, then a program is present to ensure that "segmentation" or "nature of process" conditions remain viable.

The following is taken from DOE-STD-1027-92.

In facility categorization, flexibility must be allowed in the definition of facility segments. Many DOE facilities conduct a wide variety of activities in one facility, ranging from simple assay or lab experiments to complex fluid flow separations. It is necessary to avoid placing excessive requirements on simple or even trivial co-located operations. The concept of independent facility segments should be applied where facility features preclude bringing material together or causing harmful interaction from a common severe phenomenon.

c. Discuss the role of criticality safety personnel in implementation of DOE-STD-1027-92.

The following was provided by a team of subject matter experts at the NNSA Service Center.

Criticality safety personnel evaluate CSPs to determine whether the program complies with applicable codes, standards, guide, regulations, Orders, and accepted practices.

40. Criticality safety personnel must demonstrate a working level knowledge of the 10 CFR 830, *Nuclear Safety Management*, requirements related to DSAs and the associated DOE G 421.1-2, *Implementation Guide In Developing Documented Safety Analysis to Meet Subpart B of 10 CFR 830*.

a. Discuss the four basic purposes and objectives of DSAs.

As given in 10 CFR 830.204, the purposes of the DSA follow the same principles as ISM or any good safety evaluation, that is define the scope of the operation, identify the potential accident initiators and enablers, analyze the accident conditions, and develop barriers or controls to prevent if possible, else mitigate the accident. Criticality evaluations are similar, but usually do not involve mitigation. As given in the CFR, the purposes are to

- describe the facility and the work to be performed;
- provide a systematic identification of both natural and man-made hazards associated with the facility;
- evaluate normal, abnormal, and accident conditions, including consideration of natural and man-made external events, identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents which may be beyond the design basis of the facility;
- derive the hazard controls necessary to ensure adequate protection of workers, the public, and the environment, demonstrate the adequacy of these controls to eliminate, limit, or mitigate identified hazards, and define the process for maintaining the hazard controls current at all times and controlling their use.

b. Describe the responsibilities of contractors authorized to operate DOE nuclear facilities for the development and maintenance of a DSA.

DSA Development

The following is taken from 10 CFR 830.204.

The contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must obtain approval from DOE for the methodology used to prepare the DSA for the facility unless the contractor uses a methodology set forth in table 2 of appendix A of 10 CFR 830.

This DSA must, as appropriate for the complexities and hazards associated with the facility

- describe the facility (including the design of safety SSCs) and the work to be performed;
- provide a systematic identification of both natural and man-made hazards associated with the facility;
- evaluate normal, abnormal, and accident conditions, including consideration of natural and man-made external events, identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents which may be beyond the design basis of the facility;
- derive the hazard controls necessary to ensure adequate protection of workers, the public, and the environment, demonstrate the adequacy of these controls to eliminate, controls current at all times and controlling their use;
- define the characteristics of the safety management programs necessary to ensure the safe operation of the facility, including (where applicable) QA, procedures, maintenance, personnel training, conduct of operations, emergency preparedness, fire protection, waste management, and radiation protection; and
- with respect to a nonreactor nuclear facility with fissionable material in a form and amount sufficient to pose a potential for criticality, define a CSP that
  - ensures that operations with fissionable material remain subcritical under all normal and credible abnormal conditions,
  - identifies applicable NCS standards, and
  - describes how the program meets applicable NCS standards.

DSA Maintenance

The following is taken from 10 CFR 830.202 (c).

In maintaining the safety basis for a hazard category 1, 2, or 3 DOE nuclear facility, the contractor responsible for the facility must

- update the safety basis to keep it current and to reflect changes in the facility, the work and the hazards as they are analyzed in the DSA (approved USQDs shall be included also);
- annually submit to DOE either the updated DSA for approval or a letter stating that there have been no changes in the DSA since the prior submission; and
- incorporate in the safety basis any changes, conditions, or hazard controls directed by DOE.

c. Define the following terms and discuss the purpose of each:
- **Design basis**
- **Engineered safety features**
- **Safety analysis**

Design Basis

The following is taken from DOE Order 5480.23.

The design basis is the set of requirements that bound the design of SSCs within the facility. These design requirements include consideration of safety, plant availability, efficiency, reliability, and maintainability. Some aspects of the design basis are important to safety, although others are not.

The following is taken from 10 CFR 50.2.

Design bases means that information that identifies the specific functions to be performed by an SSC of a facility, and the specific values or range of values chosen for controlling parameters as reference bounds of design. These values may be
- restraints derived from generally accepted "state of the art" practices for achieving functional goals, or
- requirements derived from analyses (based on calculations and/or experiments) of the effects of a postulated accident for which an SSC must meet its functional goals.

The following was provided by a team of subject matter experts at the NNSA Service Center.

The purpose of the design basis is to set the functional and performance parameters that the system being designed must meet.

Engineered Safety Features

The following is taken from DOE-HDBK-1188-2006

Systems, components, or structures that prevent and/or mitigate the consequences of potential accidents described in the final safety analysis report including the bounding design basis accidents.

Safety Analysis

The following is taken from DOE-HDBK-1188-2006.

Safety analysis is a documented process to provide systematic identification of hazards within a given DOE operation; describe and analyze the adequacy of the measures taken to eliminate, control, or mitigate identified hazards; and analyze and evaluate potential accidents and their associated risks.

The following is taken from DOE Order 5480.23.

The purpose of safety analysis is to document the safety bases for and commitments to the control of subsequent operations. This includes staffing and qualification of operating

crews; the development, testing, validation, and inservice refinement of procedures and personnel training materials; and the safety analysis of the person-machine interface for operations and maintenance

d. Describe the requirements for the scope and content of a DSA and discuss the general content of each of the required sections of the report.

The following is taken from DOE G 421.1-2.

The DSA for a hazard category 1, 2, or 3 DOE nuclear facility must, as appropriate for the complexities and hazards associated with the facility

- describe the facility (including the design of SSCs) and the work to be performed;
- provide a systematic identification of both natural and man-made hazards associated with the facility;
- evaluate normal, abnormal, and accident conditions, including consideration of natural and man-made external events, identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents which may be beyond the design basis of the facility;
- derive the hazard controls necessary to ensure adequate protection of workers, the public, and the environment, demonstrate the adequacy of these controls to eliminate, limit, or mitigate identified hazards, and define the process for maintaining the hazard controls current at all times and controlling their use;
- define the characteristics of the safety management programs necessary to ensure the safe operation of the facility, including (where applicable) QA, procedures, maintenance, personnel training, conduct of operations, emergency preparedness, fire protection, waste management, and radiation protection; and
- with respect to a nonreactor nuclear facility with fissionable material in a form and amount sufficient to pose a potential for criticality, define a CSP that:
  - ensures that operations with fissionable material remain subcritical under all normal and credible abnormal conditions,
  - identifies applicable NCS standards, and
  - describes how the program meets applicable NCS standards.

Content is usually governed by DOE-STD-3009, which identifies the following chapters. Except for appendix A, all chapters include a discussion of their purpose and the application of the graded approach, as well as sections on Introduction and Requirements.

- Chapter One: Site Characteristics, includes the following sections:
  - Site description
  - Environmental description
  - Natural event accident initiators
  - Man-made external accident initiators
  - Nearby facilities
  - Validity of existing environmental analyses

- Chapter Two: Facility Description, includes the following sections:
  - Facility overview
  - Facility structure
  - Process descriptions

- Confinement systems
- Safety support systems
- Utility distribution systems
- Auxiliary systems and support facilities

- Chapter Three: Hazard and Accident Analyses, includes the following sections:
  - Hazard analysis
  - Accident analysis

- Chapter Four: Safety Structures, Systems, and Components, includes the following sections:
  - Safety class structures, systems and components
  - Safety significant structures, systems and components
  - Specific administrative controls

- Chapter Five: Derivation of Technical Safety Requirements, includes the following sections:
  - TSR coverage
  - Derivation of facility modes
  - TSR derivation
  - Design features
  - Interface with TSRs from other facilities

- Chapter Six: Prevention of Inadvertent Criticality, includes the following sections:
  - Criticality concerns
  - Criticality controls
  - Criticality safety program
  - Criticality instrumentation

- Chapter Seven: Radiation Protection, includes the following sections:
  - Radiation protection program and organization
  - ALARA policy and program
  - Radiation protection training
  - Radiation exposure control
  - Radiation monitoring
  - Radiological protection instrumentation
  - Radiological protection record keeping
  - Occupational radiation exposures

- Chapter Eight: Hazardous Material Protection, includes the following sections:
  - Hazardous material protection program and organization
  - The ALARA policy and program
  - Hazardous material training
  - Hazardous material exposure control
  - Hazardous material monitoring
  - Hazardous material protection instrumentation
  - Hazardous material record keeping
  - Hazard communication program
  - Occupational chemical exposures

