National Nuclear Security Administration

Nuclear Safety Specialist
Qualification Standard

DOE-STD-1183-2007

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The Functional Area Qualification Standard References Guides are developed to assist operators, maintenance personnel, and the technical staff in the acquisition of technical competence and qualification within the Technical Qualification Program (TQP).

Please direct your questions or comments related to this document to the Office of Leadership and Career Management, TQP Manager, NNSA Albuquerque Complex.

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ACRONYMS	
A	atomic mass number
AC	administrative control
AEA	Atomic Energy Act
AED	aerodynamic equivalent diameter
AEGL	acute exposure guideline level
AIB	accident investigation board
ALARA	as low as reasonably achievable
ALI	annual limit on intake
ALOHA	Areal Locations of Hazardous Atmospheres
ANSI	American National Standards Institute
ANS	American Nuclear Society
ARF	airborne release fraction
ARM	area radiation monitor
ARS	acute radiation syndrome
ASHRAE	American Society of Heating, Refrigeration, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing Materials
BE/A	binding energy per nucleon
BIO	basis for interim operation
Bq	becquerel
CAM	continuous air monitor
CCF	common cause failure
CFAST	Consolidated Fire Growth and Smoke Transport
Ci	curie
cm	centimeter
cm/h	centimeters per hour
CRD	contractor requirements document
CSDR	conceptual safety design report
CSE	criticality safety evaluation
CSP	criticality safety programs
DAC	derived air concentration
dB	decibels
DBA	design basis accident
DBE	design/evaluation basis earthquake
DEGADIS	Dense Gas Dispersion Model
DID	defense-in-depth
DNFSB	Defense Nuclear Facilities Safety Board
DoD	Department of Defense
DOE	Department of Energy
DOE-EH	Department of Energy, Office of Environmental, Safety, and Health
DOT	Department of Transportation

DP	differential pressure
DR	damage ratio
DSA	documented safety analysis
ECFA	events and causal factor analysis
EG	evaluation guideline
EO	Executive Order
EPA	Environmental Protection Agency
EPIcode	Emergency Prediction Information code
ERPG	emergency response planning guideline
ESD	event sequence diagram
ES&H	environment, safety, and health
ESS	evaluation of the safety of the situation
ET	event tree
FAQS	functional area qualification standard
FHA	fire hazards analysis
FMEA	failure mode and effects analysis
FPE	fire protection engineer
ft.	feet
FT	fault tree
FTA	fault tree analysis
g	gram
GI	gastrointestinal
GENII	Generalized Environmental Radiation Dosimetry Software System
Gy	gray
HA	hazard analysis
HAR	hazards analysis report
HASP	health and safety plan
HAZOP	hazard and operability study
HAZWOPER	Hazardous Waste Operations and Emergency Response
HEPA	high-efficiency particulate absorption
HVAC	heating, ventilation, air-conditioning
IAEA	International Atomic Energy Agency
IAQ	indoor air quality
IARC	International Agency for Research on Cancer
I&C	instrumentation and control
ICSSC	Interagency Committee on Seismic Safety in Construction
ICRP	International Commission on Radiological Protection
IE	initiating event
in/h	inch per hour
ISM	integrated safety management
IVRs	implementation verification reviews
JCO	justification for continued operation
kg	kilogram
km	kilometer

km/h	kilometer/hour
km/s	kilometers per second
kPa	kilopascals
KSA	knowledge, skill, and ability
LANL	Los Alamos National Laboratory
lb.	pound
LCO	limiting conditions for operations
LCS	limiting control settings
LDVT	linear variable differential transformer
LET	linear energy transfer
LPF	leak path factor
LVDT	linear variable differential transformer
m	meter
MAC	minimum accident of concern
MACC2	MELCOR Accident Consequence Code System 2
MAR	material at risk
MCE	maximum considered earthquake
MCS	minimal cut sets
MELCOR	Methods for Estimation of Leakage Consequences of Release
MERV	minimum efficiency reporting value
MeV	million electron volts
mg/m^3	milligrams per cubic meter
mm	millimeter
M&O	management and operating
MOI	maximally exposed offsite individual
MORT	management oversight risk tree
mph	miles per hour
MPFL	maximum possible fire loss
m/s	meter/second
M&TE	measuring and test equipment
NCS	nuclear criticality safety
NEO	nuclear explosive operation
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NNSA	National Nuclear Security Administration
NOAA	National Oceanic and Atmospheric Administration
NPH	natural phenomena hazard
NQA	National Quality Assurance
NRC	Nuclear Regulatory Commission
NSS	nuclear safety specialist
OH^-	hydroxyl radical
OL	operating limit
OSHA	Occupational Safety and Health Administration
PAAA	Price-Anderson Amendments Act
PC	performance category

PDSA	preliminary documented safety analysis
PEP	performance evaluation plan
PHA	preliminary hazards analysis
P&IDs	piping and instrument drawings
PISA	potential inadequacy of safety analyses
PL	Public Law
PM	particulate matter
ppm	parts per million
PRA	probabilistic risk assessment
PRM	process radiation monitoring
PSDR	preliminary safety design report
PSHA	probabilistic seismic hazard assessment
PSM	process safety management
PSO	public safety officer
Pu	plutonium
PuO_2	plutonium oxide
QA	quality assurance
QAP	quality assurance program
RCRA	Resource Conservation and Recovery Act
rem	roentgen equivalent in man/mammal
RF	respirable fraction
RISC	risk-informed safety class
RTD	resistance temperature detector
SA	specific activity
SAC	specific administrative control
SAR	safety analysis report
SARP	safety analysis report for packaging
SBAA	safety basis approval authority
SC	safety-class
SDOF	single degree of freedom
SEN	Secretary of Energy Notice
SER	safety evaluation report
S&H	safety and health
SL	safety limit
SME	subject matter expert
SQA	software quality assurance
sq. ft.	square feet
SR	surveillance requirements
SSCs	structures, systems, and components
SSQA	safety software quality assurance
SS	safety-significant
Sv	sievert
TC	thermocouple
TEDE	total effective dose equivalent
TEEL	temporary emergency exposure limit

TL	thermoluminescent dosimeter
TQ	threshold quantities
TQP	Technical Qualification Program
TSR	technical safety requirements
U	uranium
uCi	microcuries
uCi/ml	microcuries per milliliter
U.S.	United States
USQ	unreviewed safety question
USQD	unreviewed safety question determination
uXv	microsieverts
V&V	verification and validation
z	vertical height
z^p	power law

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 Nuclear Safety Specialist 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.

SCOPE

This reference guide has been developed to address the competency statements in the November 2007 edition of DOE-Standard (STD)-1183-2007, *Nuclear Safety Specialist Functional Area Qualification Standard*. The qualification standard for Nuclear Safety Specialist contains 31 competency statements.

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, abbreviations, and symbols is provided at the beginning of this document. It is recommended that the candidate review the list prior to proceeding with the competencies, as the acronyms, abbreviations, and symbols 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 August 2014. 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 TQP.

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.

This reference guide includes streaming videos to help bring the learning experience alive. To activate the video, click on any hyperlink under the video title. Note: Hyperlinks to video are shown in entirety, due to current limitations of eReaders.

ACKNOWLEDGEMENTS

Some of the references and responses to the KSA statements that are related to specific nuclear safety specialist (NSS) responsibilities were provided by a team of subject matter experts (SME) in the NNSA Albuquerque Complex.

TECHNICAL COMPETENCIES

1. **Nuclear safety specialists must demonstrate a familiarity level knowledge of the fission process.**

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

The following is taken from DOE-HDBK-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 (E_{exc}).

Critical Energy

The critical energy (E_{crit}) is the minimum excitation energy required for fission to occur.

Fissile Material

A fissile material is composed of nuclides for which fission is possible with neutrons of any energy level. What is 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 (U)-235, U-233, and plutonium (Pu)-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, 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. Some examples of nuclides requiring high energy neutrons to cause fission are thorium-232, U-238, and Pu-240.

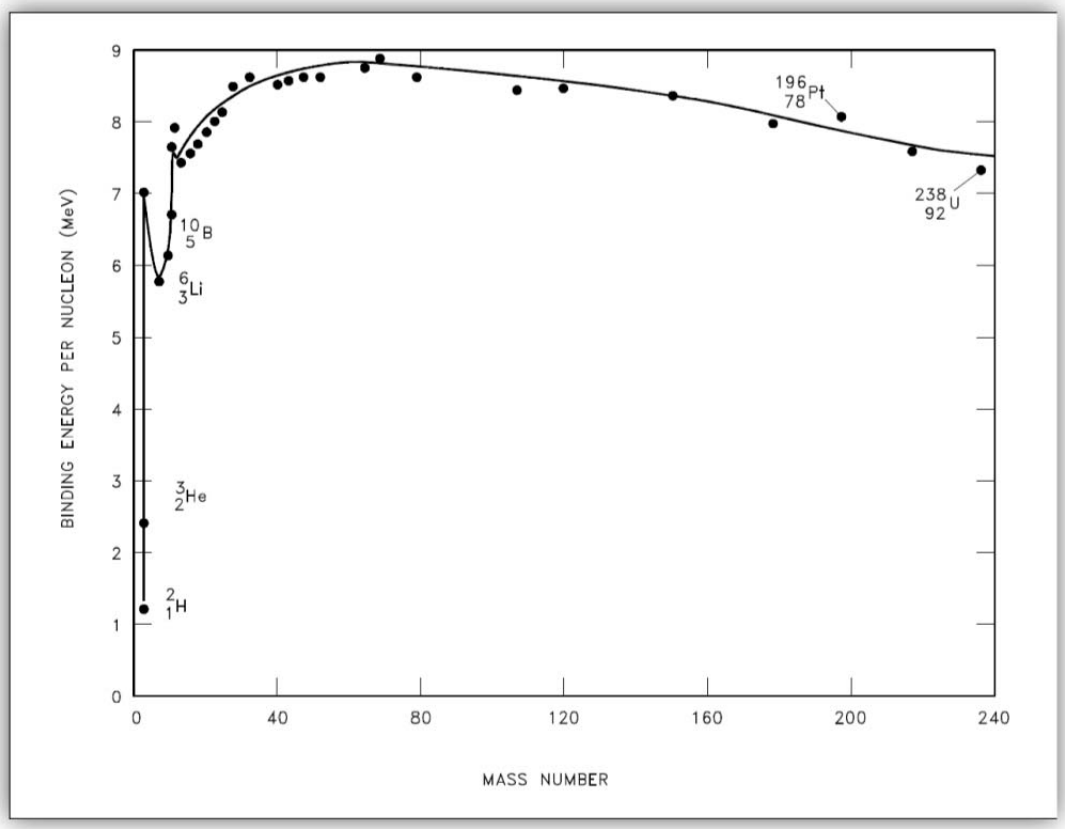
Fertile Material

All of the neutron absorption reactions that do not result in fission lead to the production of new nuclides through the process known as transmutation. These nuclides can, in turn, be transmuted again or may undergo radioactive decay to produce still different nuclides. The nuclides that are produced by this process are referred to as transmutation products. Because several of the fissile nuclides do not exist in nature, they can only be produced by nuclear reactions (transmutation). The target nuclei for such reactions are said to be fertile. Fertile materials are materials that can undergo transmutation to become fissile materials.

b. Describe the curve of binding energy per nucleon versus mass number and qualitatively describe the reasons for its shape.

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

As the number of particles in a nucleus increases, the total binding energy increases. The rate of increase, however, is not uniform. This lack of uniformity results in a variation in the amount of binding energy associated with each nucleon within the nucleus. This variation in the binding energy per nucleon (BE/A) is easily seen when the average BE/A is plotted versus atomic mass number (A), as shown in figure 1.



Source: DOE-HDBK-1019/1-93

Figure 1. Binding energy per nucleon vs. mass number

Figure 1 illustrates that as the atomic mass number increases, the binding energy per nucleon decreases for A>60. The BE/A curve reaches a maximum value of 8.79 million electron volts (MeV) at A = 56 and decreases to about 7.6 MeV for A = 238. The general shape of the BE/A curve can be explained using the general properties of nuclear forces. The nucleus is held together by very short-range attractive forces that exist between nucleons. The nucleus is being forced apart by long-range repulsive electrostatic (coulomb) forces that exist between all the protons in the nucleus.

As the atomic number and the atomic mass number increase, the repulsive electrostatic forces within the nucleus increase due to the greater number of protons in the heavy elements. To overcome this increased repulsion, the proportion of neutrons in the nucleus must increase to maintain stability. This increase in the neutron-to-proton ratio only partially compensates for the growing proton-proton repulsive force in the heavier, naturally occurring elements. Because the repulsive forces are increasing, less energy must be supplied, on the average, to remove a nucleon from the nucleus. The BE/A has decreased. The BE/A of a nucleus is an indication of its degree of stability. Generally, the more stable nuclides have higher BE/A than the less stable ones. The increase in the BE/A as the atomic mass number decreases from 260 to 60 is the primary reason for the energy liberation in the fission process. In addition, the increase in the BE/A as the atomic mass number increases from 1 to 60 is the reason for the energy liberation in the fusion process, which is the opposite reaction of fission.

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.

Video 1. The curve of binding energy
https://www.youtube.com/watch?v=YgZ-wDle_-I

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

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

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.

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

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

U-235 fissions with thermal neutrons because the binding energy released by the absorption of a neutron is greater than the critical energy for fission; therefore, U-235 is a fissile material. The binding energy released by U-238 absorbing a thermal neutron is less than the

critical energy, so additional energy must be possessed by the neutron for fission to be possible. Consequently, U-238 is a fissionable material.

Video 2. What is fission?
http://www.redorbit.com/news/video/education_1/1112777063/what-is-fission/

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

The following is taken from Wikipedia, *Nuclear Fission Product*, and DOE-HDBK-1019/1-93.

Nuclear fission products are the atomic fragments left after a large atomic nucleus fissions. Typically, a large nucleus like that of uranium fissions by splitting into two smaller nuclei, along with a few nuclei, a few neutrons, and releases of energy in the form of heat and gamma rays. The two smaller nuclei are the fission products.

The fission products produced by fission are themselves often radioactive, due to being relatively neutron-rich for their atomic number, and they undergo beta decay very soon, releasing additional energy in the form of beta particles, antineutrinos, and additional gamma rays. Fission events are thus normal sources of beta radiation and antineutrinos, even though these particles are not produced directly in the fission event itself.

The first beta decays are rapid and may release high energy beta particles or gamma radiation. However, as the fission products approach stable nuclear conditions, the last one or two decays may have a long half-life and release less energy. Fission products have half-lives of 90 years or less, except for seven long-lived fission products that have half-lives of 211,100 years and more. Therefore, the total radioactivity of a mixture of pure fission products decreases rapidly for the first several hundred years before stabilizing at a low level that changes little for hundreds of thousands of years.

Video 3. Particle physics—nuclear fission
http://www.youtube.com/watch?v=WCQwNWek3ow

2. Nuclear safety specialists must demonstrate a familiarity level knowledge of the principles and concepts for internal and external dosimetry, dose consequences, and the various methods to reduce exposure.

a. Define the following terms:
- **Committed effective dose equivalent**
- **Total effective dose equivalent**
- **Whole body**
- **Derived air concentration (DAC)**
- **Annual limit on intake (ALI)**
- **Tissue**
- **Weighting factors**
- **Stochastic effects**
- **Non-stochastic (deterministic) effects**

The following descriptions are taken from 10 CFR 20.1003, “Definitions,” unless otherwise stated.

Committed Effective Dose Equivalent

The committed effective dose equivalent is the sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues ($H_{E,50} = \sum W_T H_{T,50}$).

Total Effective Dose Equivalent

Total effective dose equivalent (TEDE) means the sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Whole Body

Whole body means, for purposes of external exposure, head, trunk (including male gonads), arms above the elbow, or legs above the knee.

Derived Air Concentration (DAC)

DAC means the concentration of a given radionuclide in air that if breathed by the reference man for a working year of 2,000 hours under conditions of light work (inhalation rate 1.2 cubic meters of air per hour), results in an intake of one ALI. DAC vales are given in 10 CFR 20.1001-20.2401, appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure, Effluent Concentrations, Concentrations for Release to Sewerage," table 1, "List of Elements," column 3, "Atomic No."