- Chapter Nine: Radioactive and Hazardous Waste Management, includes the following sections:
  - Radioactive and hazardous waste management program and organization
  - Radioactive and hazardous waste streams and sources

- Chapter Ten: Initial Testing, In-Service Surveillance, and Maintenance, includes the following sections:
  - Initial testing program
  - In-service surveillance program
  - Maintenance program

- Chapter Eleven: Occupational Safety, includes the following sections:
  - Conduct of operations
  - Fire protection

- Chapter Twelve: Procedures and Training, includes the following sections:
  - Procedure program
  - Training program

- Chapter Thirteen: Human Factors, includes the following sections:
  - Human factors process
  - Identification of human-machine interfaces
  - Optimization of human-machine interfaces

- Chapter Fourteen: Quality Assurance, includes the following sections:
  - Quality assurance program organization
  - Quality improvement
  - Documents and records
  - Quality assurance performance

- Chapter Fifteen: Emergency Preparedness Program, includes the following sections:
  - Scope of emergency preparedness
  - Emergency preparedness planning

- Chapter Sixteen: Provisions for Decontamination and Decommissioning, includes the following section:
  - Description of conceptual plans

- Chapter Seventeen: Management, Organization, and Industrial Safety Provisions, includes the following sections:
  - Organizational structure, responsibilities and interfaces
  - Safety management policies and programs

- Appendix A: Evaluation Guideline, includes the following sections:
  - Introduction
  - Evaluation guideline
  - Dose comparison calculation
  - Functional classification process

- Additional considerations

e. Discuss the approval requirements for the DSA for new facilities and subsequent changes.

The following is taken from DOE G 421.1-2.

For new facilities, approval of the PDSA is required before construction is begun. A contractor may not begin operation of a new hazard category 1, 2 or 3 DOE nuclear facility or a major modification of an existing hazard category 1, 2 or 3 DOE nuclear facility before DOE issues a SER approving the safety basis for the facility or modification.

DOE-STD-1104-96, *Review and Approval of Nonreactor Nuclear Facility Safety Analysis Reports*, or successor document (now DOE-STD-1104-2009, *Review and Approval of Nuclear Facility Safety Basis and Safety Design Basis Documents*), provides guidance on the preparation of SERs. One of the guiding principles is that the SER is primarily a management document that provides the approval authority, the basis for the extent and detail of the DSA review, and the basis for any conditions of DSA approval.

f. Define who approves facility operations prior to achieving DSA upgrade approval.

The following is taken from DOE-STD-1104-2009.

Appendix A of 10 CFR 830 states that the DOE management official for a DOE nuclear facility (i.e., the assistant Secretary, the assistant administrator, or the office director who is primarily responsible for the management of the facility) has primary responsibility within DOE for ensuring that the safety basis for the facility is adequate and complies with the safety basis requirements of 10 CFR 830. It further states that the DOE management official is responsible for ensuring the timely and proper 1) review of all safety basis documents submitted to DOE and 2) preparation of a SER concerning the safety basis for a facility.

DOE O 413.3A, chg 1, assigns the authority to designate a safety basis approval authority with the authority to review and approve safety basis and safety design basis documents to the PSO. By assigning responsibilities for the review and approval of the DSA to another individual, the DOE management official for the facility establishes that individual as the new approval authority.

Assigning responsibilities carries concurrent delegation of authority recognized by the line management and those responsible for monitoring and auditing implementation of the 10 CFR 830.

g. Discuss the requirements for the contractor to maintain the DSA current.

The following is taken from 10 CFR 830.202 (c).

In maintaining the safety basis for a hazard category 1, 2, or 3 DOE nuclear facility, the contractor responsible for the facility must

- update the safety basis to keep it current and to reflect changes in the facility, the work and the hazards as they are analyzed in the DSA (approved USQDs shall be included also);
- annually submit to DOE either the updated DSA for approval or a letter stating that there have been no changes in the DSA since the prior submission; and
- incorporate in the safety basis any changes, conditions, or hazard controls directed by DOE.

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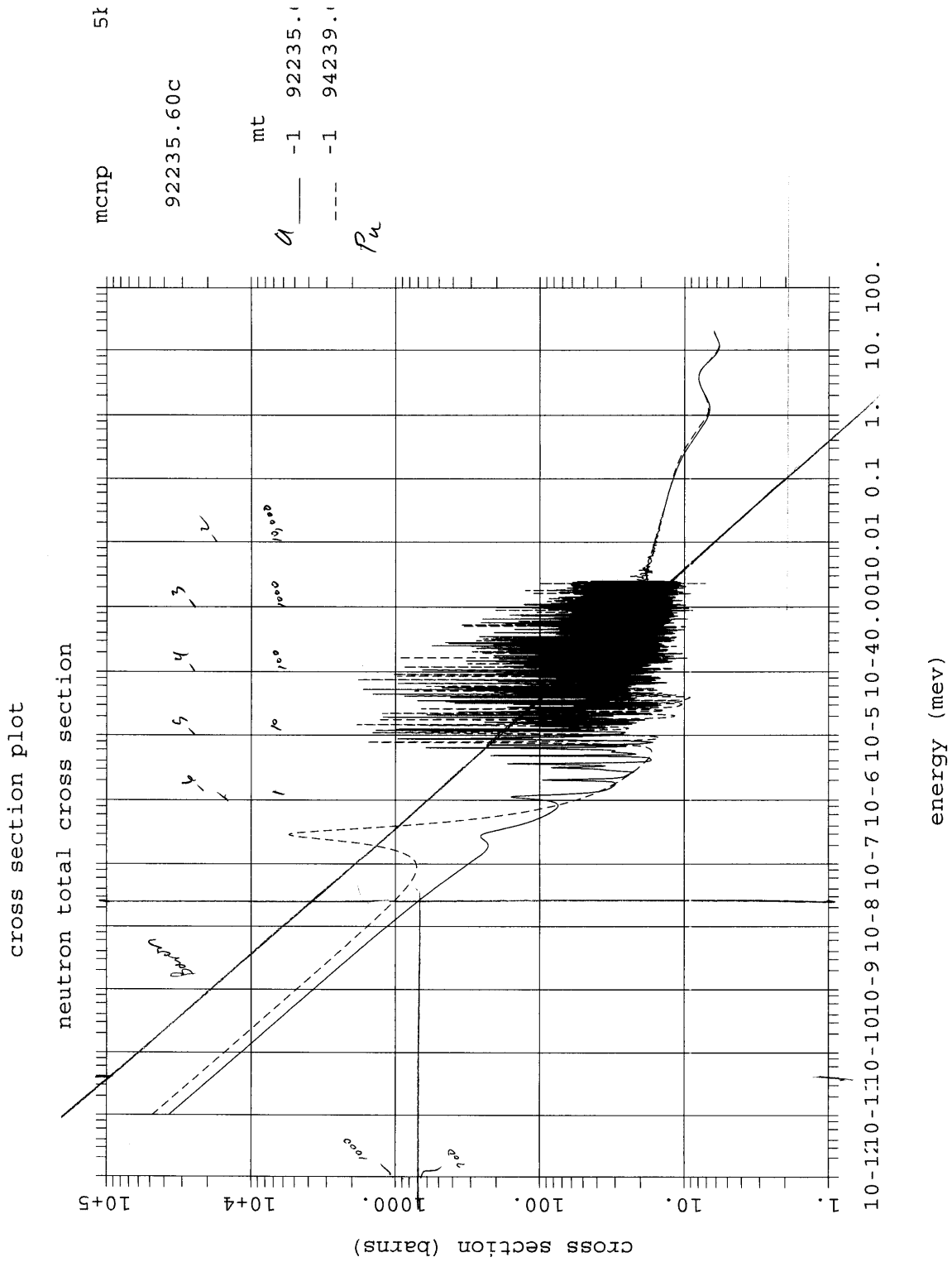
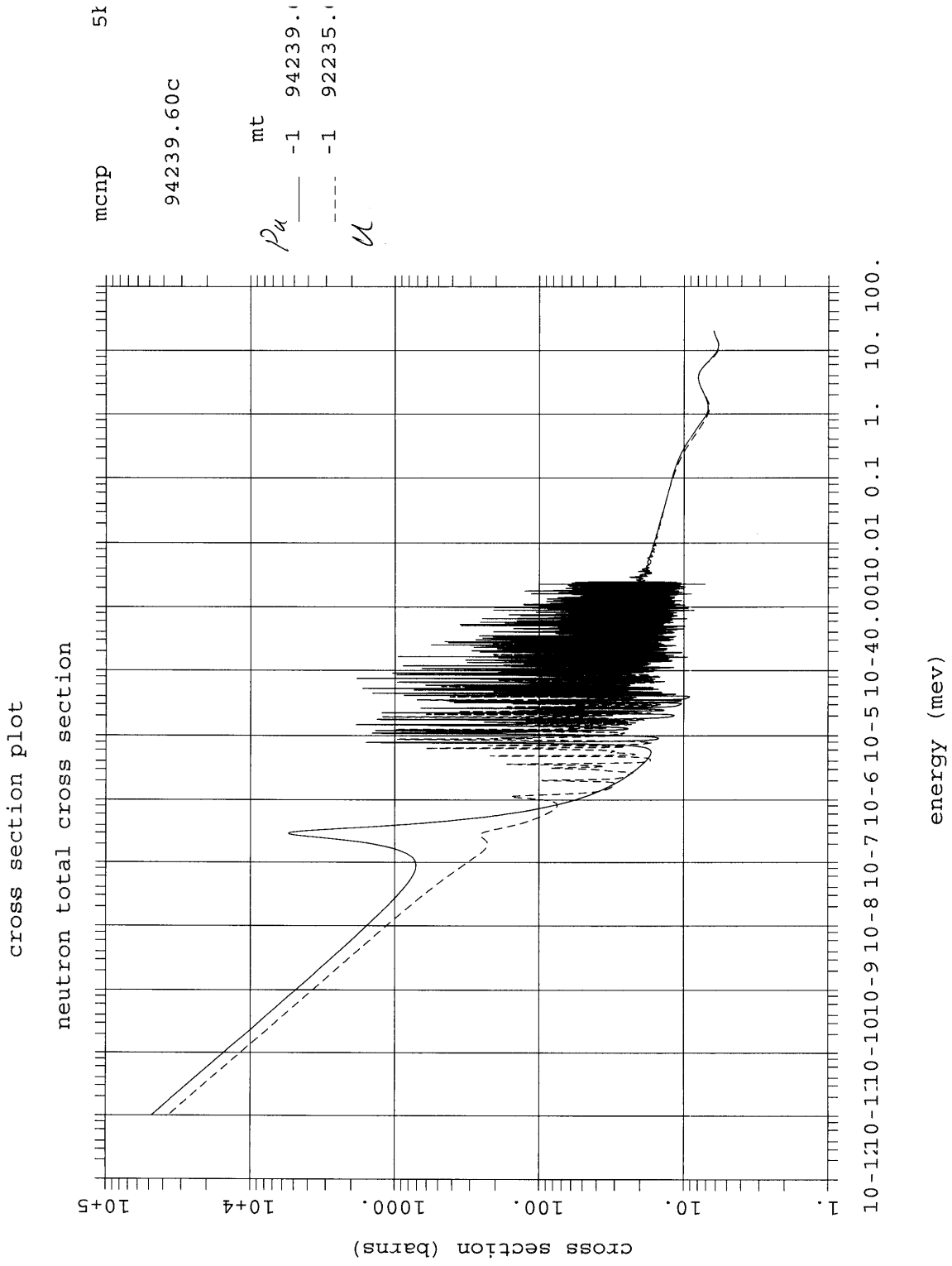
Appendix A. Cross Section Plots

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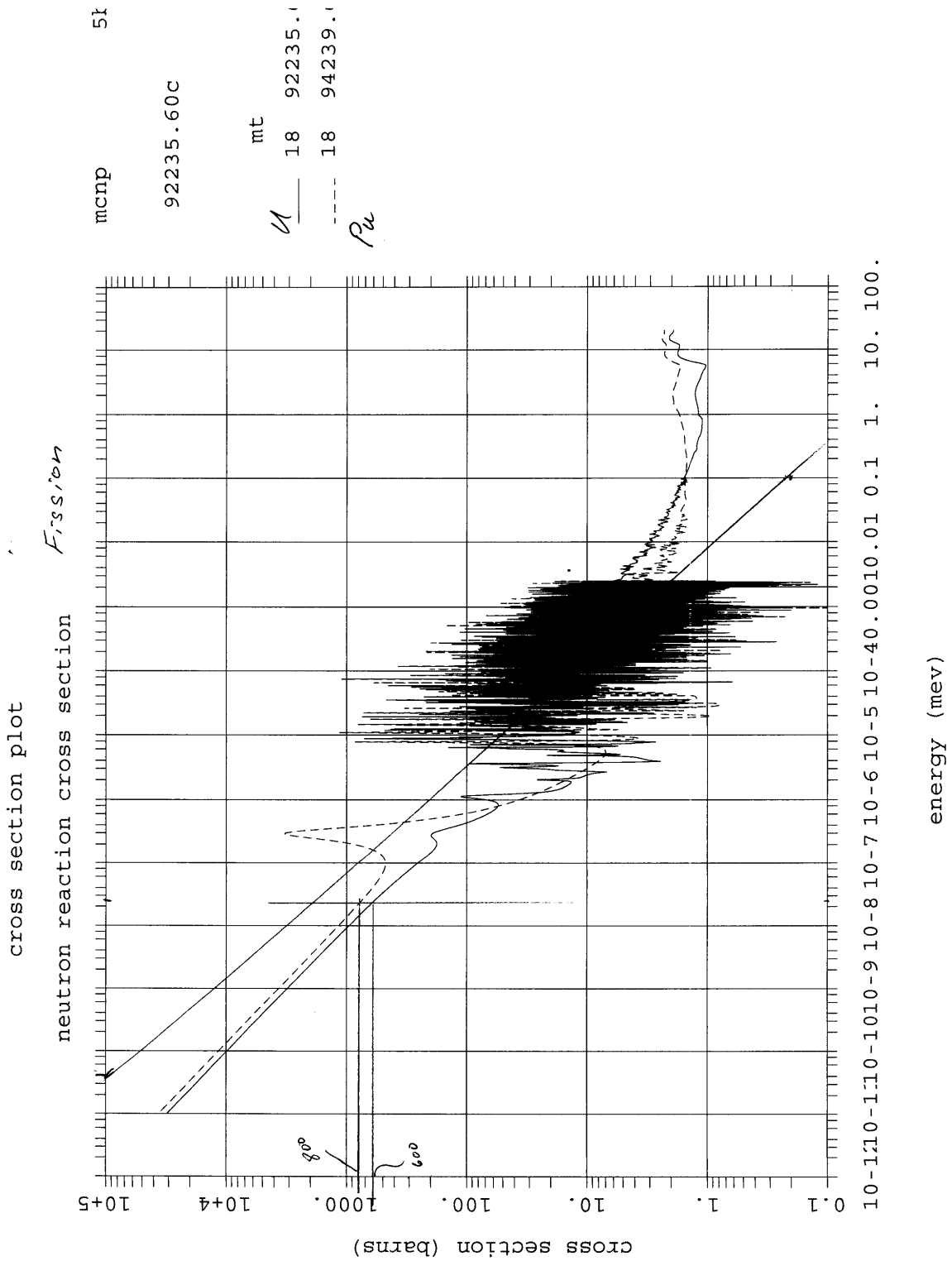
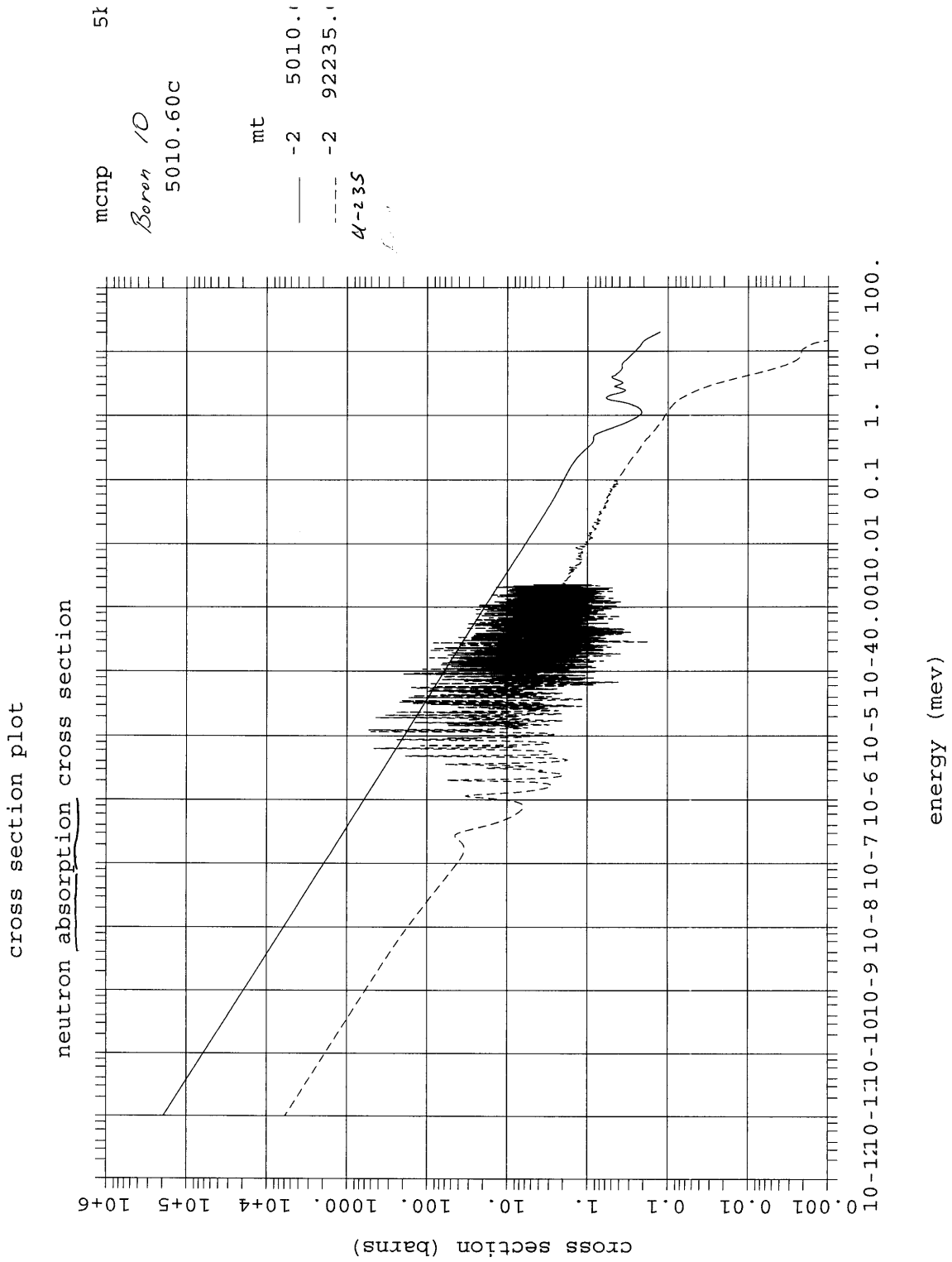
neutron total cross section
mcnp
5k
94239.60c
mt
-1 94239.
-1 92235.
cross section (barns)
10+5
10+4
1000.
100.
10.
1.
10-11 10-10 10-9 10-8 10-7 10-6 10-5 10-4 0.0010.01 0.1 1. 10. 100.
energy (mev)