Annual Limit on Intake (ALI)

ALI means the derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI is the smaller value of intake of a given radionuclide in a year by the reference man that would result in a committed effective dose equivalent of 5 roentgen equivalent in man/mammals (rems) (0.05 sievert [Sv]) or a committed dose equivalent of 50 rems (0.5 Sv) to any individual organ or tissue. (ALI values for intake by ingestion and by inhalation of selected radionuclides are given in 10 CFR 20.1001-20.2401, appendix B, table 1, columns 1, "Name," and 2, "Atomic Symbol."

Tissue

The following is taken from DOE-HDBK-1122-99, module 2.04 (Archived).

Radiosensitive tissues include the following:

- Germinal (reproductive) cells of the ovary and testis (e.g., spermatogonia)
- Hematopoietic (bloodforming) tissues: red bone marrow, spleen, lymph nodes, thymus
- Basal cells of the skin
- Epithelium of the gastrointestinal(GI) tract (interstitial crypt cells)

Radioresistant tissues include the following:

- Bone
- Liver
- Kidney
- Cartilage
- Muscle
- Nervous tissue

Radiosensitivity not only differs from one cell or tissue to another but between individuals and genders.

Weighting Factors

Weighting factor, for an organ or tissue, is the proportion of the risk of stochastic effects resulting from irradiation of that organ or tissue to the total risk of stochastic effects when the whole body is irradiated uniformly.

Stochastic Effects

Stochastic effects mean health effects that occur randomly and for which the probability of the effect occurring, rather than its severity, is assumed to be a linear function of dose without threshold. Hereditary effects and cancer incidence are example of stochastic effects.

Non-stochastic (Deterministic) Effects

Non-stochastic effect means health effects, the severity of which varies with the dose and for which a threshold is believed to exist. Radiation-induced cataract formation is an example of a non-stochastic effect (also called a deterministic effect).

Video 4. Stochastic and non-stochastic effects
http://www.youtube.com/watch?v=m-W_9g9Nnuk

b. Discuss International Commission on Radiological Protection (ICRP) publications 26, 30, 60, 68, 71, and 72 as they relate to dose conversion factors and consequence analysis in hazard categorization and accident analysis. [Note: ICRPs 26 and 60 have been superseded by ICRP 103.]

ICRP 103, The 2007 Recommendations of the International Commission on Radiological Protection

These recommendations update the radiation and tissue weighting factors in the quantities equivalent and effective dose and update the radiation detriment, based on the latest available scientific information of the biology and physics of radiation exposure. They maintain the commission's three fundamental principles of radiological protection, namely justification, optimization, and the application of dose limits, clarifying how they apply to radiation sources delivering exposure and to individuals receiving exposure.

ICRP 30, Limits for Intakes of Radionuclides by Workers

This is the first of a series of reports recommending ALIs of radionuclides by workers. It includes the main text for the whole series and data on twenty-one elements having radioisotopes that are of considerable importance in radiological protection. The actual ALI values in this publication have become obsolete with the newer dosimetry and dose limits of

ICRP 103, and the dose coefficients in ICRPs 68, *Dose Coefficients for Intakes of Radionuclides by Workers*; 69, *Age-Dependent Doses to Members of the Public from Intake of Radionuclides—Part 3 Ingestion Dose Coefficients*; 71, *Age-Dependent Doses to Members of the Public from Intake of Radionuclides—Part 4 Inhalation Dose Coefficients*; and 72, *Age-Dependent Doses to the Members of the Public from Intake of Radionuclides—Part 5 Compilation of Ingestion and Inhalation Coefficients*, should be used to determine ALIs. However, the vast body of biokinetic information in ICRP 30 still forms the basis of much of the calculations underlying those later reports.

ICRP 68, Dose Coefficients for Intakes of Radionuclides by Workers

The main aim of this report is to give values of dose coefficients for workers using the new model of the respiratory tract taken from the superseded ICRP 61, *Annuals Limits on Intake of Radionuclides by Workers Based on the 1990 Recommendations*.

ICRP 71, Age-Dependent Doses to Members of the Public from Intake of Radionuclides—Part 4 Inhalation Dose Coefficients

An ongoing objective of ICRP is to evaluate dose coefficients (doses per unit intake) for members of the public. The purpose of ICRP 71 is to provide updated inhalation dose coefficients for selected radioisotopes of hydrogen, carbon, sulphur, calcium, iron, cobalt, nickel, zinc, selenium, strontium, zirconium, niobium, molybdenum, technetium, ruthenium, silver, antimony, tellurium, iodine, cesium, barium, cerium, lead, polonium, radium, thorium, uranium, neptunium, plutonium, americium, and curium. Age-dependent biokinetic models for calcium, curium, and for decay products formed following the intake of lead, radium, tellurium, thorium, and uranium are provided in annexes.

ICRP 72, Age-Dependent Doses to the Members of the Public from Intake of Radionuclides—Part 5 Compilation of Ingestion and Inhalation Coefficients

The purpose of ICRP 72 is to summarize data on age-dependent committed effective dose coefficients for members of the public from intakes by ingestion and inhalation of radioisotopes of the 91 elements described in ICRP 56, *Age-Dependent Doses to Members of the Public from Intake of Radionuclides—Part 1*; 67, *Age-Dependent Doses to Members of the Public from Intake of Radionuclides—Part 2 Ingestion Dose Coefficients*; 68; 69, and 71. These dose coefficients have been adopted in the International Atomic Energy Agency (IAEA) publication on international basic safety standards for protection against ionizing radiation, and in the Euratom Directive. The report does not give committed equivalent dose coefficients to tissues and organs. This report will be useful to operational health physicists and to regulatory and advisory bodies responsible for radiation protection.

c. Describe the following aspects of dose reduction:

- **Time**
- **Distance**
- **Shielding**
- **Inverse square law**
- **As low as reasonably achievable (ALARA)**

Time

The following is taken from Muldoon's Introduction to Health Physics Page, *Time, Distance, and Shielding*.

The dose from a radiation source is directly proportional to the amount of time spent in the radiation field. Minimize time near a source of radiation by planning ahead.

Distance

The following is taken from Muldoon's Introduction to Health Physics Page, *Time, Distance, and Shielding*.

Increase the distance from a radiation source, to decrease the radiation dose received. By doubling the distance from a source, the dose from the source is four times less.

Shielding

The following is taken from DOE-HDBK-1130-2008.

Shielding is attenuating material used to reduce the transmission of radiation. The two general types of shielding are primary and secondary. Primary shielding is material sufficient to attenuate the useful beam to the required level. Secondary shielding is material sufficient to attenuate stray radiation to the required level.

Video 5. Radiation safety
http://www.youtube.com/watch?v=ZniP9fkHC2o

Inverse Square Law

The following is taken from DOE-HDBK-1130-2008.

Three basic ways to reduce external doses are to

1. minimize time
2. maximize distance
3. use shielding

Minimize time near a source of radiation by planning ahead. Increase distance by moving away from the source of radiation whenever possible. The dose from X-ray sources is inversely proportional to the square of the distance. This is called the inverse square law, that is, when the distance is doubled, the dose is reduced to one-fourth of the original value.

Proper facility design uses the amount and type of shielding appropriate for the radiation hazard. Lead, concrete, and steel are effective in shielding against X-rays.

Video 6. Inverse square law
http://www.youtube.com/watch?v=F-xNMdIXJIs

As Low As Reasonably Achievable (ALARA)

The following is taken from DOE-HDBK-1130-2008.

Because the effects of chronic doses of low levels of ionizing radiation are not precisely known, assume there is some risk for any dose. The ALARA principle is to keep radiation dose as low as reasonably achievable, considering economic and social constraints.

Video 7. ALARA principle
http://www.youtube.com/watch?v=Psy_VWTKIY0

d. Describe the hierarchy of controls; engineered, administrative, and personnel protective equipment objectives of 10 CFR 835 for controlling personnel exposure from external sources of radiation in areas of continuous occupational occupancy.

The following is taken from DOE-HDBK-1141-2008.

The minimization and control of internal dose should be conducted in accordance with the following hierarchy of controls:

- Engineered controls, including containment of radioactive material at the source wherever applicable, should be the primary method of minimizing airborne radioactivity and internal dose to workers. Engineered controls are devices such as gloveboxes, glove bags, portable filtration units, and containment tents. They should be used to prevent worker inhalation of radionuclides. Portable and fixed/permanent shielding using dense materials (lead) or portable plastic interlocking fluid filled containers are engineered features, used to minimize external radiation dose. The use of these devices reduces the spread of contamination, cleanup time, and decontamination costs. These measures help maintain doses ALARA. In addition, they can reduce the need for respirators and the impact on work in nearby areas. Engineered controls should be used in accordance with technical instructions, proper training, and effective administrative controls (ACs). Site-specific manuals should contain generic instructions on the design, controls, training, and use of engineered controls.
- AC including access restrictions and the use of specific work practices designed to minimize airborne contamination should be used as the secondary method to minimize worker internal dose.
- Only when engineered and ACs have been applied and the potential for airborne radioactivity still exists, should personnel protective equipment, including the use of respiratory protection be considered. The selection of respiratory protection equipment should include consideration of worker safety, comfort, and efficiency. The use of positive pressure respiratory protection devices is recommended wherever

practicable to alleviate fatigue and increase comfort. Use of respiratory protection should be considered under the following conditions:

- Entry into posted airborne radioactivity areas
- During breach of contaminated systems or components
- Work in areas or on equipment with removable contamination levels greater than 100 times the values in DOE-STD-1098-2008, *Radiological Control*, table 2-2, "Summary of Surface Contamination Values (see 10 CFR 835, appendix D)"

3. Nuclear safety specialists must demonstrate a familiarity level knowledge of criticality control, safety parameters, alarm systems, and poisons.

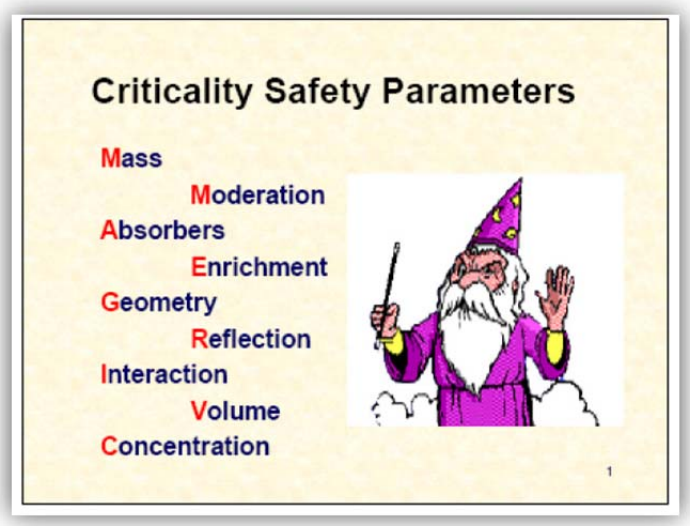
a. Discuss the effects and applications of the following factors relevant to criticality safety of operations:

- **Mass**
- **Shape**
- **Interaction and separation**
- **Moderation**
- **Reflection**
- **Concentration**
- **Volume**
- **Density**
- **Neutron absorbers**
- **Heterogeneity**
- **Enrichment**

Mass

The following is taken from Reference Handbook EG&G Rocky Flats, *Nuclear Criticality*, revision 3.

For a nuclear criticality to occur, a sufficient mass of fissionable material must be present to support a chain reaction. The smallest mass of fissionable material that supports a self-sustaining chain reaction under a given set of conditions is known as the critical mass. For example, the critical mass of a sphere of 95 percent pure Pu-239 metal in a bare condition is 9.8 kilograms (kg). Bare condition means that the fissionable material has nothing surrounding it to absorb or reflect escaping neutrons back into the mass. Keeping fissionable material in a bare condition is not realistic in the workplace; therefore, it is necessary to examine other conditions that affect the amount of fissionable material needed to reach a critical mass.



Source: NNSA Service Center Safety Department, local training document mnemonic

Figure 2. Minimum efficiency reporting value (MERV)

Shape

The following is taken from Reference Handbook EG&G Rocky Flats, *Nuclear Criticality*, revision 3.

Another effective way to control criticality is to control the shape of the fissionable material. Nuclear criticality occurs if the neutron production rate is equal to, or exceeds, the neutron loss rate. Therefore, unplanned criticality can be prevented by making the loss rate large when compared to the production rate.

One way to avoid accidental criticality is to select an appropriate container. Table 1 shows how surface area can vary in different shapes while volume remains constant.

Table 1. Surface areas for sphere, cylinder, and disk

Shape	Volume (liters)	Surface Area (cm^2) [b]	Surface Area Compared to Sphere
Sphere	1.0	484	1.00
Cylinder (H/D[a]=1/1)	1.0	554	1.14
Long, thin cylinder (H/D=5/1)	1.0	694	1.43
Disk (H/D=1/5)	1.0	755	1.56

Notes:

[a]H/D: height/diameter ratio

[b]cm: centimeters

Source: Reference Handbook, EG&G Rocky Flats, Nuclear Criticality, Revision 3

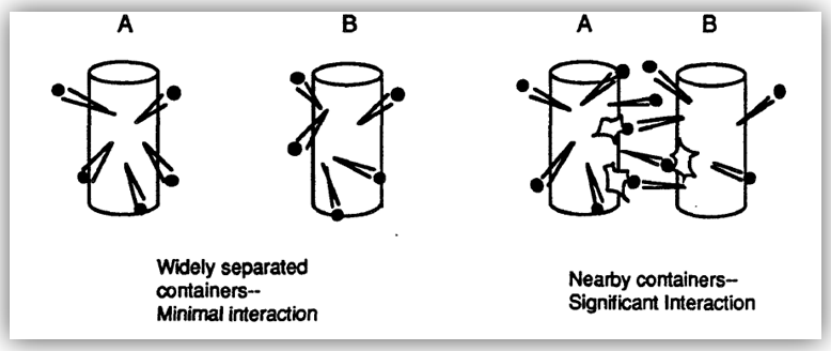
To help avoid accidental criticality, the best container shapes are flat disks or thin cylinders. The sphere is the worst of all possible shapes because the sphere has the smallest surface area for a given volume.

A fissionable solution of a certain concentration and volume can be safely stored in a long, thin cylindrical container; however, when the solution is transferred to a spherical container, an accidental criticality may occur. The solution has not changed in concentration or volume; the only change is in container shape. The solution, safe in the first container, becomes critical in the second container because of the reduction in the neutron leakage rate caused by the reduction in surface area. The increased number of neutrons available for the fission reaction, as a result of the change in shape, was sufficient to permit a chain reaction.

Interaction and Separation

The following is taken from Reference Handbook EG&G Rocky Flats, *Nuclear Criticality*, revision 3.

In figure 3, if two containers (A&B) of fissionable material are close enough to each other, a certain number of neutrons leak out of container A and enters container B and causes fission. This process is called interaction. Interaction decreases the neutron loss rate and increases the neutron production rate.

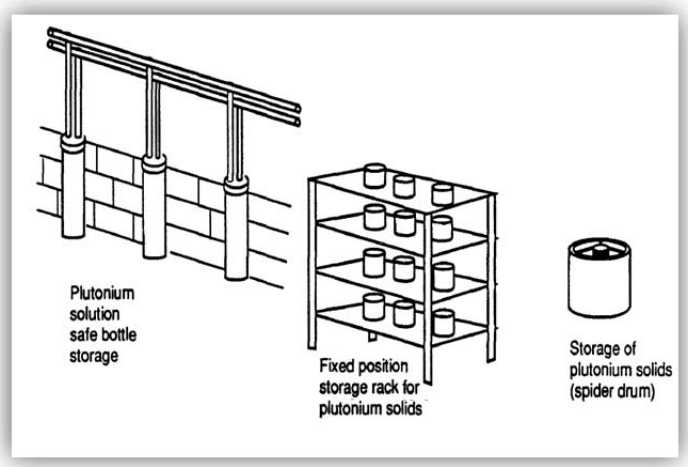


Source: Reference Handbook, EG&G Rocky Flats, Nuclear Criticality, Revision 3

Figure 3. Interaction as a function of distance

If container A is left by itself without a reflector, all of the neutrons leaking from the container are lost and cause no further fission. The two containers form a more reactive system together than when they are separated. The amount of interaction between the two containers depends on the distance between them. The farther apart the two containers are spaced, the less they will interact. As the containers come closer together, interaction increases and so does the reactivity of the system.