Appendix A. Cross Section Plots

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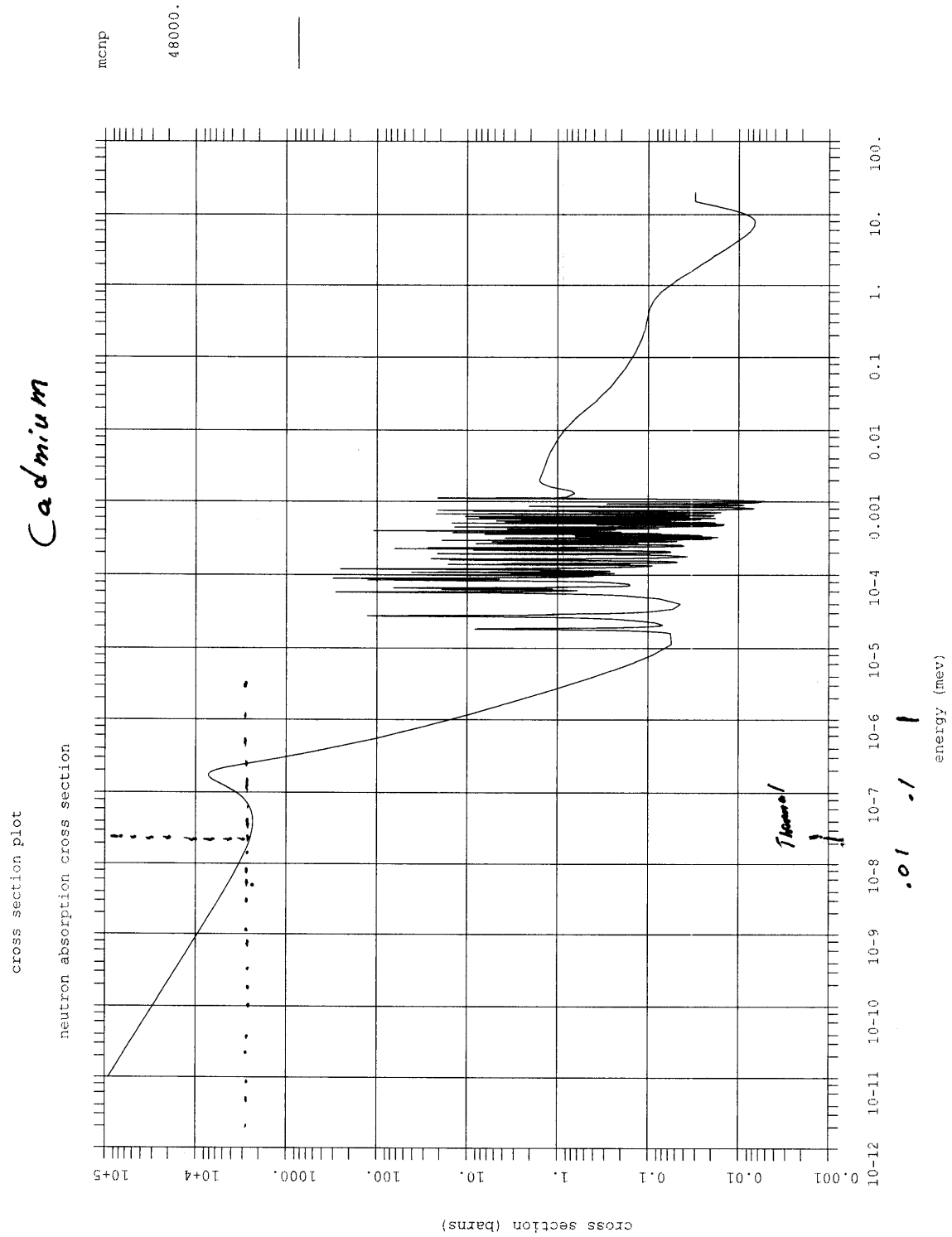
neutron absorption cross section
mcnp
Boron 10
5010.60c
mt
-2 5010.
-2 92235.
U-235
cross section (barns)
0.001 0.01 0.1 1. 10. 100. 1000. 10+4 10+5 10+6
10-11 10-10 10-9 10-8 10-7 10-6 10-5 10-4 0.0010.01 0.1 1. 10. 100.
energy (mev)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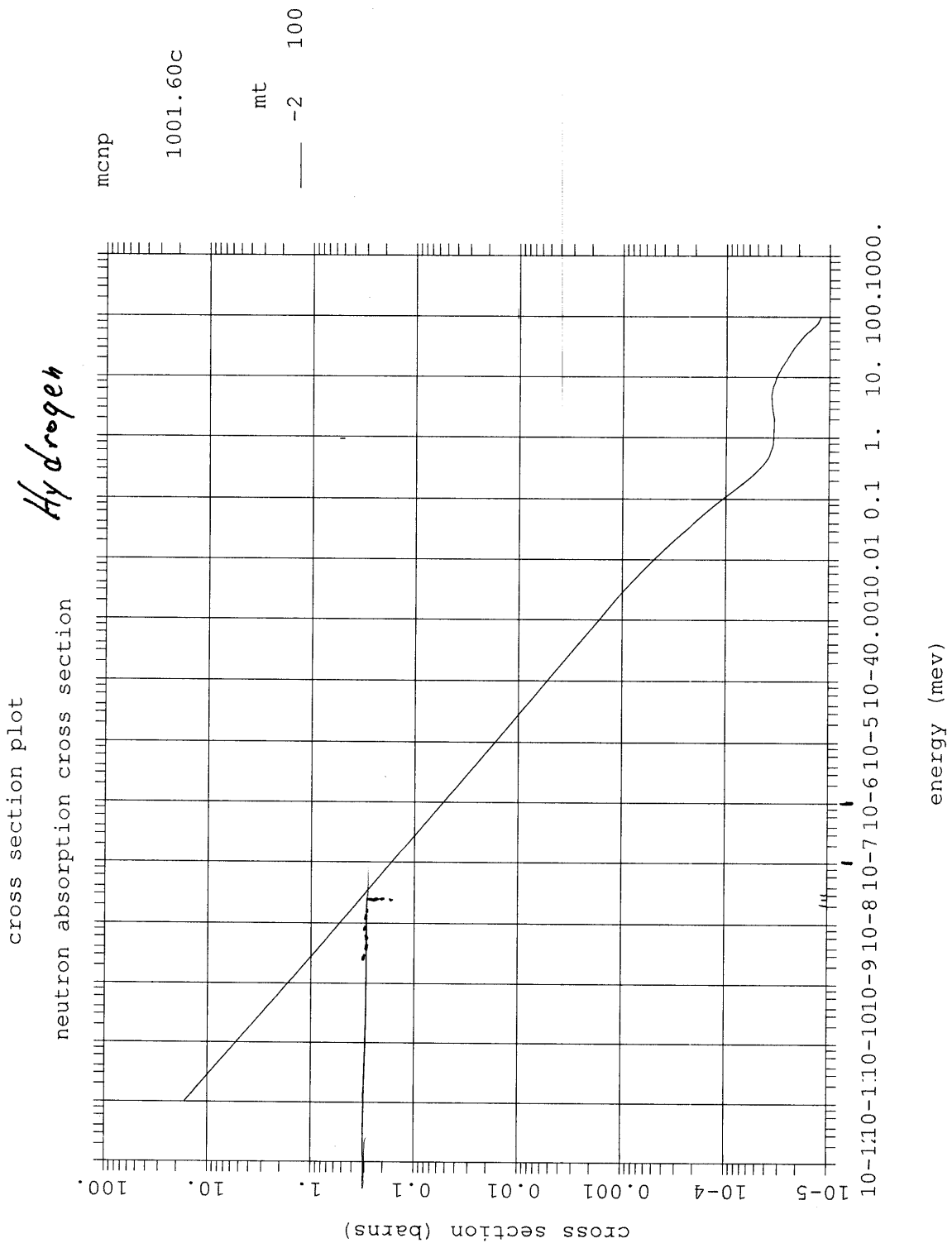
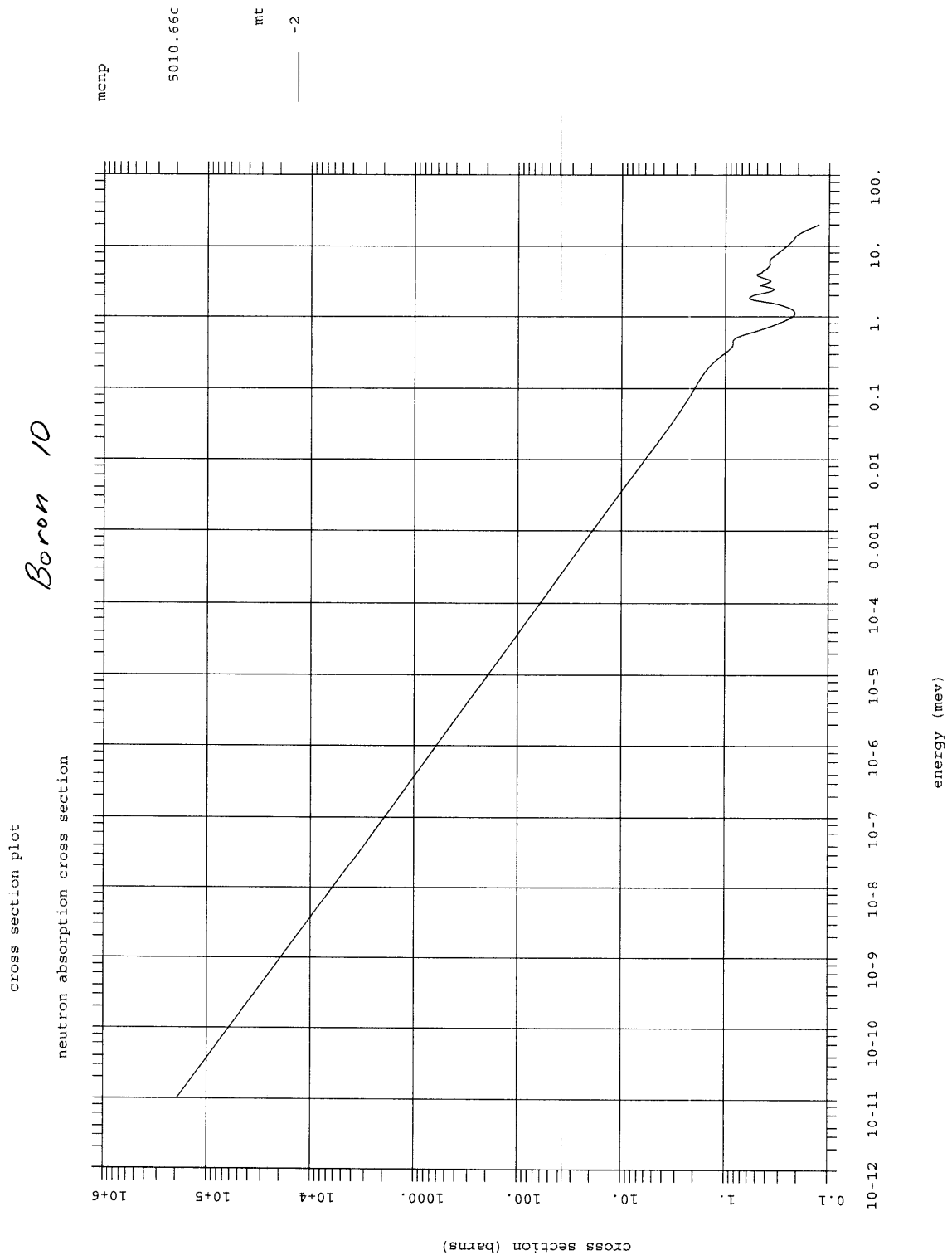
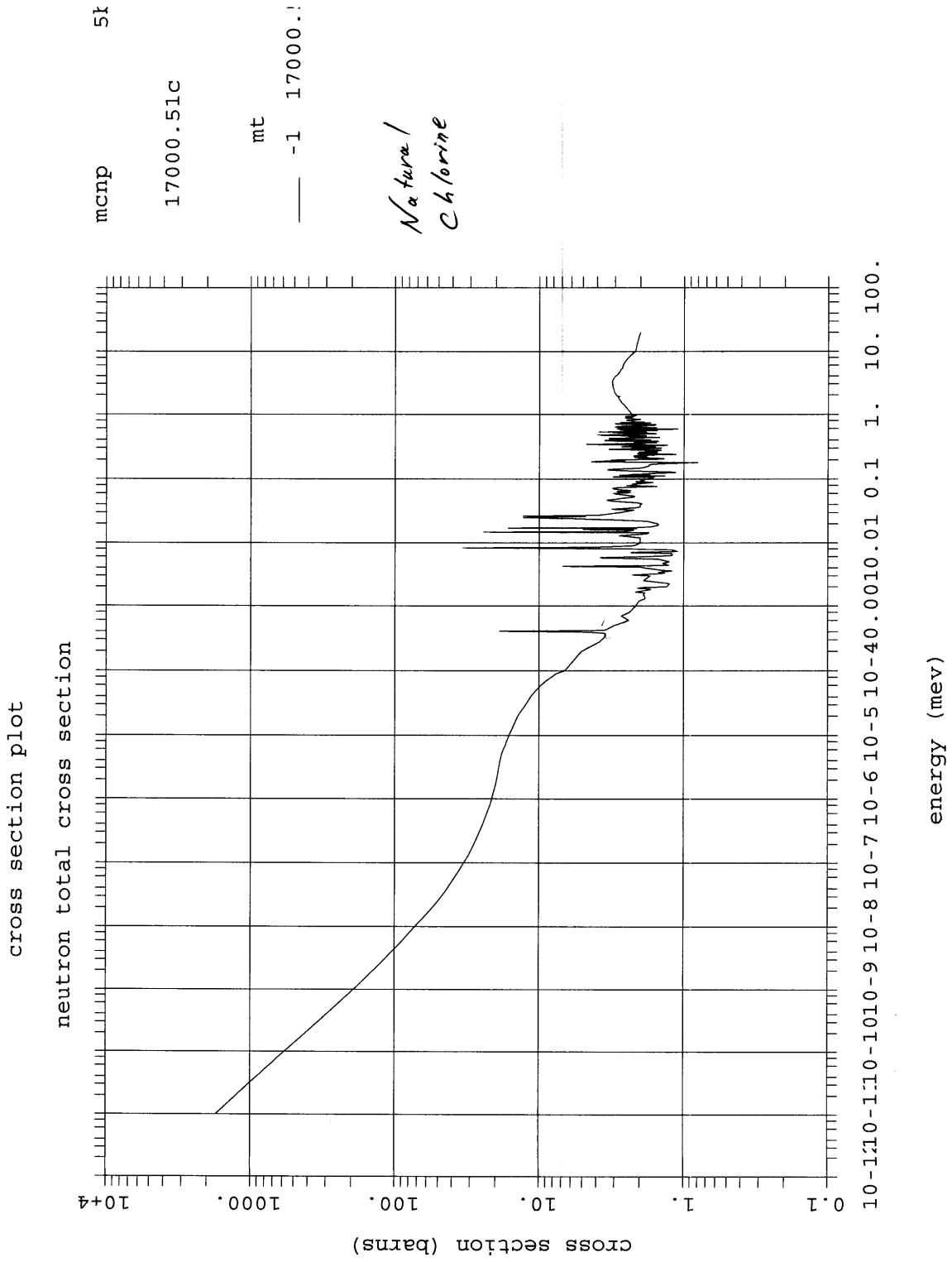
neutron absorption cross section
Cadmium
mcnp
48000.
cross section (barns)
0.001
0.01
0.1
1.
10.
100.
1000.
10+4
10+5
energy (mev)
10-12
10-11
10-10
10-9
10-8
10-7
10-6
10-5
10-4
0.001
0.01
0.1
1.
10.
100.
Thermal
.01
1