There are methods of separating fissionable materials so that interaction is reduced. Bottles can be physically separated by engineered features, or solids can be separated by a rack arrangement. (See figure 4.)



Source: Reference Handbook, EG&G Rocky Flats, Nuclear Criticality, Revision 3

Figure 4. Fissionable material separation

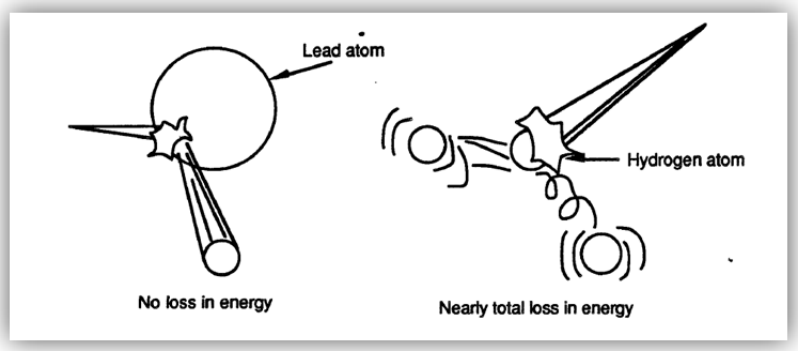
Moderation

The following is taken from Reference Handbook EG&G Rocky Flats, *Nuclear Criticality*, revision 3.

The energy (speed) of a neutron affects its ability to cause fission. In most fissionable material, a slow neutron is much more likely to cause fission than a fast neutron. All neutrons produced by fission are fast neutrons, with a speed of about 30,000,000 miles per hour (mph) when first produced, but neutrons can be slowed down by colliding with other nuclei within the material. This process of slowing down neutrons is called moderation, and any substance, wet or dry, used for this purpose is called a moderator.

When a neutron collides with the nucleus of an atom, the nucleus recoils taking with it some of the neutron's energy. The neutron then bounces away in a new direction, but with less speed than it had before the collision. Thus, the neutron is moderated or slowed down. How much of the neutron's speed is lost in a collision depends on the mass of the nucleus with which it collides.

Figure 5 contains an illustration of the collision of neutrons with nuclei of lead and hydrogen atoms. The lead nucleus has about 207 times more mass than the neutron, so the situation is similar to a BB colliding with a bowling ball. The bowling ball (lead) will not move very much, and the BB (neutron) will not lose much speed per collision. Lead and other heavy elements do not make effective moderators.



Source: Reference Handbook, EG&G Rocky Flats, Nuclear Criticality, Revision 3

Figure 5. Neutron collision with lead and hydrogen nuclei

When a neutron collides with the nucleus of a hydrogen atom, with about the same mass as a neutron, the situation is similar to two billiard balls colliding. On the average, the neutron will lose half its energy in a single collision with a hydrogen atom. Thus, hydrogen and materials containing hydrogen, such as water, oil, paraffin, and people, make effective moderators. The presence of an effective moderator can substantially reduce the amount of fissionable material needed to achieve critical mass. For example, the minimum critical mass for a solution of Pu containing 1,000 grams (g) Pu per liter is approximately 4,500 g of Pu. If a sufficient amount of moderator (water) were added to the 1,000 g per liter solution to create a solution with a concentration of 100 g per liter, the minimum critical mass would be reduced to approximately 850 g.

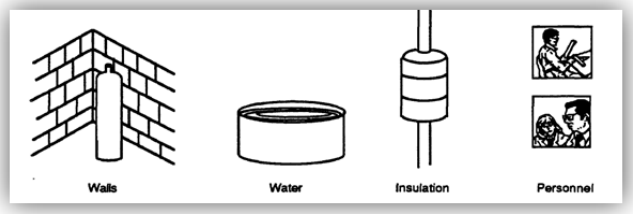
Reflection

The following is taken from Reference Handbook EG&G Rocky Flats, *Nuclear Criticality*, revision 3.

Because critical mass depends on the neutron leakage rate, the characteristics of the substance surrounding the fissionable material must be considered. If the surroundings deflect escaping neutrons back into the fissionable material, less fissionable material is necessary to reach a critical mass.

Consider a bare sphere of Pu-239 metal. Neutrons will escape from the surface of the bare sphere, be lost to the system, and cause no further fissions. If the sphere is surrounded by water, some of the neutrons that leaked out will hit atoms in the water and bounce back into the sphere. These neutrons can cause fissions.

This process is called reflection, and the substance used is called a reflector (see figure 6 for examples). Reflection is caused by the same scattering reaction mentioned previously. A certain number of neutrons leak out of the fissionable material, are scattered by atoms in the reflector, and consequently bounce back into the fissionable material.



Source: Reference Handbook, EG&G Rocky Flats, Nuclear Criticality, Revision 3

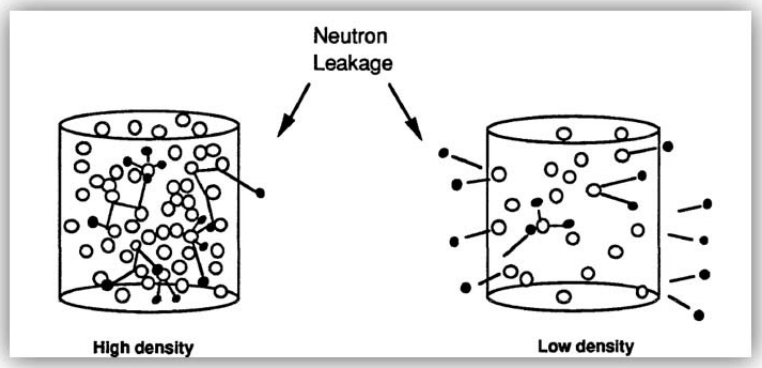
Figure 6. Examples of reflectors

All effective moderators are also effective reflectors. Water, oil, polyethylene, concrete, paraffin, and any other hydrogenous material can make an effective reflector. The human body contains a large amount of water and is also an effective reflector. However, some reflectors are not effective moderators.

Concentration

The following is taken from Reference Handbook EG&G Rocky Flats, *Nuclear Criticality*, revision 3.

Density (concentration) refers to the number of atoms within a unit volume. As the density increases, the atoms are packed closer together. Higher concentrations increase the chance of a neutron causing fission because more fissionable material is in the same space. High density and low density are compared in figure 7.



Source: Reference Handbook, EG&G Rocky Flats, Nuclear Criticality, Revision 3

Figure 7. High density vs. low density

For example, consider different concentrations of dry homogeneous Pu oxide (PuO_2) that is a powder. At a density of 2 g of PuO_2 per cubic centimeter (cm), the critical mass is about 800 kg. When the density is increased to 5 g of PuO_2 per cubic cm, the critical mass decreases to about 128 kg. Clearly, changes in density affect critical mass.

Volume

The following is taken from Reference Handbook EG&G Rocky Flats, *Nuclear Criticality*, revision 3.

The volume of fissionable material must be controlled in much the same way as is mass. Because fissionable material is often stored in solution, a certain volume of fissionable material may correspond to an equivalent critical mass. Containers of limited volume are used to ensure that the accumulation of fissionable material solutions does not approach a condition of criticality.

In areas such as a glovebox, where an accumulation of solution (fissile or non-fissile) presents a criticality hazard, criticality drains are typically used. Criticality drains allow solutions produced by process equipment malfunction to accumulate to a predetermined depth. Once the maximum depth is reached, solutions will flow out of the criticality drain and onto the process area floor, or in some cases, to an engineered collection system.

Density

Refer to the information under concentration for information related to density.

Neutron Absorbers

The following is taken from Reference Handbook EG&G Rocky Flats, *Nuclear Criticality*, revision 3.

Certain isotopes readily absorb neutrons. Once absorbed, these neutrons cannot cause further fissions. Such material, called neutron poison, is used to decrease the neutron population.

Two elements frequently used as neutron poisons are cadmium and boron. When sufficient neutron poison material is added to fissionable material, enough neutrons are captured (absorbed) to prevent a criticality accident. The most common form of neutron poison is the boron-containing Pyrex glass Raschig ring used in large volume storage tanks. (See figure 8.)

Source: Reference Handbook, EG&G Rocky Flats, Nuclear Criticality, Revision 3

Figure 8. Raschig ring

Heterogeneity

The following is taken from INTECH, *Nanoindentation Based Analysis of Heterogeneous Structural Materials*.

Structural materials exhibit several types of heterogeneity. The first type of heterogeneity comes from mixing of components that do not react chemically in the composite-like sand, fibers, and other additives. Such heterogeneity is usually known in advance and is given by the mixing proportions. The second type of heterogeneity comes from chemical reactions that are evolving after the mixing of basic components. As a result of these reactions, new phases are produced. Formation of the new phases in the structural composites include fully or partly reacted matrix, unreacted grains of the raw material, interfacial zones with different

chemical and mechanical properties, and porosity. Structural materials based on cement or waste materials usually include both types of the heterogeneity.

Video 8. Understanding heterogeneity
http://www.youtube.com/watch?v=eKGBvB7r5nI

Enrichment

The following is taken from the International Agency for Research on Cancer (IARC), *IARC Monographs on the Evaluation of Carcinogenic Risks to Humans*, volume 78.

Enrichment is an isotopic separation process by which the relative abundance of the isotopes of a given element is altered; thus, producing a form of the element that has been enriched in one or more isotopes and depleted in others. In U enrichment, the percentage of U-235 in natural U is increased from 0.7 percent to >90 percent in a gaseous diffusion process based on the different thermal velocities of the constituents of natural uranium (U-234, U-235, and U-238).

Video 9. How to enrich uranium
http://www.youtube.com/watch?v=pl_E3aIL7G0

b. Discuss the influence of the presence of non-fissionable materials mixed with, or in contact with, fissionable material on nuclear criticality safety.

The following is taken from Wikipedia, *Fissile vs Fissionable*.

Fissile is distinct from "fissionable." A nuclide capable of undergoing fission (even with a low probability) after capturing a high energy neutron is referred to as "fissionable." A fissionable nuclide that can be induced to fission with low-energy thermal neutrons with a high probability is referred to as "fissile." Although the terms were formerly synonymous, fissionable materials include those (such as U-238) that can be fissioned only with high-energy neutrons. As a result, fissile materials (such as U-235) are a subset of fissionable materials.

U-235 fissions with low-energy thermal neutrons because the binding energy resulting from the absorption of a neutron is greater than the critical energy required for fission; therefore, U-235 is a fissile material. By contrast, the binding energy released by U-238 absorbing a thermal neutron is less than the critical energy, so the neutron must possess additional energy for fission to be possible. Consequently, U-238 is a fissionable material but not a fissile material.

An alternative definition defines fissile nuclides as those nuclides that can be made to undergo nuclear fission and produce neutrons from such fission that can sustain a nuclear chain reaction in the correct setting. Under this definition, the only nuclides that are fissionable are those nuclides that can be made to undergo nuclear fission but produce insufficient neutrons, in either energy or number, to sustain a nuclear chain reaction. As such, while all fissile isotopes are fissionable, not all fissionable isotopes are fissile. In the arms control context, particularly in proposals for a Fissile Material Cutoff Treaty, the term

"fissile" is often used to describe materials that can be used in the fission primary of a nuclear weapon. These are materials that sustain an explosive fast fission chain reaction.

Under all definitions given above, U-238 is fissionable, but because it cannot sustain a neutron chain reaction, it is not fissile. Neutrons produced by fission of U-238 have lower energies than the original neutron (they behave as in an inelastic scattering), usually below one MeV (i.e., a speed of about 14,000 kilometers per second [km/s]), the fission threshold to cause subsequent fission of U-238, so fission of U-238 does not sustain a nuclear chain reaction.

c. Discuss the concept of contingencies for checking the validity of criticality safety limits (SL).

The following is taken from DOE-STD-1134-99.

American National Standards Institute (ANSI)/American Nuclear Society (ANS)-8.1, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, appendix A, "Summary Statements of the Criteria," contains a list of typical scenarios to consider when reviewing the contingency analysis for a particular evaluation. The reviewer must be familiar with the operation being evaluated to perform an adequate review. Without such knowledge, no decision can be made relative to whether postulated abnormal events are "anticipated," "unlikely," or "incredible," or if any credible contingencies have been omitted.

Each contingency that could lead to criticality should be shown to be unlikely, independent, and non-concurrent with other contingencies that could lead to criticality. If this cannot be done, then the contingency under evaluation becomes part of the normal operating conditions (i.e., an anticipated event).

Two simplifications are possible here. Criticality scenarios that are deemed "incredible" and those that are not physically possible need no contingency analysis. The reviewer's task in each of these two cases is to decide whether the arguments against criticality occurring are sound. The reviewer should take care to note the definition of "contingency" and "credible" in DOE-STD-3007-2007, *Guidelines for Preparing Criticality Safety Evaluations at Department of Energy Nonreactor Nuclear Facilities*. If a quantitative probabilistic risk assessment (PRA) is utilized in the criticality safety evaluation (CSE) despite the qualitative definition of "credible," the reviewer should rigorously scrutinize any calculations and assumptions leading to extremely small probabilities. In the more likely situation, the experienced, professional judgment of engineering and operational personnel will be the basis for this argument. The bases for these judgments should be carefully documented in the nuclear criticality safety (NCS) limits and requirements for NCS controls.

In other cases, where criticality is possible, the reviewer's task is to ensure that no credible single failure can result in the potential for a criticality accident. To accomplish this, all credible failure modes known to the reviewer that are applicable to the process should be bounded by the analysis or adequate barriers to its occurrence must be in place. The reviewer should ask, "How bad can the situation credibly get? If it gets that bad, will it remain subcritical?" and ensure that the documented contingency analysis considers the scenario. This contingency discussion must contain a clear description of the process upsets (i.e.,

contingencies) considered by the NCS analyst. The contingencies should be specific enough to provide a definitive boundary to the process upset. For example, rather than stating a mass contingency as "overbatch," state "double-batching" if this has been determined to be the maximum credible overbatch. The former means any overbatch no matter how small would have to be an "unlikely" event. The latter concludes that while a small overbatch would be an "anticipated" event, double-batching would be an "unlikely" event. The reviewer need not document all conceivable abnormal pathways or scenarios, only those deemed reasonable and credible.

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

Criticality Accident
The following is taken from Wikipedia, *Criticality Accident*.

A criticality accident is an uncontrolled nuclear chain reaction. It is sometimes referred to as a critical excursion or a critical power excursion and represents the unintentional assembly of a critical mass of a given fissile material, such as enriched uranium or Pu, in an unprotected environment.

Minimum Accident of Concern (MAC)
The following is taken from DOE PNNL-18348, *Master's Thesis—Criticality Alarm System Design Guide with Accompanying Alarm System Development for the Radioisotope Production L.*

The MAC represents the smallest accident in terms of fission yield and dose rate that the criticality alarm system must detect. ANSI/ANS-8.3-1997, *Criticality Accident Alarm System*, defines the MAC as one that delivers "the equivalent of an absorbed dose rate in free air of 0.2 gray/minute (Gy/min) (20 radiation absorbed dose/minute [rad/min]) at 2 meters from the reacting material."

Process Area
The following is taken from ANSI/ANS-8.3.

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

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

The following is taken from the Canadian Nuclear Safety Commission, GD-327, *Guidance for Nuclear Criticality Safety*.

Installation
A criticality alarm system 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 will be documented.

If criticality accidents of lesser magnitude than the MAC given are of concern, then other detection methods (e.g., audible personnel dosimetry) should be considered.

Coverage
A criticality alarm system will be installed in areas where
- inadvertent criticality can occur
- excessive radiation dose to personnel is credible should the inadvertent criticality occur

Detection
In areas in which criticality alarm coverage is required, a means will be provided to detect a criticality accident and to signal that prompt protective action is required. A basic consideration in the design of a criticality accident alarm system is the definition of the lower magnitude of the event size to be detected, termed the MAC.