cross section plot
neutron absorption cross section
Hydrogen
mcnp
1001.60c
mt
-2 100
cross section (barns)
100.
10.
1.
0.1
0.01
0.001
10-4
10-5
10-11 10-10 10-9 10-8 10-7 10-6 10-5 10-4 0.001 0.01 0.1 1. 10. 100. 1000.
energy (mev)


mcnp
5010.66c
mt
-2
Boron 10
cross section plot
neutron absorption cross section
cross section (barns)
10+6
10+5
10+4
1000.
100.
10.
1.
0.1
energy (mev)
10-12
10-11
10-10
10-9
10-8
10-7
10-6
10-5
10-4
0.001
0.01
0.1
1.
10.
100.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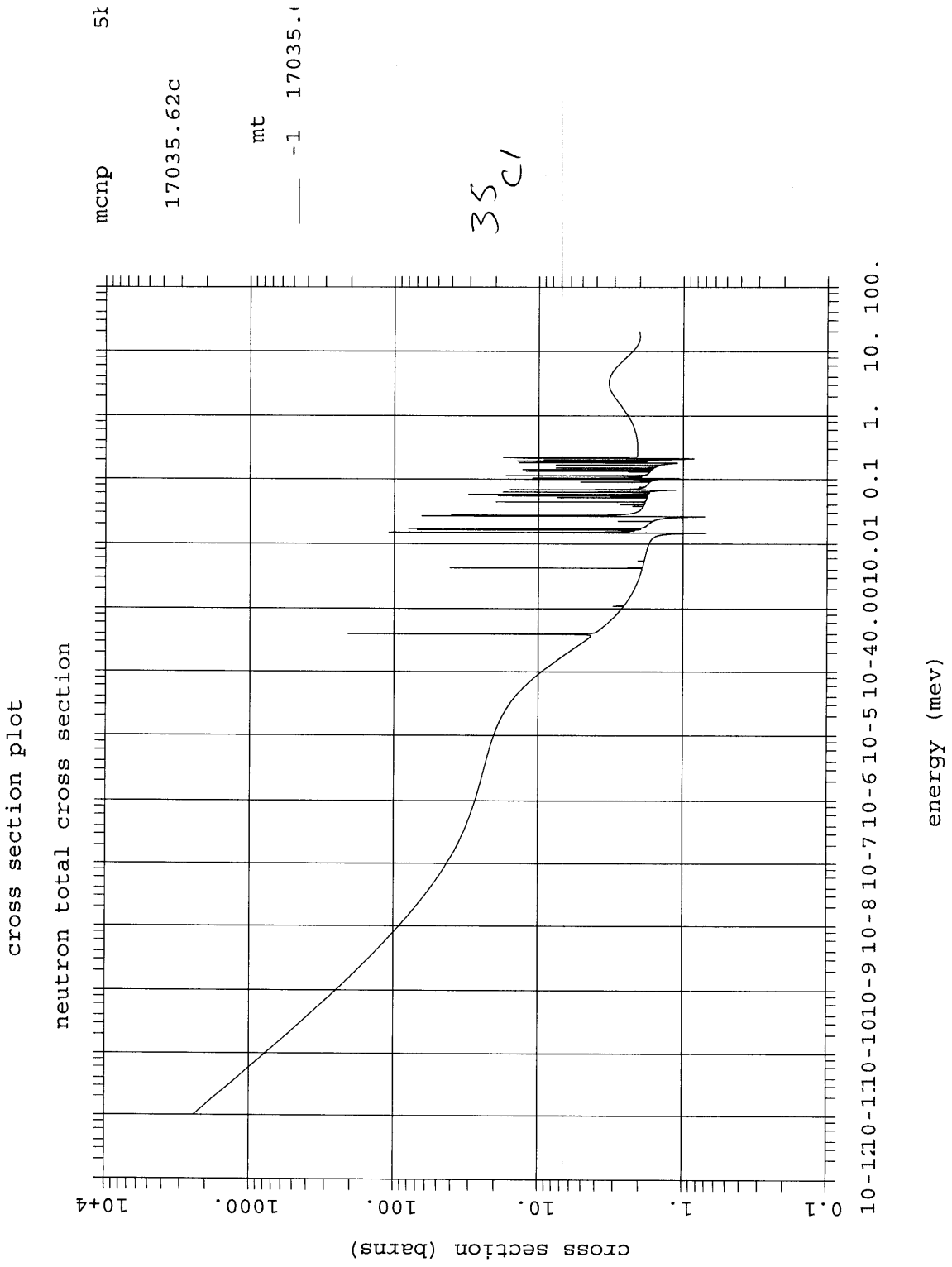
neutron total cross section
mcnp 5t
17000.51c
mt
-1 17000.!
Natural Chlorine
cross section (barns)
0.1
1.
10.
100.
1000.
10+4
10-11 10-10 10-9 10-8 10-7 10-6 10-5 10-4 0.0010.01 0.1 1. 10. 100.
energy (mev)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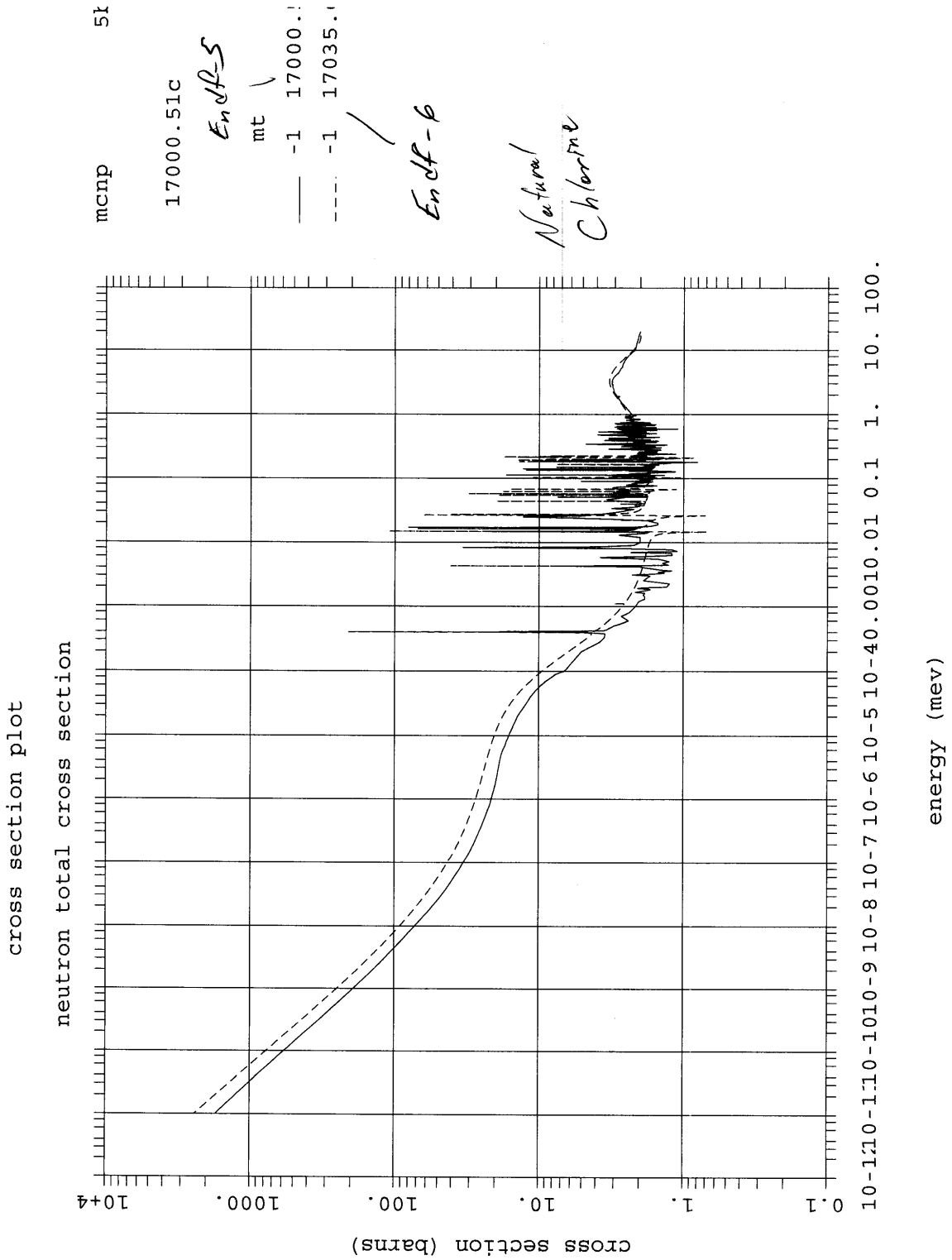
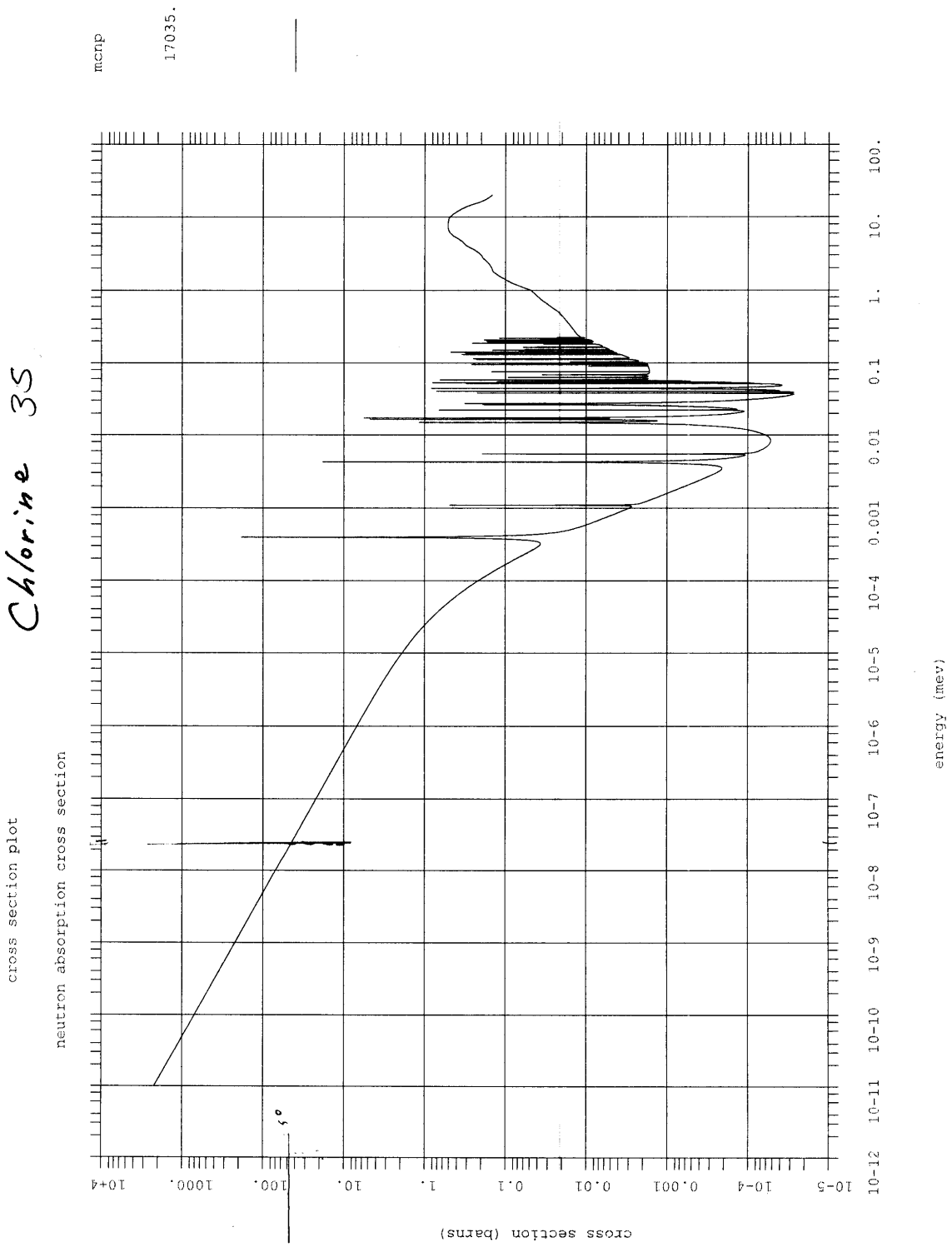
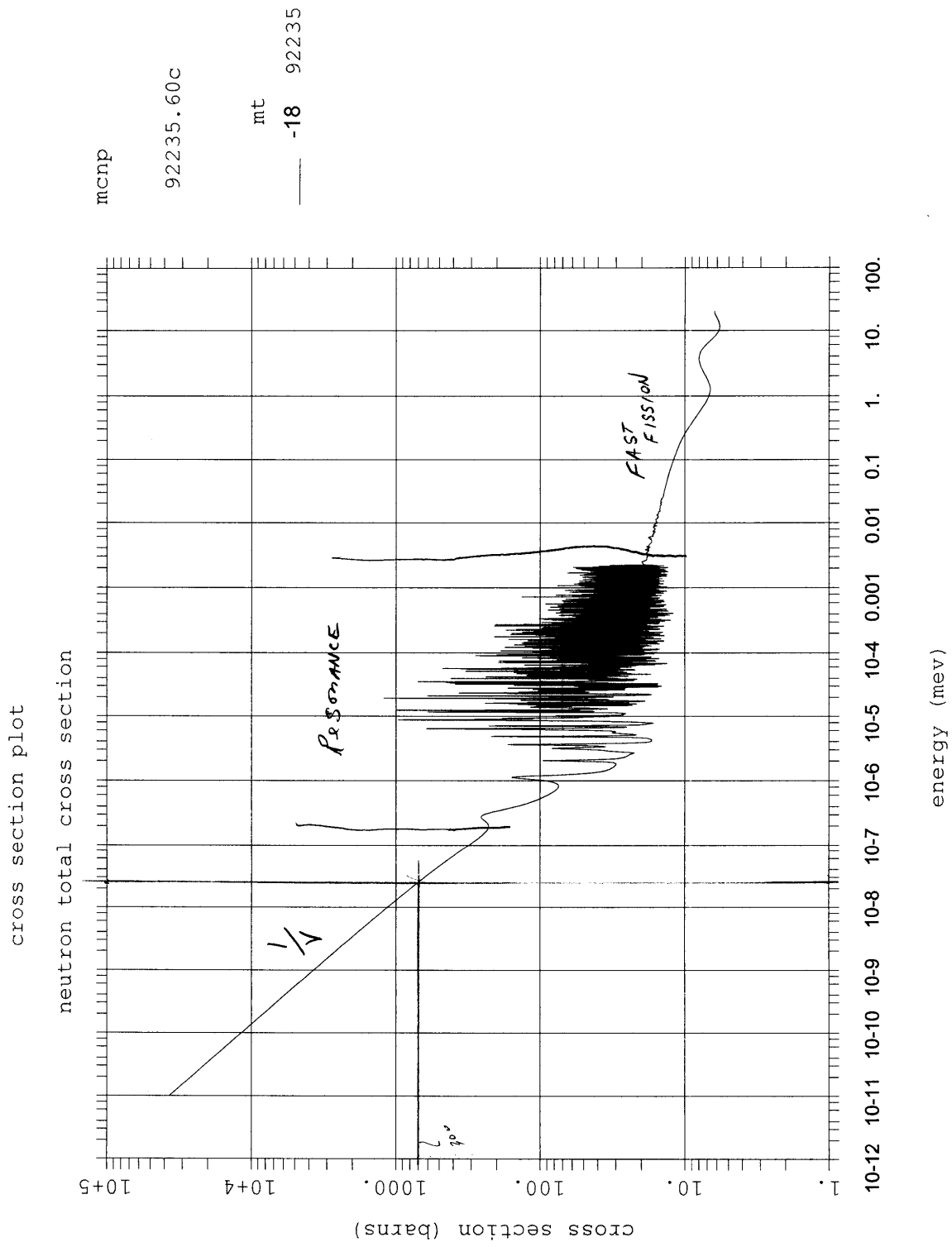
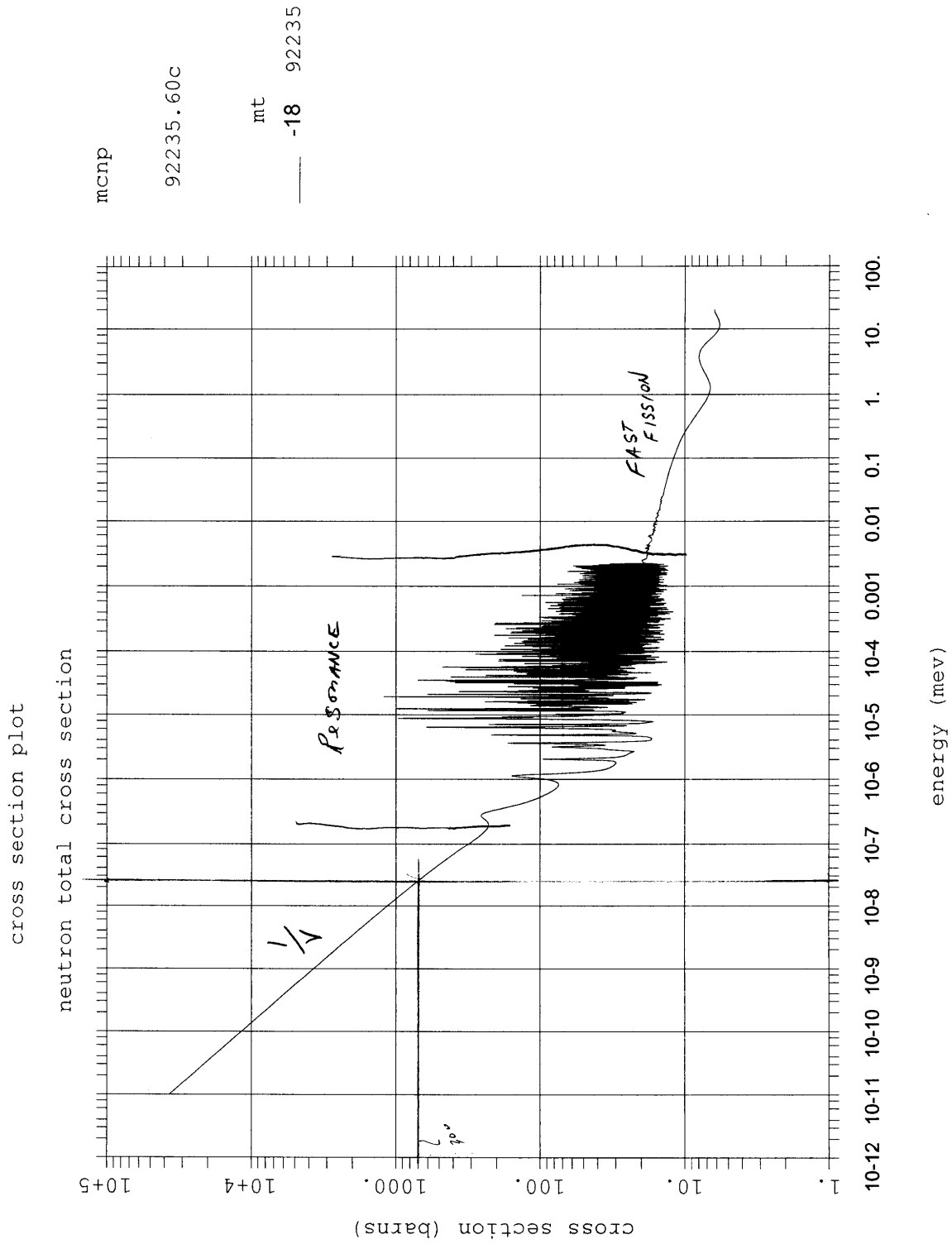
neutron total cross section
mcnp
5k
17035.62c
mt
-1 17035.
35Cl
cross section (barns)
10+4
1000.
100.
10.
1.
0.1
10-11 10-10 10-9 10-8 10-7 10-6 10-5 10-4 0.001 0.01 0.1 1. 10. 100.
energy (mev)