The system will be designed to produce the criticality alarm signal within ½ second of detector recognition of a criticality accident. The alarm trip point will be set low enough to detect the MAC and high enough to minimize the probability of an alarm from sources other than criticality.

Alarms
Criticality alarm signals will be for prompt evacuation or other protective actions. The alarm signals should be uniform throughout the system. The signals will be distinctive from other signals or alarms that require a response different from the response necessary in the event of a criticality accident. The signal generators will be automatically and promptly actuated upon detection of a criticality accident. The audio generators should produce an overall sound pressure level of at least 75 decibels (dB), but not less than 10 dB above the maximum ambient noise level typical of each area for which audio coverage is to be provided.

Dependability

Criticality alarm systems will be designed so that alarm actuation will occur as a result of the minimum duration transient. It may be assumed that the minimum duration of the radiation transient is one millisecond.

The system will be designed for high reliability and should use components that do not require frequent servicing (such as lubrication or cleaning). The system should be designed to minimize the effects of non-use, deterioration, power surges, and other adverse conditions. The design of the system should be as simple as is consistent with the objectives of ensuring reliable actuation of the criticality alarm signal and 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.

All components of the system should be located or protected to minimize damage in case of fire, explosion, corrosive atmosphere, or other extreme conditions. The system should be designed to minimize the potential for failure due to human error. Major system components should be labeled.

The system should remain operational in the event of seismic shock equivalent to the site-specific design basis earthquake, or to the equivalent value specified by the national building code that applies to the structure.

f. Describe the use of neutron poisons.

The following is taken from Wikipedia, *Neutron Poison*.

A neutron poison is a substance with a large neutron absorption cross section, in applications such as nuclear reactors. In such applications, absorbing neutrons is normally an undesirable effect; however, neutron-absorbing materials, called poisons, are intentionally inserted into some types of reactors to lower the high reactivity of their initial fresh fuel load. Some of these poisons deplete as they absorb neutrons during reactor operation, while others remain relatively constant.

g. Define the following terms:

- **Burnable poison**
- **Non-burnable poison**
- **Chemical shim**

The following is taken from Wikipedia, *Neutron Poison*.

Burnable poisons

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.

Non-burnable Poisons

A non-burnable poison is one that maintains a constant negative reactivity worth over the life of the core. 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 that approximates non-burnable characteristics.

Chemical Shim

Soluble poisons, called chemical shim, produce spatially uniform neutron absorption when dissolved in the water coolant. The most common soluble poison in commercial pressurized water reactors is boric acid that is often referred to as soluble boron.

h. Explain the purpose and use of Raschig rings as a neutron poison.

The following is taken from Wikipedia, *Raschig Ring*.

Raschig rings are pieces of tube (approximately equal in length and diameter) used in large numbers as a packed bed within columns for distillations and other chemical engineering processes. They are usually ceramic or metal and provide a large surface area within the volume of the column for interaction between liquid and gas or vapor.

Source: Wikipedia, Raschig Ring

Figure 9. Raschig rings

In a distillation column, the reflux or condensed vapor runs down the column, covering the surfaces of the rings, while vapor from the reboilers goes up the column. As the vapor and liquid pass each other countercurrently in a small space, they tend towards equilibrium. Thus, less volatile material tends to go downward, more volatile material upward. Raschig rings are also used for devices where gas and liquid are put in contact for purposes of gas absorption, stripping, or chemical reaction, and as a support for biofilms in biological reactors.

Raschig rings made from borosilicate glass are sometimes employed in the handling of nuclear materials, where they are used inside vessels and tanks containing solutions of fissile material where they act as neutron absorbers and prevent a potential criticality accident.

4. **Nuclear safety specialists must demonstrate a working level knowledge of terminology used in nuclear safety analysis.**

   a. **Define the following accident related terms:**
      - **Accident**
      - **Authorization basis**
      - **Beyond design basis accident**
      - **Design basis**
      - **Design basis accidents**
      - **Evaluation guideline**
      - **Safety basis**
      - **Safety analysis**
      - **Consequence**
      - **Frequency**
      - **Risk**
      - **External event**
      - **Internal event**

Accident
The following is taken from DOE-STD-3009-94, Chg 3.

An accident is an unplanned sequence of events that results in undesirable consequences.

Authorization Basis
The following is taken from DOE G 450.4-1B (Archived).

An authorization basis is the safety documentation supporting the decision to allow a process or facility to operate. Included are corporate operational and environmental requirements as found in regulations and specific permits, and for specific activities, work packages, or job safety analyses.

Beyond Design Basis Accident
The following is taken from DOE-STD-3009-94, Chg 3.

Beyond design basis accident is an accident of the same type as a design basis accident(DBA), but, defined by parameters that exceed in severity the parameters defined for the DBA.

Design Basis
The following is taken from DOE-STD-3009-94, Chg 3.

The design basis is the set of requirements that binds the design of systems, structures, and components (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, others are not.

Design Basis Accidents
The following is taken from DOE G 420.1-1 (Archived).

A DBA is an accident postulated for the purpose of establishing functional and performance requirements for SSCs.

Video 10. Fukushima design accident
http://atomicinsights.com/more-accurate-headline-would-be-fukushima-containment-worked/

Evaluation Guideline
The following is taken from DOE-STD-3009-94, Chg 3.

The evaluation guideline (EG) is the radioactive material dose value that the safety analysis evaluates against. It is established for the purpose of identifying and evaluating safety-class (SC) SSCs. Onsite EGs are not required for adequate documentation of a safety basis utilizing the overall process of DOE-STD-3009-94, Chg 3, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analysis*.

Safety Basis
The following is taken from DOE-STD-3009-94, Chg 3.

The safety basis is the documented safety analysis (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 Analysis
The following is taken from DOE-STD-3009-94, Chg 3.

A safety analysis is 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.

Consequence
The following is taken from DOE-STD-3009-94 (Archived).

Consequence is the effect on a member of the public, a worker, or the environment, resulting from exposure to a hazard.

Frequency
The following is taken from DOE-STD-3009-94 (Archived).

Frequency is the expected number of occurrences of a hazardous scenario, generally expressed as an annual likelihood of occurrence. Frequency is typically determined by considering a combination of applicable factors such as the initiating event (IE) likelihood per year, failure probabilities of enabling events, number of repetitive operations over time, and percentage of time the hazardous material is present.

Risk
The following is taken from DOE-STD-3009-94, Chg 3.

Risk is the quantitative or qualitative expression of possible loss that considers the probability that an event will occur and the consequences of that event.

External Event
The following is taken from DOE-STD-3009-94 (Archived).

An external event is an accident IE outside the facility and not associated with facility operations. In a DSA, the term "external event" normally refers only to a human-initiated event, such as a traffic accident, an accident at an adjacent facility, a utility accident, an airplane crash, or wildfire (whether initiated by humans or not). Events caused by natural phenomena hazards (NPHs) such as earthquakes, tornados, high winds, lightning, and flooding, are external events but are generally classified as natural events or NPH events in a DSA.

Internal Events
The following is taken from DOE-STD-3009-94 (Archived).

An internal event is an accident IE associated with facility operations. Examples include operator error and equipment failure. In a DSA, internal events generally lead to operational accidents such as spills/chemical releases, fires, explosions, and criticality events.

b. Define the following hazard-related terms:
- **Hazard**
- **Hazard categorization**
- **Hazard category 1**
- **Hazard category 2**
- **Hazard category 3**
- **Hazardous material**

The following are taken from DOE-STD-3009-94, Chg 3.

Hazard
A hazard is a source of danger (i.e., material, energy source, or operation) with the potential to cause illness, injury, or death to personnel or damage to an operation or to the environment (without regard for the likelihood or credibility of accident scenarios or consequence mitigation).

Hazard Categorization
Hazard categorization is the evaluation of the consequences of unmitigated releases to categorize facilities or operations into hazard categories.

Hazard Category 1
Hazard category 1 is when the hazard analysis shows the potential for significant offsite consequences.

Hazard Category 2
Hazard category 2 is when the hazard analysis shows the potential for significant onsite consequences.

Hazard Category 3
Hazard category 3 is when the hazard analysis shows the potential for only significant localized consequences.

Hazardous Material
Hazardous material is any solid, liquid, or gaseous material that is toxic, explosive, flammable, corrosive, or otherwise physically or biologically threatening to health.

Candidate hazards include radioactive materials, hazardous chemicals as defined by Occupational Safety and Health Administration (OSHA) in 29 CFR 1910.1200, "Hazard Communication," and 29 CFR 1910.1450, "Occupational Exposure to Hazardous Chemicals in Laboratories"; any material assigned a reportable quantity value in 40 CFR 302, "Designation, Reportable Quantities and Notification," table 302.4, "List of Hazardous Substances and Reportable Quantities"; threshold planning quantities in 40 CFR 355, "Emergency Planning and Notification," appendix A, "The List of Extremely Hazardous Substances and Their Threshold Planning Quantities"; threshold planning quantities in 29 CFR 1910.119, "Process Safety Management of Highly Hazardous Chemicals"; level of concern quantities in the Environmental Protection Agency's (EPA's) *Technical Guidance for Hazard Analysis—Emergency Planning for Extremely Hazardous Substances*; or materials rated as three or four in National Fire Protection Association (NFPA) 704, *Identification of the Fire Hazards of Materials*.

Video 11. Introduction to hazardous materials
http://www.youtube.com/watch?v=-gdXIEW-WdE

c. Define the following safety control related terms:
- **Limiting conditions for operations**
- **Limiting control settings**
- **Safety limit**
- **Specific administrative control**
- **Administrative control**

The following are taken from DOE-STD-3009-94, Chg 3.

Limiting Conditions for Operations (LCO)
LCOs are the limits that represent the lowest functional capability or performance level of safety-related SSCs required for safe operations.

Limiting Control Settings (LCS)
LCSs are settings on safety systems that control process variables to prevent exceeding an SL.

Safety Limit (SL)
SLs are limits on process variables associated with those SC physical barriers, generally passive, that are necessary for the intended facility functions and that are required to guard against the uncontrolled release of radioactive materials.

Specific Administrative Control (SAC)
SACs are ACs designated as an SAC if 1) it is identified in the DSA as a control needed to prevent or mitigate an accident scenario; and 2) it has a safety function that would be safety-significant (SS) or SC if the function were provided by an SSC.

Administrative Control (AC)
ACs are provisions relating to organization and management procedures, recordkeeping, assessment, and reporting necessary to ensure the safe operation of a facility. In general, the AC section addresses 1) the requirements associated with ACs, including those for reporting violations of the technical safety requirement (TSR); 2) the staffing requirements for facility positions important to safe conduct of the facility; and 3) the commitments to the safety management programs identified in the DSA as necessary components of the safety basis for the facility.

d. Differentiate between the following categories of individuals who may be affected by an accident at a Department nuclear facility:
- **Offsite individual**
- **Onsite individual**
- **Public**
- **Worker (including collocated worker)**

The following is taken from the Defense Nuclear Facilities Safety Board (DNFSB)/Technology (TECH)-20, *Protection of Collocated Workers at the Department of Energy's Defense Nuclear Facilities and Sites*.

Offsite Individual
Offsite personnel are 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.

Onsite Individual
Other onsite personnel are 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 are 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.

Worker (Including Collocated Worker)

Immediate workers are 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 especially hazardous rooms or areas of the facility.

Collocated workers are 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.

Public

The following is taken from 10 CFR 835.2

A member of the public means an individual who is not a general employee. An individual is not a "member of the public" during any period in which the individual receives an occupational dose.

e. Differentiate between the function of structures, systems, and components in the following classifications:
- **Safety-class structures, systems, and components (SC SSCs)**
- **Safety-significant structures, systems, and components (SS SSCs)**
- **Defense-in-depth (DID)/important to safety**

Safety-Class Structures, Systems, and Components (SC SSC)

The following is taken from DOE-STD-3009-94, Chg 3.

SC SSCs are SSCs 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 (SS SSC)

The following is taken from DOE-STD-3009-94, Chg 3.

SS SSCs are SSCs that are not designated as SC SSCs, but whose preventive or mitigative function is a major contributor to DID and/or worker safety as determined from safety analyses.

Defense-in-Depth (DID)/Important to Safety

The following is taken from the U.S. Nuclear Regulatory Commission (NRC) Glossary.

DID is an approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon.

DID includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.

Video 12. Disaster in Japan
http://neutroneconomy.blogspot.com/2011/03/when-media-gets-it-right-and-wrong.html

f. Differentiate between the function and contents of the following documents:
- **Documented safety analysis (DSA)**
- **Preliminary documented safety analysis (PDSA)**
- **Safety analysis report (SAR)**
- **Basis for interim operation (BIO)**
- **Technical safety requirement (TSR)**
- **Preliminary hazards analysis (PHA)**

Documented Safety Analysis (DSA)
The following is taken from DOE G 421.1-2A.

The DSA for a DOE hazard category 1, 2, or 3 nuclear facility in accordance with 10 CFR 830.204, "Documented Safety Analysis" must, as appropriate for the complexities and hazards associated with the facility or activity
- describe the facility, activities, and operations (including the design of SSCs, 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 that 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 them 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) quality assurance (QA), procedures, maintenance, personnel training, conduct of operations, emergency preparedness, fire protection, waste management, and radiation protection; and
- with respect to a nonreactor facility with fissionable material in a form and amount sufficient to pose a potential for criticality, define a criticality safety program that
  - ensures that operations with fissionable material remain subcritical under all normal and credible abnormal conditions
  - identifies applicable NCS standards
  - describes how the program meets applicable NCS standards.

In accordance with 10 CFR 830.204, a DSA must provide a systematic identification of natural and man-made hazards to demonstrate that all relevant accidents have been considered, and appropriate preventive and mitigative measures have been derived to ensure adequate protection of workers, the public, and the environment. The facility documentation

should be in sufficient detail to support the safety analyses. Safe harbor provisions for the preparation of DSAs must conform to one of the methodologies set forth in 10 CFR 830, appendix A to subpart B, "General Statement of Safety Basis Policy," table 2, or an alternate methodology approved by DOE.

Preliminary Documented Safety Analysis (PDSA)

The following is taken from DOE-STD-1189-2008.

The PDSA is the 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
- an initial listing of the safety management programs that must be developed to address operational safety considerations

The PDSA, as opposed to the hazard analysis (HA), 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.

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. To provide a baseline understanding of the adequacy of controls, the accident analysis in the PDSA should describe how the selected controls adequately prevent/mitigate the accidents, including how the controls provide DID, if warranted, based on accident frequency and control reliability. The analysis should provide an adequate understanding of the baseline mitigated consequences for the facility. The discussion puts the hazard controls' effectiveness in accident context and provides the baseline safety analysis for the evaluation of changes, as the facility DSA is developed for the transition to operation.

Safety Analysis Report (SAR)

The following is taken from the International Atomic Energy Agency (IAEA), Safety Series No. 35-G1, *Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report*.

The SAR will be prepared by the operating organization for the justification of the design and it will be the basis for the safe operation of the research reactor. The SAR is an important link between the operating organization and the regulatory body, since it is the main document for the licensing of the reactor. In addition, the preparation of an SAR serves the following purposes:

- To aid the designer in confirming that individual systems are integrated correctly, since the reactor design and the development of the SAR are complementary, interactive processes

- To ensure the safety analysis has properly identified the safety issues relevant to the design and that safety analysis and design are consistent
- To aid in the appreciation of the relevant design criteria, their limitations and requirements, and in the evaluation of the hazards posed by the facility
- To aid operators in training and familiarization with the facility
- To ensure the establishment of operational limits and conditions on certain parameters that have to be met at all stages in the life of the reactor in order to have an adequate protection of the margins of safety for the reactor

The SAR will give a detailed description of the reactor site, the reactor itself, the experimental facilities, and all other facilities with safety significance. It will provide a detailed description of the general safety principles and criteria applied to the design for the protection of the reactor, the operating personnel, the general public, and the environment. It will analyze the potential hazards associated with the operation of the reactor. It will contain safety analyses of accident sequences and of the safety features incorporated in the design to avoid or to minimize accidents, or to mitigate their consequences through design and operating procedures.

The SAR will provide a set of operational limits and conditions to be incorporated into the license for operation. It will provide details of the conduct of operations intended by the operating organization, including its organization and the quality assurance program (QAP) established for the design and operation of the facility to include details of the emergency plan of the facility.

Basis for Interim Operation (BIO)

The following is taken from DOE G 421.1-2A.

Traditionally, DSAs have been used as the long-term safety basis document for nuclear facilities usually under steady-state conditions. There are primarily two cases where the rule allows a BIO to be used as the appropriate safety basis documentation: 1) for a DOE nuclear facility with a limited operational life; and 2) during transition phases, including transition surveillance and maintenance and deactivation. A BIO is applicable to a nuclear facility in transition as the facility moves through the appropriate life-cycle states, providing accurate safety documentation for rapidly changing activities. A BIO can be linked to a series of tasks or activities.

The following is taken from DOE-STD-3011-2002.

The BIO must include the following:

- Facility categorization according to DOE-STD-1027-92, Chg 1, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*
- A description of the facility (including the work to be performed)
- A systematic identification of hazards associated with the facility
- Evaluation of normal, abnormal, and accident conditions (including potential NPHs that might be associated with long-term status) that might be associated with the

generation or release of radioactive or other hazardous materials, including consideration of the need for analysis of beyond DBA.

- Derivation and classification of hazard controls necessary to protect workers, the public, and the environment
- Definition of the characteristics of safety management programs necessary to ensure safe operation, including criticality safety, when criticality hazards exist

Technical Safety Requirements (TSRs)

The following is taken from DOE G 423.1-1A.

TSRs define the performance requirements of SSCs and identify the safety management programs used by personnel to ensure safety. TSRs 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 HA of the activities to be performed and identification of the potential sources of safety issues. Safety analyses to identify and analyze a set of bounding accidents that take into account all potential causes of releases of radioactivity also contribute to development of TSRs.

Preliminary Hazards Analysis (PHA)

The following is taken from DOE-STD-1189-2008.

The PHA document provides a broad hazard-screening tool that includes a review of the types of operations that will be performed in the proposed facility and identifies the hazards associated with these types of operations and facilities. The results of the PHA are used to determine the need for additional, more detailed analysis; serve as a precursor where further analysis is deemed necessary; and serves as a baseline HA when further analysis is not indicated. The PHA is most applicable in the conceptual design stage, but it is useful for existing facilities and equipment that have not had an adequate baseline HA.

The following is taken from the U.S. Department of Labor, OSHA, *Process Safety Management Guidelines for Compliance*.

A PHA is one of the most important elements of the process safety management (PSM) program. A PHA is an organized and systematic effort to identify and analyze the significance of potential hazards associated with the processing or handling of highly hazardous chemicals. A PHA provides information that will assist employers and employees in making decisions for improving safety and reducing the consequences of unwanted or unplanned releases of hazardous chemicals.

g. Differentiate between the controls which have the following designations:

- **Mitigating controls**
- **Preventive controls**
- **Administrative controls**
- **Specific administrative controls**
- **Design features**
- **Passive controls**
- **Active controls**
- **Safety SSCs**
- **Controls that provide confinement**
- **Controls that provide containment**

Mitigating Controls

The following is taken from DOE-STD-3009-94, Chg 3.

Mitigating controls are any SSC that serves to mitigate the consequences of a release of hazardous materials in an accident scenario.

Preventive Controls

The following is taken from DOE-STD-3009-94, Chg 3.

Preventive controls are any SSC that serves to prevent the release of hazardous material in an accident scenario.

Administrative Controls (ACs)

The following is taken from DOE-STD-3009-94, Chg 3.

ACs are provisions relating to organization and management procedures, recordkeeping, assessment, and reporting necessary to ensure the safe operation of a facility. In general, the AC section addresses 1) the requirements associated with ACs; 2) the staffing requirements for facility positions important to safe conduct of the facility; and 3) the commitments to the safety management programs identified in the DSA as necessary components of the safety basis for the facility.

Specific Administrative Controls (SACs)

The following is taken from DOE-STD-1186-2004.

SACs are ACs that are selected to provide preventive and/or mitigative functions for specific potential accident scenarios, and that have safety importance equivalent to engineered controls that would be classified as SC SSCs or SS SSCs if the engineered controls were available and selected.

Design Features

The following is taken from 10 CFR 830.3.

Design features means the design features of a nuclear facility specified in the TSRs that, if altered or modified, would have a significant effect on safe operation.

Passive Controls

The following is taken from DOE G 423.1-1A.

Passive controls are normally passive characteristics of the facility not subject to change by operations personnel (e.g., shielding, structural walls, relative locations of major components, installed poisons, or special materials).

Active Controls

The following is taken from DOE G 423.1-1A.

Active controls are normally described in the DSA and are the subject of the various TSRs, so they are not normally described in the design features section. All changes or modifications that impact the safety basis of the facility are subject to the unreviewed safety question (USQ) process. The design features section captures those permanently built-in features critical to safety that do not require, or infrequently require, maintenance or surveillance.

Safety SSCs

The following is taken from DOE G 420.1-1A.

Safety SSCs are SC SSCs and SS SSCs.

Controls That Provide Confinement

The following is taken from DOE G 420.1-1A.

Effluent monitoring and control SSCs are generally designed to operate in conjunction with physical barriers to form a confinement system to limit the release of radioactive or other hazardous materials into the environment and to prevent or minimize the spread of contamination within the facility.

Adequate instrumentation and controls (I&Cs) should be provided to assess system performance and to allow the necessary control of system operations. Equipment in safety systems is required to be appropriately qualified or protected to ensure reliable operation during normal operating conditions; during anticipated operations occurrences; and during the following DBAs, including a design basis earthquake. SC air filtration units, effluent transport systems, or effluent collection systems are to be designed to remain functional throughout DBAs and to retain collected radioactive and hazardous materials after the accident.

Controls That Provide Containment

The following is taken from DOE G 420.1-1A.

In some cases, SS SSCs rely on supporting SSCs to perform their intended safety function. DOE O 420.1C, *Facility Safety*, requires that support SSCs be designed as SC or SS SSCs if their failures prevent SS SSCs or SACs from performing their safety functions. For example, an SC designation may be appropriate for an I&C system that supports a tritium containment system if failure of the I&C support system could lead to either failure or reduced availability of the SC containment barrier. However, if the support system would not lead to immediate

failure of the SS SSC, such as for a heat tracer on a fire protection line, combined with a safety alarm, providing adequate time for restoration action, the support system may not need to be classified as an SS SSC. The classification of the supporting SSCs would be at the same level as the SS SSCs or SACs that they could impact.

Video 13. Confinement and containment of hazardous materials
http://www.youtube.com/watch?v=N3welRaTRhg

Priority of Controls

The following is taken from DOE G 420.1-1A.

Prioritization of items for a facility safety strategy is as follows:

- Minimization of hazardous materials is the first priority.
- SS SSCs are preferred over ACs.
- Passive SSCs are preferred over active SSCs.
- Preventive controls are preferred over mitigative controls.
- Facility safety SSCs are preferred over personal protective equipment.

In addition, the following should be considered during design development:

- Controls closest to the hazard are preferred since they may provide protection to the largest population of potential receptors, including workers and the public.
- Controls that are effective for multiple hazards are preferred since they can be resource effective.

h. Differentiate between the following types of facilities:

- **Nuclear facility**
- **Category A nuclear reactor**
- **Category B nuclear reactor**
- **Nonreactor nuclear facility**
- **Radiological (below hazard category 3 nuclear facility)**

Nuclear Facility

The following is taken from DOE Order 5480.30, Chg 1.

Nuclear facility means reactor and nonreactor nuclear facilities.

Category A Nuclear Reactor

The following is taken from DOE Order 5480.30, Chg 1.

Category A nuclear reactor means those test and research reactors designated by DOE based on power level (e.g., design thermal power rating of 20 megawatts steady state and higher), potential fission product inventory, and experimental capability.

Category B Nuclear Reactor
The following is taken from DOE Order 5480.30, Chg 1.

Category B nuclear reactor means those test and research reactors designated by DOE based on power level (e.g., design thermal power rating of less than 20 megawatts steady state), potential fission product inventory, and experimental capability.

Nonreactor Nuclear Facility
The following is taken from 10 CFR 830.3.

Nonreactor nuclear facility means those facilities, activities, or operations that involve, or will involve, radioactive and/or fissionable materials in such form and quantity that a nuclear or a nuclear explosive hazard potentially exists to workers, the public, or the environment; but, does not include accelerators and their operations; 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.

Radiological (Below Hazard Category 3 Nuclear Facility)
The following is taken from DOE-STD-1027-92, Chg 1.

Facilities that do not meet or exceed hazard category 3 threshold criteria but still possess some amount of radioactive material may be considered radiological facilities. Radiological facilities are exempt from SAR/DSA, but they are not exempt from other safety requirements.

i. Differentiate between the following chemical terms:
- **Acute exposure guidelines Level (AEGL)-1**
- **AEGL-2**
- **AEGL-3**
- **Emergency response planning guideline (ERPG)-1**
- **ERPG-2**
- **ERPG-3**
- **Temporary emergency exposure limit (TEEL)-1**
- **TEEL-2**
- **TEEL-3**

Acute Exposure Guidelines Level (AEGL)-1
The following is taken from the Environmental Protection Agency, *Acute Exposure Guideline Levels (AEGL),* "Definitions."

AEGL-1 is the airborne concentration, expressed as parts per million or milligrams per cubic meter (ppm or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, could experience notable discomfort, irritation, or certain asymptomatic non-sensory effects. However, the effects are not disabling and are transient and reversible upon cessation of exposure.

AEGL-2

The following is taken from the Environmental Protection Agency, *Acute Exposure Guideline Levels (AEGL),* "Definitions."

AEGL-2 is the airborne concentration (expressed as ppm or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, could experience irreversible or other serious, long-lasting adverse health effects or an impaired ability to escape.

AEGL-3

The following is taken from the Environmental Protection Agency, *Acute Exposure Guideline Levels (AEGL)*, "Definitions."

AEGL-3 is the airborne concentration (expressed as ppm or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, could experience life-threatening health effects or death.

Video 14. Acute exposure guideline level (AEGL)
http://www.youtube.com/watch?v=HkV39fi3Cyo

The following is taken from the American Industrial Hygiene Association, *Emergency Response Planning Guidelines*, "ERPG Levels."

The emergency response planning committee develops guidelines for responding to potential releases of airborne substances for use in community emergency planning. ERPGs are air concentration guidelines for single exposures to agents and are intended for use as tools to assess the adequacy of accident prevention and emergency response plans, including transportation emergency planning, community emergency response plans, and incident prevention and mitigation.

Emergency Response Planning Guideline (ERPG)-1

The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to one hour without experiencing more than mild, transient adverse health effects or without perceiving a clearly defined objectionable odor.

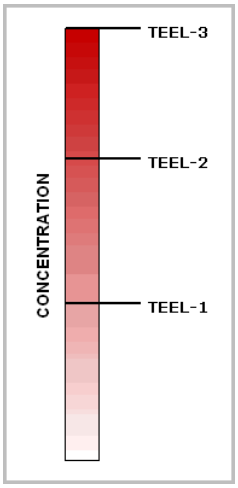
ERPG-2

The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair an individual's ability to take protective action.

ERPG-3

The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.

The following is taken from the U.S. Department of Commerce, National Oceanic and Atmospheric Administration (NOAA), National Ocean Service, *Temporary Emergency Exposure Limits (TEELs)*.



TEELs are guidelines designed to predict the response of members of the general public to different concentrations of a chemical during an emergency response incident. TEELs estimate the concentrations at which most people will begin to experience health effects if they are exposed to a hazardous airborne chemical for a given duration. TEELs are used in similar situations as the 60-minute AEGLs and ERPGs.

A chemical may have up to three TEEL values, each of which corresponds to a specific tier of health effects. The three TEEL tiers are defined as follows:

- TEEL-3 is the airborne concentration, expressed as ppm or mg/m^3, of a substance above which it is predicted that the general population, including susceptible individuals, could experience
  life-threatening adverse health effects or death.
- TEEL-2 is the airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience irreversible or other serious, long-lasting, adverse health effects or an impaired ability to escape.
- TEEL-1 is the airborne concentration of a substance above which it is predicted that the general population, including susceptible individuals, could experience notable discomfort, irritation, or certain asymptomatic, non-sensory effects. However, these effects are not disabling and are transient and reversible on cessation of exposure.

j. Identify the types of chemical or toxicological hazards that may be found in nuclear facilities.

The following is taken from DOE-STD-6003-96.

The types of chemical or toxicological hazards that may be found in nuclear facilities include any solid, liquid, or gaseous material that is toxic, explosive, flammable, corrosive, or otherwise physically or biologically threatening to health.

These and other materials, including radioactive materials, are defined in 29 CFR 1910.1200; 29 CFR 1910.1450; reportable quantities pursuant to 40 CFR 302; threshold planning quantities pursuant to 40 CFR 355, appendix A; threshold planning quantities per 29 CFR 1910.119; level of concern quantities in the EPA's, *Technical Guidance for Hazards Analysis: Emergency Planning for Extremely Hazardous Substances*; and materials rated as 3 or 4 in NFPA 704.

5. **Nuclear safety specialists must demonstrate a working level knowledge of the principal hazard and accident analysis methods.**

   a. **Identify and discuss the use of different methods for qualitative hazard analysis. Identify specific strengths and weaknesses with the various methods.**

DOE-STD-3009-94 references *Guidelines for Hazard Evaluation Procedures*, third edition (2008) published by the American Institute of Chemical Engineers. OSHA has accepted these guidelines as the standard for analytical adequacy in characterizing commercial chemical processes that perform the same type of unit operations conducted at DOE nonreactor nuclear facilities.

The following is taken from the Chemical Emergency Prevention and Planning Newsletter, *Process Hazard Analysis*.

The process hazard analysis is a thorough, orderly, and systematic approach for identifying, evaluating, and controlling the hazards of processes involving highly hazardous chemicals. The facility will perform a process hazard analysis on all processes covered by the EPA risk management program rule or OSHA PSM standard.

The process hazard analysis methodology selected must be appropriate to the complexity of the process and must identify, evaluate, and control the hazards involved in the process. First, the facility must determine and document the priority order for conducting process hazard analyses based on a rationale that includes such considerations as the extent of the process hazards, the number of potentially affected employees, the age of the process, and the operating history of the process. The process hazard analyses should be conducted as soon as possible.

The facility will use one or more of the following methods, as appropriate, to determine and evaluate the hazards of the process being analyzed:

- Checklist
- What-if
- What-if/checklist
- Hazard and operability study (HAZOP)
- Failure mode and effects analysis (FMEA)
- Fault tree analysis
- An appropriate equivalent methodology

Checklist

Checklists are primarily used for processes that are covered by standards, codes, and industry practices—for example: storage tanks designed to the American Society for Mechanical Engineers (ASME) standards, ammonia handling covered by OSHA (29 CFR 1910.111, "Storage and Handling of Anhydrous Ammonia"), propane facilities subject to NFPA 58, *Liquefied Petroleum Gas Code*.

Checklists are easy to use and can help familiarize new staff with the process equipment. Checklists are a highly cost-effective way to identify customarily recognized hazards. Checklists are, however, dependent on the experience of the people who develop them; if the

checklist is not complete, the analysis may not identify hazardous situations. A checklist analysis involves touring the process area and comparing equipment to the list.

What-If

A what-if is a brainstorming approach in which a group of people familiar with the process ask questions about possible deviations or failures. A scribe or recorder takes down all of the questions on flip charts or a computer. The questions are then divided into specific areas of investigation, usually related to consequences of interest. Each area is then addressed by one or more team member(s).

What-if analyses are intended to identify hazards, hazardous situations, or accident scenarios. It requires a basic understanding of the process and an ability to combine possible deviations from design intent with outcomes. It is a powerful procedure if the staff is experienced; otherwise, the results are likely to be incomplete.