cross section plot
neutron total cross section
mcnp
17000.51c
mt
-1 17000.
-1 17035.
Endf-5
Endf-6
Natural Chlorine
cross section (barns)
10+4
1000.
100.
10.
1.
0.1
10-11 10-11 10-10 10-9 10-8 10-7 10-6 10-5 10-4 0.0010.01 0.1 1. 10. 100.
energy (mev)


cross section plot
neutron absorption cross section
Chlorine 35
mcnp
17035.
cross section (barns)
energy (mev)
10+4
1000.
100.
10.
1.
0.1
0.01
0.001
10-4
10-5
10-12
10-11
10-10
10-9
10-8
10-7
10-6
10-5
10-4
0.001
0.01
0.1
1.
10.
100.


mcnp
92235.60c
mt
-18 92235
cross section plot
neutron total cross section
1/v
RESONANCE
FAST FISSION
cross section (barns)
10+5
10+4
1000.
100.
10.
1.
energy (mev)
10-12
10-11
10-10
10-9
10-8
10-7
10-6
10-5
10-4
0.001
0.01
0.1
1.
10.
100.


mcnp
92235.60c
mt
-18 92235
cross section plot
neutron total cross section
1/v
RESONANCE
FAST FISSION
cross section (barns)
1.
10.
100.
1000.
10+4
10+5
energy (mev)
10-12
10-11
10-10
10-9
10-8
10-7
10-6
10-5
10-4
0.001
0.01
0.1
1.
10.
100.

SELF-ASSESSMENT FOR DOE CRITICALITY SAFETY PROGRAMS

OFFICE OF NUCLEAR AND FACILITY SAFETY,
EH-3 OFFICE OF ENVIRONMENT, SAFETY AND HEALTH

U.S. DEPARTMENT OF ENERGY

I. SCOPE

The Department of Energy (DOE) issued DOE P 450.5, *Line Environment, Safety and Health Oversight*, to set forth its expectations for line management environment, safety and health (ES&H) oversight. DOE line oversight and contractor self-assessments together ensure that field elements and contractors adequately implement the DOE safety management system. Both DOE and contractor line managers must acquire and maintain sufficient knowledge of program activities in order to make informed decisions on safety resources for these activities. The Department's line organizations have the following responsibilities:

- Develop ES&H performance objectives, measures, and expectations tied to DOE's strategic goals and objectives, as well as to performance goals and objectives of the safety management system elements.
- Develop contract performance measures and performance indicators that are linked to the DOE safety management system.
- Develop a high level of performance assurance that results in improved ES&H performance.

II. PURPOSE

The purpose of this document is to provide an assessment tool to evaluate the elements of the DOE nuclear criticality safety (NCS) oversight program. The requirements are based on the criteria outlined in DOE P 450.5.

III. ASSESSMENT REQUIREMENTS

DOE LINE ENVIRONMENT, SAFETY AND HEALTH OVERSIGHT

Criteria for the review of DOE criticality safety programs were extracted from DOE P 450.5.

Criterion: Elements of the DOE criticality safety program must be documented.

a. Are the responsibilities of the DOE NCS program manager clearly defined and understood?
b. Are the elements of a DOE NCS surveillance plan documented?

Criterion: DOE must acquire and maintain sufficient knowledge of program activities in order to make informed decisions on criticality safety resources for these activities**.**

a. Are routine meetings held with contractor NCS management?
b. Are periodic meetings held with DOE contractor operations management?
c. Does the DOE NCS program manager review budget requests made by contractor NCS management?
d. Does the DOE NCS program manager review budget requests made by contractor operations management?
e. Does the DOE NCS program manager have input to the DOE site budget process?

Criterion: DOE maintains operational awareness of contractor work activities, typically through DOE line managers and staff such as Facility Representatives and criticality safety subject matter experts.

a. Do the DOE NCS program manager and Facility Representatives work closely on NCS-related issues in the field?
b. Does the DOE NCS program manager routinely spend time in the field performing walkdowns and interacting with Operations?
c. Does the DOE NCS program manager review contractor occurrence reports related to criticality safety programs?

Criterion: DOE reviews performance against formally established criticality safety performance measures, performance indicators, and contractor self-assessments.

a. Have contractor NCS program performance measures been established? See Appendix A for examples.
b. Is progress on the performance measures routinely reported to DOE?
c. Are contractor NCS self-assessments reviewed by the DOE NCS program manager?
d. Does the NCS program manager provide reports and feedback on contractor self-assessments to senior DOE site management?

Criterion: DOE performs criticality safety reviews and assessments in support of required readiness assessments, operational readiness reviews, safety management system documentation and onsite verification reviews, and authorization basis documents including CSEs.

a. Does the DOE NCS program manager participate in readiness assessments, operational readiness reviews, and ISM reviews when necessary?
b. Does the DOE NCS program manager participate in the review and approval of facility NCS-related authorization basis documents (e.g., safety analysis reports, bases for interim operations, USQs, and TSRs)?
c. Does the DOE NCS program manager review a sample of contractor CSEs on a routine basis?

Criterion: DOE performs periodic appraisals of the contractor criticality safety program, including for-cause criticality safety reviews, as necessary.

a. Have facility criticality safety surveillances been incorporated into the field office assessment plan?
b. Are appraisals and reviews documented?
c. Are corrective actions tracked to closure?
d. Does the DOE NCS program manager perform assessments of the contractor criticality safety program in accordance with a documented plan?
e. Are outside DOE NCS subject matter experts occasionally utilized to assist with reviews to provide independent feedback?

Criterion: DOE has a designated focal point for coordinating criticality safety oversight activities.

a. Has the DOE field office designated a single NCS focal point (i.e., NCS program manager)?
b. Has the DOE NCS program manager been qualified by completing the requirements in the Federal NCS qualification standard?

c. Does the DOE NCS program manager routinely meet with an assistant field office manager responsible for NCS?
d. Does the DOE NCS program manager represent the single authority on NCS issues to the contractor?
e. Does the DOE NCS program manager represent the field office on the criticality safety coordinating team (CSCT)?