Video 15. Strategic what-if analysis
http://www.youtube.com/watch?v=uKrxNmgoRjs

What-If/Checklist

A what-if/checklist combines the creative, brainstorming aspects of the what-if with the systematic approach of the checklist. The combination of techniques can compensate for the weaknesses of each and can help the team identify hazards and accident scenarios that are beyond the experience of the team members. The checklist provides a more detailed systematic approach that can fill in gaps in the brainstorming process. The technique is generally used to identify the most common hazards that exist in a process. It is often the first process hazard analysis conducted on a process, with subsequent analyses using more detailed approaches.

This technique is usually performed by a team experienced in the design, operation, and maintenance of the process. The number of people required depends on the complexity of the process.

HAZOP

The HAZOP was originally developed to identify hazards and operability problems at chemical process plants, particularly for processes using technologies with which the plant was not familiar. The technique has been found to be useful for existing processes as well. A HAZOP requires an interdisciplinary team and an experienced team leader.

The purpose of a HAZOP is to review a process or operation systematically to identify whether process deviations could lead to undesirable consequences. The technique can be used for continuous or batch processes and can be adapted to evaluate written procedures at any stage in the life of a process.

HAZOP requires more resources than simpler techniques. A simpler process or a review with a narrow scope may be done by as few as three or four people, if they have the technical skills and experience. A large or complex process usually requires a team of five to seven people.

Video 16. HAZOP analysis
http://www.youtube.com/watch?v=-rLiAKoJUDk

Failure Mode and Effect Analysis (FMEA)

A FMEA evaluates the ways in which equipment fails and the system's response to the failure. The focus of the FMEA is on single equipment failures and system failures. It usually generates recommendations for increasing equipment reliability. The FMEA does not examine human errors directly, but will consider the impact on equipment of human error. The FMEA is not efficient for identifying an exhaustive list of combinations or equipment failures that lead to accidents.

An FMEA produces a qualitative, systematic list of equipment, failure modes, and effects and usually produces a table that, for each item of equipment, includes a description, a list of failure modes, the effects of each failure, safeguards that exist, and actions recommended to address the failure. Generally, when analyzing impacts, analysts assume that existing safeguards do not work. More optimistic assumptions may be satisfactory as long as all equipment failure modes are analyzed on the same basis.

Video 17. What is FMEA?
http://www.youtube.com/watch?v=IWPt6exPwjI

Fault Tree Analysis (FTA)

An FTA is a deductive technique that focuses on a particular accident or main system failure and provides a method for determining causes of the event. The fault tree (FT) is a graphic that displays the combinations of equipment failures and human errors that can result in the accident. The FTA starts with the accident and identifies the immediate causes. Each immediate cause is examined to determine its causes until the basic cause of each are identified. The strength of FTA is its ability to identify combinations of basic equipment and human failures that can lead to an accident, allowing the analyst to focus preventive measures on significant basic causes.

The FTA is well-suited for analyses of highly redundant systems. For systems vulnerable to single failures that can lead to accidents, FMEA or HAZOP are better techniques to use. FTA is often used when another technique has identified an accident that requires more detailed analysis. The FTA looks at component failures and faults. An FTA requires a detailed knowledge of how the plant or system works, detailed process drawings and procedures, and knowledge of component failure modes and effects. The FTA needs well-trained and experienced analysts. Although a single analyst can develop an FT, input and review from others is needed.

Other Techniques

The RMP rule allows other techniques if they are functionally equivalent. Guidelines include descriptions of a number of other techniques including preliminary hazard review, cause-consequence analysis, event tree (ET) analysis, and human reliability analysis. One may develop a hybrid technique that combines features of several techniques or apply more than one technique.

b. Discuss methods used to categorize and bin hazardous conditions associated with nuclear safety analysis.

The following is taken from DOE-STD-1027-92, Chg 1.

DOE-STD-1027-92, Chg 1, section 3.0, "Hazard Categorization," contains a uniform methodology to develop the initial hazard categorization specified in the preliminary assessment of facility hazards of DOE Order 5480.23, *Nuclear Safety Analysis Reports* (Archived).

Hazard categories 1, 2, or 3 are classified depending only on the quantities of radioactive material in the facility. Only facilities that fall below the category 3 threshold are exempt from the requirements. However, these facilities should have ACs in place to ensure minimum values are not exceeded through introduction of new material.

Once an HA has been performed, the hazard categorization can be finalized. The final categorization is based on an "unmitigated release" of available hazardous material. For the purposes of hazard categorization, "unmitigated" is meant to consider material quantity, form, location, dispersability, and interaction with available energy sources, but not to consider safety features that will prevent or mitigate a release.

The nuclear hazard categorization summary includes the following:

- Category 3—HA shows the potential for only significant localized consequences.
- Category 2—HA shows the potential for significant onsite consequences.
- Category 1—HA shows the potential for significant offsite consequences.

Hazard Category 1

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 that possesses this hazard potential is the class A nuclear reactors. In addition, the public safety officer (PSO) may designate other facilities as category 1 if he/she feels there exists the potential for significant offsite consequences.

Hazard Category 2

The approach for designating category 2 hazards was constructed from existing regulations that define minimum thresholds for many radionuclides and hazardous chemicals on the basis of consequences from these hazards in the immediate vicinity of a facility. For radioactive materials, 10 CFR 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," derived quantities above which byproduct material licensees must provide a dose of 1 rem at 100 meters under very conservative meteorological conditions.

The threshold value for fissile material is the minimum theoretical mass necessary for a nuclear criticality to occur with moderation and reflection. Category B nuclear reactors should be classified as category 2 since critical quantities of fissile materials are present in these facilities but not in sufficient quantities to represent a significant offsite impact. See table 2 for threshold amounts.

Hazard Category 3

Category 3 is designed to capture facilities that largely include lab operations, low-level waste handling facilities, and research machines that possess less than the category 2 quantities of material and are considered to represent a low hazard. Essentially all industrial facilities have a potential for significant localized consequences because the potential to injure workers from typical industrial accidents is always present. However, category 3 facilities pose additional hazards due to the presence of radionuclides.

Table 2. Thresholds for radionuclides

Isotopes	Category 2[1] Curies	Threshold in Grams	Category 3[2] Curies	Threshold in Grams
H-3	3.0E+05	3.0E+01	1.6E+04*	1.6E+00*
C-14	1.4E+06	3.1E+05	4.2E+02	9.4E+01
Na-22	6.3E+03	1.0E+00	2.4E+02	3.8E-02
P-32	4.4E+03	1.5E-04	1.2E+01	4.2E-05
P-33	3.0E+04	1.9E-01	9.4E+01	6.0E-04
P-32, acid	2.2E+06	7.7E-02	1.2E+01	4.2E-05
P-33, acid	1.5E+07	9.6E+01	9.4E+01	6.0E-04
S-35	2.5E+04	5.8E-01	7.8E+01	1.8E-03
Cl-36	1.4E+03	4.3E+04	3.4E+02	1.0E+04
K-40	4.7E+03	6.8E+08	1.7E+02	2.4E+07
Ca-45	4.7E+06	2.6E+02	1.1E+03	6.2E-02
Ca-47	4.8E+06	7.8E+00	7.0E+02	1.1E-03
Sc-46	1.4E+06	4.0E+01	3.6E+02	1.1E-02
Ti-44	3.2E+04	1.9E+02	6.2E+01	3.6E-01
V-48	3.0E+06	1.8E+01	6.4E+02	3.8E-03
Cr-51	1.0E+08	1.1E+03	2.2E+04	2.4E-01
Mn-52	4.0E+06	8.8E+00	3.4E+02	7.6E-04
Fe-55	1.1E+07	4.6E+03	5.4E+03	2.2E+00
Fe-59	1.8E+06	3.7E+01	6.0E+02	1.2E+02
Co-60	1.9E+05	1.7E+02	2.8E+02	2.5E-01
Ni-63	4.5E+06	8.0E+04	5.4E+03	9.5E+01
Zn-65	1.6E+06	1.9E+02	2.4E+02	2.9E-02
Ge-68	5.8E+05	8.8E+01	1.0E+03	1.5E-01
Se-75	3.4E+05	2.4E+01	3.2E+02	2.2E-02
Kr-85	2.8E+07	7.2E+04	2.0E+04	5.1E+01
Sr-89	7.7E+05	2.7E+01	3.4E+02	1.2E-02
Sr-90	2.2E+04	1.6E+02	1.6E+01	1.2E-01
Y-91	6.5E+05	2.7E+01	3.6E+02	1.5E-02
Zr-93	8.9E+04	3.6E+07	6.2E+01	2.5E+04
Zr-95	1.5E+06	6.9E+01	7.0E+02	3.3E-02
Nb-94	8.6E+04	4.6E+05	2.0E+02	1.1E+03
Mo-99	7.8E+06	1.6E+01	3.4+03	7.1E-03
Tc-99	3.8E+06	2.3E+08	1.7E+03	1.0E+05
Ru-106	6.5E+03	1.9E+00	1.0E+02	3.0E-02
Ag-110m	5.3E+05	1.1E+02	2.6E+02	5.5E-02
Cd-109	2.9E+05	1.1E+02	1.8E+02	7.0E-02

Cd-113	1.8E+04	5.3E+16	1.1E+01	3.2E+13
In-114m	3.7E+05	1.6E+01	2.2E+02	9.5E-03
Sn-113	3.2E+06	3.2E+02	1.3E+03	1.3E-01
Sn-123	9.5E+05	1.2E+02	3.2E+02	3.9E-02
Sn-126	3.3E+05	1.2E+07	1.7E+02	6.0E+03
Sb-124	1.3E+06	7.5E+01	3.6E+02	2.1E-02
Sb-126	2.5E+06	3.0E+01	2.8E+02	3.4E-03
Te-127m	1.5E+05	1.6E+01	4.0E+02	4.2E-02
Te-129m	1.4E+05	4.7E+00	4.0E+02	1.3E-02
I-125	2.4E+03	1.4E-01	5.6E-01	3.2E-05
I-131	1.8E+03	1.4E-02	9.2E-01	7.4E-06
Xe-133	1.8E+06	9.6E+00	2.0E+04	1.1E-01
Cs-134	6.0E+04	4.6E+01	4.2E+01	3.3E-02
Cs-137	8.9E+04	1.0E+03	6.6E+01	6.9E-01
Ba-133	4.0E+06	1.6E+04	1.1E+03	4.3E+00
Ba-140	7.8E+06	1.1E+02	6.0E+02	8.2E-03
Ce-141	3.3E+06	1.2E+02	1.0E+03	3.5E-02
Ce-144	8.2E+04	2.6E+01	1.0E+02	3.1E-02
Pm-145	1.1E+06	7.6E+03	2.0E+03	1.4E+01
Pm-147	8.4E+05	9.0E+02	1.0E+03	9.5E-01
Sm-151	9.9E+05	3.7E+04	1.0E+03	3.8E+01
Eu-152	1.3E+05	7.5E+02	2.0E+02	1.2E+00
Eu-154	1.1E+05	4.2E+02	2.0E+02	7.6E-01
Eu-155	7.3E+05	1.6E+03	9.4E+02	2.0E+00
Gd-153	1.4E+06	3.9E+02	1.0E+03	2.8E-01
Tb-160	1.3E+06	1.1E+02	5.6E+02	5.0E-02
Ho-166m	4.0E+04	2.2E+04	7.2E+01	4.0E+01
Tm-170	1.2E+06	2.1E+02	5.2E+02	8.7E-02
Hf-181	2.2E+06	1.3E+02	7.6E+02	4.5E-02
Ir-192	1.2E+06	1.3E+02	9.4E+02	1.0E-01
Au-198	9.3E+06	3.8E+01	2.0E+03	8.2E-03
Hg-203	4.3E+05	3.1E+01	3.6E+02	2.6E-02
Pb-210	2.2E+03	2.9E+01	3.6E-01	4.7E-03
Bi-207	2.2E+06	4.3E+04	5.0E+02	1.1E+01
Bi-210	1.5E+05	1.2E+00	3.2E+02	2.6E-03
Po-210	3.5E+02	7.8E-02	1.9E+00	4.2E-04
Rn-222	1.6E+08	1.1E+03	1.0E+01	6.5E-05
Ra-223	3.8E+03	7.4E-02	6.2E+01	1.2E-03
Ra-224	9.9E+03	6.1E-02	2.0E+02	1.2E-03
Ra-225	3.8E+03	9.6E-02	7.2E+01	1.8E-03
Ac-225	2.9E+03	4.9E-02	3.2E+01	5.5E-04
Ac-227	4.3E+00	5.9E-02	4.2E-02	5.8E-04
Th-228	9.2E+01	1.1E-01	1.0E+00	1.2E-03
Th-230	8.9E+01	4.4E+03	6.2E-01	3.1E+01
Th-232	1.8E+01	1.6E+08	1.0E-01	9.1E+05
U-233	2.2E+02***	2.3E+04***	4.2E+00	4.4E+02
U-234	2.2E+02	3.5E+04	4.2E+00	6.7E+02
U-235	2.4E+02***	1.1E+08***	4.2E+00	1.9E+06

U-238	2.4E+02	7.1E+08	4.2E+00	1.3E+07
Np-237	5.8E+01	8.3E+04	4.2E-01	6.0E+02
Np-238	9.1E+05	3.5E+00	1.3E+03	5.0E-03
Pu-238	6.2E+01	3.6E+00	6.2E-01	3.6E-02
Pu-239	5.6E+01***	9.0E+02***	5.2E-01	8.4E+00
Pu-241	2.9E+03	2.8E+01	3.2E+01	3.1E-01
Am-241	5.5E+01	1.6E+01	5.2E-01	1.5E-01
Am-242m	5.6E+01	5.8E+00	5.2E-01	5.3E-02
Am-243	5.5E+01	2.8E+02	5.2E-01	2.6E+00
Cm-242	1.7E+03	5.1E-01	3.2E+01	9.7E-03
Cm-245	5.3E+01	3.1E+02	5.2E-01	3.0E+00
Cf-252	2.2E+02	4.1E-01	3.2E+00	5.9E-03

[1]For isotopes not listed, users may refer to LA-12846-MS, *Specific Activities and DOE-STD-1027-92 Hazard Category 2 Thresholds*, Los Alamos National Laboratory (LANL) Fact Sheet or to 10 CFR 30.72, schedule C, "Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release," and adjust the values consistent with the X/Q value described in DOE-STD-1027-92, attachment 1. (Note that although LA-12846-MS misstates the category 2 threshold criterion, its use of the proper X/Q negates any effect of the misstatement. See "Radiological Criteria," and "Meteorological Conditions," for clarification.)

Any other beta-gamma emitter – 4.3E+05 Ci
Mixed fission products – 1.0E+03 Ci
Any other alpha emitter – 5.5E+01 Ci

[2]For isotopes not listed below, users may refer to LA-12891-MS, *Table of DOE-STD-1027-92 Hazard Category 3 Threshold Quantities for the ICRP-30 List of 757 Radionuclides*, LANL Fact Sheet, for threshold quantities of any isotopes of interest.
*At the recommendation of the Tritium Focus Group, the category 3 tritium threshold value has been increased from 1.0E+03 Ci and 1.0E+04 Ci and 1.6E+00 grams, consistent with the methodology of EPA used for the other nuclides.
** Provided as an example to indicate that when a substance such as P_{32} is used in a solution (i.e., phosphoric acid) for experimentation, medical treatment, etc., it should no longer be considered as highly volatile/combustible.
***To be used only if segmentation or nature of process precludes potential for criticality. Otherwise, use the criticality lists for U_{233}, U_{235}, and Pu_{239} of 500, 700, and 450 grams, respectively.

Source: DOE-STD-1027-92, Chg 1

c. Identify and discuss the methods used to determine and analyze failure modes of SSCs, ACs, and control programs.

The following is taken from Wikipedia, *Failure Mode and Effects Analysis*.

FMEA was one of the first systematic techniques for failure analysis. It is an inductive reasoning (forward logic) single-point of failure analysis and is a core task in reliability engineering, safety engineering, and quality engineering. Quality engineering is concerned with the process (manufacturing and assembly) type of FMEA.

A successful FMEA activity helps to identify potential failure modes based on experience with similar products and processes—or based on common physics of failure logic. It is widely used in development and manufacturing industries in various phases of the product life cycle.

Functional analyses are needed as an input to determine correct failure modes, at all system levels, functional FMEA or piece-part (hardware) FMEA. An FMEA is used to structure mitigation for risk reduction based on either failure (mode) effect severity reduction or on lowering the probability of failure, or both. The FMEA is in principle a full inductive

(forward logic) analysis; however, the failure probability can only be estimated or reduced by understanding the failure mechanism. Ideally this probability will be lowered to "impossible to occur" by eliminating the root causes. It is therefore important to include in the FMEA an appropriate depth of information on the causes of failure.

Different types of FMEA analysis include the following:

- Functional—Before design solutions are provided functions can be evaluated on potential functional failure effects. General mitigations can be proposed to limit consequences of functional failures or limit the probability of occurrence in this early development. It is based on functional breakdown of a system. This type can be used for software evaluation.
- Design—Analysis of systems or subsystems in the early design concept stages to analyze the failure mechanisms and lower-level functional failures, especially to different concept solutions in more detail. This type may be used in trade-off studies.
- Process—Analysis of manufacturing and assembly processes. Quality and reliability may be affected from process faults. The input includes work process/task breakdown.

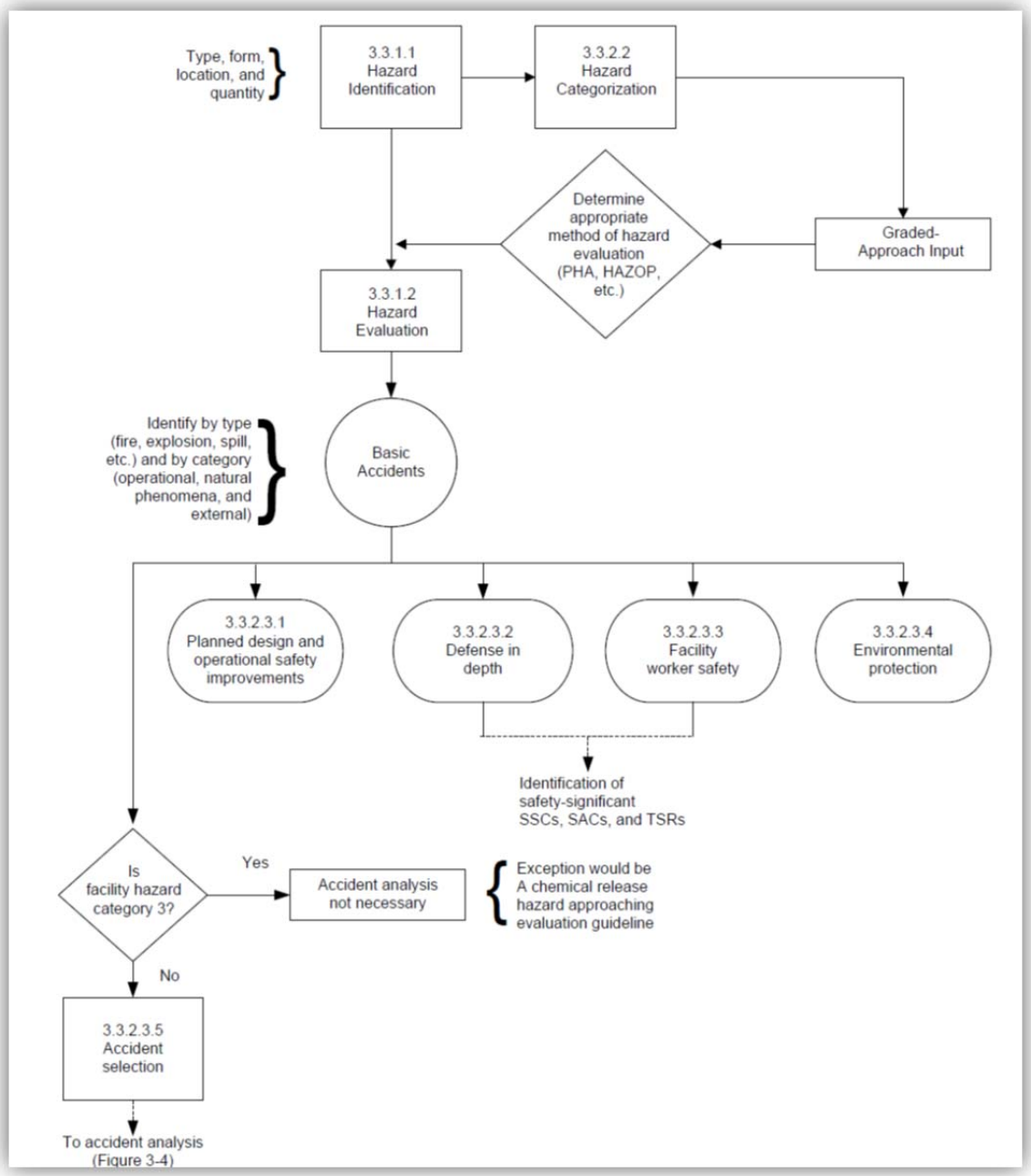
Video 18. Failure mode effects and analysis
http://www.youtube.com/watch?v=7aKLKE6KAjA

d. Identify and discuss methods available to reviewers to determine if a hazard analysis has omitted important accident vulnerabilities.

The following is taken from DOE-STD-3009-94.

DOE-STD-3009-94, section 3.3, "Hazard Analysis," describes the hazard identification and evaluation performed. The purpose of this information is to present a comprehensive evaluation of potential process-related, natural events, and man-made external hazards that can affect the public, workers, and the environment due to single or multiple failures. Consideration is given to all modes of operation, including startup, shutdown, and abnormal testing or maintenance configurations.

A basic flowchart for hazard/accident analysis is provided in figure 10. The major features of HA and the graded approach are captured in this figure. Hazard identification provides the basis for the final hazard categorization of the facility. That categorization is input for the graded approach for hazard evaluation.



Source: DOE-STD-3009-94

Figure 10. Flowchart for performing a hazard analysis

Figure 10 identifies the specific point where the analyst must move beyond the general outline and use the graded approach to specifically determine appropriate HA methodology. Application of a graded approach is based on the judgment and experience of the analysts and results in the selection of a hazard evaluation technique such as PHA, HAZOP, etc. More elaborate techniques will generally be associated with more complex processes. Experience and capabilities of analysts are a major consideration in efficient performance of a comprehensive hazard evaluation.

Systematic application of the chosen techniques to the operations in a facility generates a number of basic accidents based on types of events and system performance in response to the events. These accidents can be binned in accordance with predefined consequence and frequency ranking thresholds. Products of the hazard evaluation include the following:

- Identification of planned design and operational safety improvements
- Summary of DID including identification of SS SSCs and other items needing TSR coverage, including relevant programs covered under TSR ACs
- Summary of significant worker protection features including identification of SS SSCs and relevant programs covered under TSR ACs
- Summary of design and operational features that reduces the potential for large material releases to the environment
- Selection of a limited set of bounding accidents (i.e., DBAs) to be further developed in DOE-STD-3009-94, section 3.4, "Accident Analysis"

Video 19. Job hazard analysis
http://www.youtube.com/watch?v=wNb_ffyHmVw

e. Identify and discuss the relationship between hazard analysis and the postulation of accidents for quantitative consequence analysis in DSA for DOE nuclear facilities. Describe what factors govern the choice of an accident warranting further consequence analysis.

The following is taken from DOE-STD-3009-94, Chg 3.

Accident analysis entails the formal quantification of a limited subset DBAs. These accidents represent a complete set of bounding conditions. The identification of DBAs results from the hazard evaluation ranking of the complete spectrum of facility accidents. The approach used at any specific facility is based on the detail needed for a given facility and the experience of the analysts. HA involves ranking risk for accidents by consequence and frequency. The ranking schemes are designed to separate the lower risk accidents that are adequately assessed by hazard evaluation from higher risk accidents that may warrant additional quantitative analysis if the phenomena involved are not simplistic. Rankings use a bin system, for example; frequency bins should typically cover two orders of magnitude. Although the exercise of binning is essentially qualitative, analysts often use a simple numerical basis for judgments to provide consistency. See tables 3 and 4 for examples of hazard evaluation ranking mechanisms.

Table 3. Qualitative severity classification table

Descriptive Word	Description
No	Negligible onsite and offsite impact on people or the environs
Low	Minor onsite and negligible offsite impact on people or the environs
Moderate	Considerable onsite impact on people or the environs; only minor offsite impact
High	Considerable onsite and offsite impacts on people or the environs

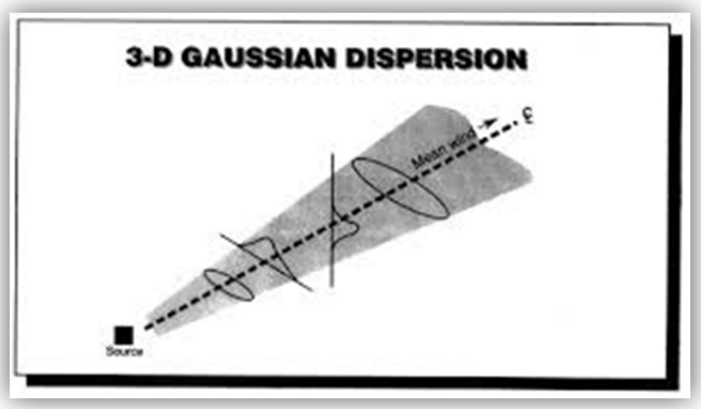
Source: DOE-STD-3009-94

Table 4. Qualitative likelihood classification table

Descriptive Word	Estimated Annual Likelihood of Occurrence	Description
Anticipated	$10^{-1}>p>10^{-2}$	Incidents that may occur several times during the lifetime of the facility (incidents that commonly occur).
Unlikely	$10^{-2}>p>10^{-4}$	Accidents that are not anticipated to occur during the lifetime of the facility. Natural phenomena of this probability class include: uniform building code-level earthquake, 100-year flood, maximum wind gust, etc.
Extremely Unlikely	$10^{-4}>p>10^{-6}$	Accidents that will probably not occur during the life cycle of the facility. This class includes the DBAs.
Beyond Extremely Unlikely	$10^{-6}>p$	All other accidents.

Source: DOE-STD-3009-94

Another methodology would be to use a summary of historical data. Before beginning the evaluation, a conservative Gaussian plume estimation of the amount of material needed outside the building to cause a certain dose might be performed to aid in defining thresholds of significance. (See figure 11.) However, the ranking of frequency and consequence into thresholds of significance is more of a qualitative than a quantitative exercise. An important factor in estimating binning thresholds for public consequences is to tie the thresholds to the EG so that accidents that could challenge the guideline are correctly identified for formal accident analysis.



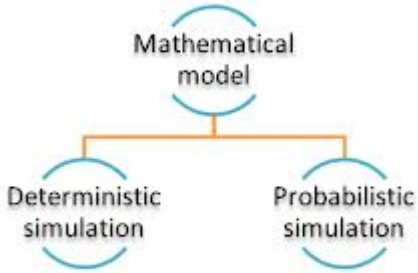
Source: Thistle, Jr., Harold W.; Milton E. Teske; and Richard C. Reardon, Weather Effects on Drift Meteorological Factors and Spray Drift: An Overview

Figure 11. Gaussian dispersion

This accident selection activity identifies the process and criteria used to select the unique and representative potential accidents to be included in the accident analysis. Unique accidents are those with sufficiently high-risk estimates that individual examination is needed. Representative accidents bound a number of similar accidents of lesser risk. 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 HA should be selected unless the bounding consequences are low. Accidents are identified and listed by accident category and type.

f. Identify and discuss essential elements of deterministic and probabilistic risk assessment (PRA) techniques.

The following is taken from the Technical Panel of the EQP Risk Assessment Forum, *Using Probabilistic Methods to Enhance the Role of Risk Analysis in Decision-Making with Case Study Examples*.



PRA is a group of techniques that provide estimates of the range and likelihood of hazard, exposure, or risk, rather than a single-point estimate. PRA includes techniques that can be applied formally to address variability and uncertainty. It is used in sciences, business, economics, and other fields to examine existing data and estimate the chance of an event, from health effects to rain to metal fatigue. One can use probability to quantify the frequency of occurrence or the degree of belief in information. Probability distributions are interpreted as representing the relative frequency of a given state of the system; whereas, for uncertainty, they represent the degree of belief or confidence that a given state of the system exists. PRA often is defined narrowly to mean a statistical or

thought process used to analyze and evaluate the variability of available data or to look at uncertainty across data sets.

Deterministic risk analysis often is considered a traditional approach to risk analysis because of the existence of established guidance and procedures regarding its use, the ease with which is can be performed, and its limited data and resource needs. The use of defaults supporting deterministic risk assessment provides a procedural consistency that allows for risk assessments to be feasible and tractable. Deterministic risk assessment approach is used to address assessment-related uncertainties primarily through the incorporation of predetermined default values and conservative assumptions. It addresses variability by combining input parameters intended to be representative of typical or higher-end exposure (considered to be conservative assumptions). The intention is often to provide a margin of safety or to construct a screening level estimate of high-end exposure and risk. Deterministic risk assessment provides an estimation of exposures and resulting risk that addresses uncertainties and variabilities in a qualitative manner.

g. Given an accident source term of radionuclide/hazardous chemical release, discuss the factors that should be considered in selection of an appropriate computer code for offsite transport and deposition.

The following is taken from DOE/EH-4.2.1.2-Criteria, *Software Quality Assurance Plan and Criteria for the Safety Analysis Toolbox Codes*.

Safety analysis software for the DOE "toolbox" was designated by the Department of Energy, Office of Environmental, Safety, and Health (DOE/EH) in March 2003. The supporting basis for this designation was provided by a DOE-chartered safety analysis software group in the technical report, *Selection of Computer Codes for DOE Safety Analysis Applications*, (August 2002). The codes for toolbox status, their version, and area of applicability are listed in table 5. Later versions of the codes may be selected based on recommendations by the software developers and information obtained in the course of the software quality assurance (SQA) implementation program.

Eventually, each of these six codes and their respective development programs will undergo evaluation of their SQA attributes relative to established requirements identified in DOE-EH-4.2.1.2-Criteria and is termed an SQA evaluation. The SQA evaluation will assess those measures requiring action before the individual codes meet current SQA-compliant standards, and will be documented in a series of reports.

Table 5. Software designated for DOE safety analysis toolbox

Code	Version or Revision	Area of Applicability
ALOHA	5.2.3	Chemical release/Dispersion and consequence
CFAST	3.1.6	Fire analysis
EPIcode	6.0	Chemical release/Dispersion and consequence
GENII	2.0*	Radiological dispersion and consequence
MACCS2**	1.12	Radiological dispersion and consequence
MELCOR	1.8.5	Leak path factor

*Version 1.485 may be advised for interim use before potential upgrades are completed. Recommendation to be based on near-term evaluation.
**Also MACCS, version 1.5.11.1.

Source: DOE/EH-4.2.1.2-Criteria

The following is taken from DOE-STD-3009-94, Chg 3.

DOE-STD-3009-94, Chg 3, appendix A, "Evaluation Guideline," provides detailed guidance on evaluating dispersion and consequences. Appendix A states

- the EG is 25 rem TEDE to the maximally exposed offsite individual (MOI) at the site boundary;
- the 95th percentile of the distribution of the doses to the MOI is used;
- the method for dispersion and transport should be consistent with Nuclear Regulatory Guide 1.145, revision 1, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, using hourly average meteorological data, straight-line Gaussian dispersion (exposure duration <8 hours);
- exposure starts when the plume reaches the MOI; and
- for elevated releases, the plume touchdown can occur beyond the site boundary.

Accidents with unique dispersion characteristics, such as explosions, may be modeled using phenomenon-specific codes more accurately representing the release conditions. Discussion should be provided justifying the appropriateness of the model to the specific situation. For accident phenomena defined by weather extremes, actual meteorological conditions associated with the phenomena may be used for comparison to the EG.

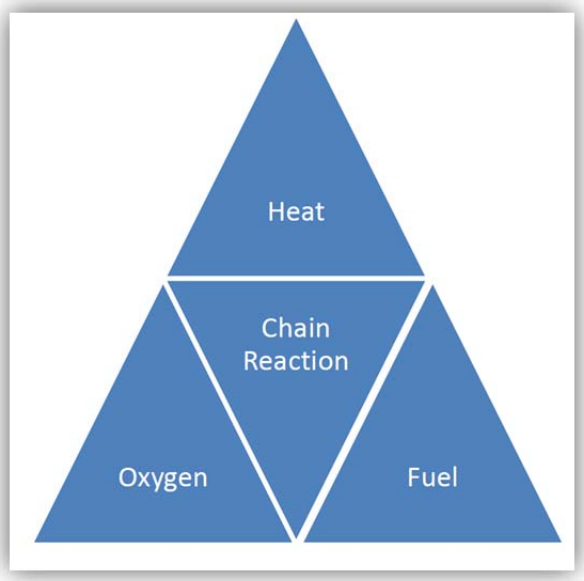
h. Discuss the physics of fires and explosions as the means of generating airborne plumes of hazardous materials and damaging barriers to releases. Also, describe how the physics affects the quantities of or rate at which hazardous materials may become airborne as a result of spills, evaporation, entrainment, fires, and other accidents.

Physics of Fire

The following is taken from DOE-HDBK-1081-YR (Draft).

The fire tetrahedron shows that for combustion to occur, fuel, an oxidizing agent, a heat source, and an uninhibited chain reaction all need to be present in the same place at the same

time. If any one of the legs of the tetrahedron is removed, the fire will be extinguished. See figure 12 for the new fire tetrahedron.



Source: DOE-HDBK-1081-YR (Draft)
Figure 12. Fire tetrahedron

OXIDIZING AGENTS
An oxidizing agent is a chemical substance that gives up oxygen easily, removes hydrogen from another substance, or attracts electrons. By far the most common oxidizing agent is the oxygen in the earth's atmosphere. However, there are many chemical compounds that act as oxidizing agents. Some of these agents react with fuels more readily and violently than oxygen.

FUEL
A fuel is the substance that reacts with the oxidizing agent during combustion. Fuels can be solids, liquids, gases, and even metals. Familiar fuels are coal, firewood, and gasoline.

HEAT SOURCE
Normally, a heat source such as a flame or spark is required to ignite a mixture of a fuel and oxidizing agent. That is, heat needs to be added or the fuel and oxidizing agent will not react. The fuels (listed above) react so readily with oxygen that a heat source is not required for ignition.

UNINHIBITED CHAIN REACTION
Once the fire is started this exothermic reaction feeds heat back to the fuel to produce the gaseous fuel used in the flame. In other words, the chain reaction provides the heat necessary to maintain the fire.

Video 20. Fire tetrahedron

http://www.bing.com/videos/search?q=fire+tetrahedron&qs=n&form=QBVR&pq=fire+tetrahedron&sc=8-16&sp=-1&sk=#view=detail&mid=5BE0E239BA613D7D97975BE0E239BA613D7D9797

Physics of Explosion

The following is taken from Ask.com, *What is the Physics of an Explosion?*

An explosion is described in physics as a speedy increase in volume and discharge of energy in an intense manner, typically with the production of high temperatures and liberation of gases. Explosions may occur in nature, and most natural explosions emerge due to volcanic processes of different types.

The following is taken from DOE-HDBK-3010-94.

The main types of accidents of common concern in nonreactor nuclear facilities include the following:

- Spill—Material experiences instability/shear stress at the surface of the mass resulting in sub-division of the overall mass. Airflow patterns around and through the material mass, including induced turbulence, accelerate overall subdivision. Mass breakup is further enhanced by impact with ground surface. The material subdivision can generate particles sufficiently small that they remain airborne for a significant period of time.
- Fire—Generates heat and combustion gases that may destroy/stress the radioactive material and/or the substrate on which radioactive materials may be deposited, compromised barriers, and/or pressurize containers/enclosure that may lead to the airborne release of contained radioactive materials. Mass flux of vapors from the reacting surfaces suspend material in the air. This material is then entrained in general convective currents that provide transport for particulate materials.
- Explosion—Generates shock and blast effects with potential for gas flow subsequent to the explosive event that may subdivide/deagglomerate and entrain material. Explosive reactions may result from chemical or physical reactions. Shock waves are supersonic pressure waves that can transmit an impulse to materials and the surrounding structures resulting in shattering of solid items. Shock waves are a true wave phenomenon and involve little gross motion of propagating medium. The potential for damage from shock waves has been extensively characterized. Blast effects are typically subsonic and involve material entrained in the gas flow. Blast effects are often more damaging. Blast effects are not subject to the same reflection/amplification phenomena as shock waves because they have significant momentum and inertia. The gas expanding from the explosion zone carries material from the explosion site. If the explosion is adjacent to the MARs, then blast effects can cause damage above and beyond the initial impulse loading. Some explosive reactions may be followed by chemical reactions, material vaporization, or fires that lead to substantial gas flows following the explosive event. These gas flows may entrain material. Deflagrations do not involve shock, but can simulate blast effects. Under proper conditions, deflagrations can transition to detonations and produce shock waves.

- Evaporation—Under most realistic scenarios involving the heating of aqueous solutions during postulated accidents in nonreactor fuel cycle facilities, the relative vapor pressures of the solvent and the solute preclude evaporation of the solute as a viable mechanism for the airborne release of the solute. Instead, the airborne release is postulated to result from the entrainment of minute drops of the bulk liquid formed by the mechanical disintegration of the surface of the bulk liquid. Mechanical disintegration mechanisms include bubble breakup during boiling, jet drops formed from the collapse of the crater remaining after bubble breakup, and secondary drops from the reentry of jet drops. Drops are carried to the bulk flow by convective and vapor flow away from the heated liquid. An increase in surface disruption would increase the airborne release, although capture of secondary drops by the large number of primary particles may place a limit on the release.
- Entrainment—Defined as the ratio between mass of liquid entrained/mass of vapor generated and was calculated by the correlation developed by Rozen et al in 1970.

Video 21. West Texas explosion: explaining the physics behind blast
http://www.cbsnews.com/news/west-texas-explosion-explaining-the-physics-behind-blast/

i. Discuss the phenomena and modeling of airborne dispersion of toxic materials, addressing weather effects, turbulent mixing, mixing heights, plume temperature, evolution and potential settling or plate out of particulates and aerosols, precipitation, building wake, and surface roughness effects.

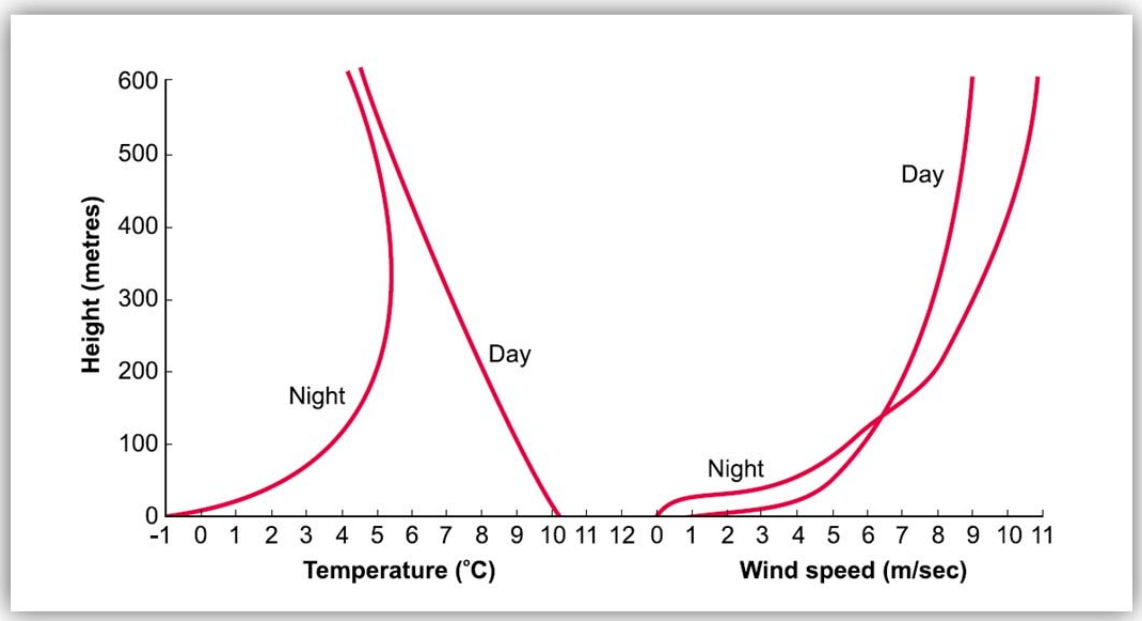
The following is taken from Stockholm Environment Institute, *Foundation on Air Quality Management in Asia,* "Modelling."

A dispersion model is a set of mathematical equations used for determining what happens to pollutants emitted in the atmosphere. Once pollutants are released into the atmosphere they are transported by air motions that lower the air pollutant concentrations in space over a period of time. Meteorological parameters averaged over one-hour time intervals are usually used to describe this phenomenon. Meteorological parameters include wind speed, wind direction, turbulence, mixing height, atmospheric stability, temperature, and inversion.

Wind

Wind is a velocity vector having direction and speed. Usually only the horizontal components of this vector are considered since the vertical component is relatively small. The wind direction is the direction from which the wind comes. Through wind speed, continuous pollutant releases are diluted at the point of release. Concentrations in the plume are inversely proportional to the wind speed.

Wind speed is expressed in the unit meter/second (m/s). At the ground wind speed must be zero. Therefore, the wind speed is lower close to the ground than at higher elevations. Figure 13 shows typical relationships between wind speed and height during day and nighttime. The wind speed (u(z)) at a vertical height (z) above the ground is a function of z and proportional to the power law (z^P) where p is an exponent that depends primarily on atmospheric stability. The exponent p varies and is approximately 0.07 for unstable conditions and 0.55 for stable conditions.



Source: Stockholm Environment Institute, Modelling

Figure 13. Variation of wind speed and temperature with height

Figure 13 also shows examples of the height dependence of temperature during day and nighttime. During the day, while the ground is heated up by solar radiation, temperature decreases linearly with height. During the night, temperature first increases with height and then starts to decrease similarly to that observed during the day.

Objects on the surface that the wind is flowing over will exert friction on the wind near to the surface. Height and spacing of the objects influence the magnitude of friction and the wind speed gradient as a function of height. The effect is described by the roughness length that ranges for urban areas between 1 and 3 m, for suburban areas approximately 0.5 to 1 m and for level areas between 0.001 m and 0.3 m.

Turbulence

Turbulence is the wind fluctuations over time scales smaller than the average time used to estimate the mean wind speed. Turbulence consists of eddies (circular movements of air) that may be oriented horizontally, vertically, or at all orientations in between. There are two kinds of turbulence: mechanical and buoyant.

Mechanical turbulence is caused by objects on the surface and by wind shear, a slower moving air stream next to a faster moving current. Mechanical turbulence increases with wind speed and the roughness length. The turbulence created by wind shear is due to the increase of wind speed with height.

Buoyant turbulence is caused by heating or cooling of air near the earth's surface. During a sunny day with clear skies and light wind, the heating of the earth's surface creates an upward heat flux that heats the lower layers of the air. The heated air goes upward and creates an upward-rising thermal stream—positive buoyant turbulence. At night with light

winds, the outgoing infrared radiation cools the ground and the lower layers of the air above while temperature at the higher air layers is unaffected. The cooling near the ground results in a net downward heat flux and leads to a temperature inversion that is a temperature structure inverted from the usual decrease of temperature with height. The temperature inversion causes the atmosphere to become stable and inhibit vertical motion—negative buoyant turbulence.

The generation of mechanical turbulence is always positive but smaller than the positive buoyant turbulence. The negative buoyant turbulence during nighttime tends to reduce mechanical turbulence.

Video 22. Sergio Garcia's evaluation of denser than air dispersion models
http://www.youtube.com/watch?v=nPF_CW-csP8

Mixing Height

During an hourly period on a sunny day, upward-rising thermal streams that characterize unstable atmospheric conditions will move in the wind direction. A series of upward and compensating downward motions will result in substantial vertical dispersion of the pollutants. Since the eddy structures point to all possible directions, there is substantial horizontal dispersion. In contrast, at night, with clear skies and light wind, a minimum of buoyant turbulence is extant, characterizing stable thermal conditions that damps out mechanical turbulence.

If the net heat flux at the ground is nearly zero, the condition is characterized as neutral. The vertical thermal structure is a slight decrease of temperature with height, approximately 10°C with each 1,000 m height increase (dry adiabatic lapse rate). Atmospherically neutral conditions can be caused by

- cloudy conditions
- windy conditions
- transitional conditions near sunrise and sunset

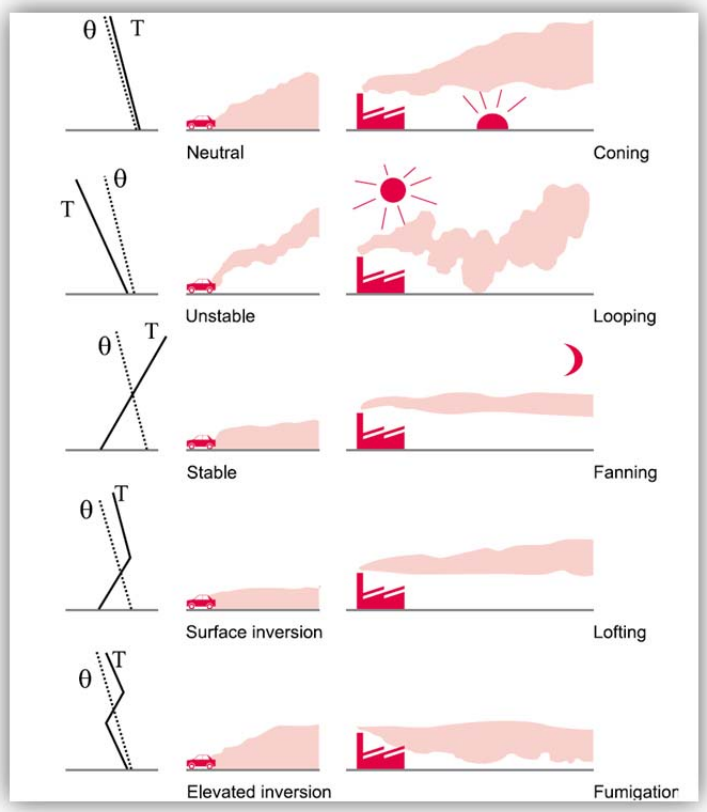
These conditions induce an intermediate level of dispersion. The mixing height is defined as the upper limit to dispersion of atmospheric pollutants. Under unstable conditions, there is a vigorous vertical mixing from the ground to approximately 1 km and then negligible vertical mixing above that height. For stable conditions the mixing height is much lower.

Stability and Plume Types

Under unstable conditions a visible continuously emitted plume will appear as large loops. (See figure 14.) This plume structure is called "looping." Looping is due to the upward motions of the heat flux and the compensating downward motions occurring as the plume is transported downward. Considerable vertical and horizontal dispersion of the effluent is taking place during every one-hour period.

Under neutral conditions turbulence is mostly mechanical. As turbulent eddies have many different orientations the resulting vertical and horizontal dispersion is relatively symmetrical. The resulting plume looks like a cone and dispersion under neutral conditions is described as "coning." (See figure 14.)

Vertical motion of the plume is inhibited under stable conditions with a temperature inversion and a low mixing height. Horizontal motion is not influenced by temperature and the horizontal extension of the plume may take many appearances. If the horizontal spreading is considerable, the plume is said to be "fanning." (See figure 14.)



Source: Stockholm Environment Institute, Modelling

Figure 14. Vertical dispersion under various conditions for low and high elevations of the source

Inversion, Fumigation, and Stagnation

By definition, an inversion exists when warmer air overlies cooler air. There are four processes to produce an inversion:

1. Cooling of a layer of air from below (surface inversion)
2. Heating of a layer of air from above (elevated inversion)
3. Flow of a layer of warm air over a layer of cold air (surface inversion)
4. Flow of a layer of cold air under a layer of warm air (elevated inversion)

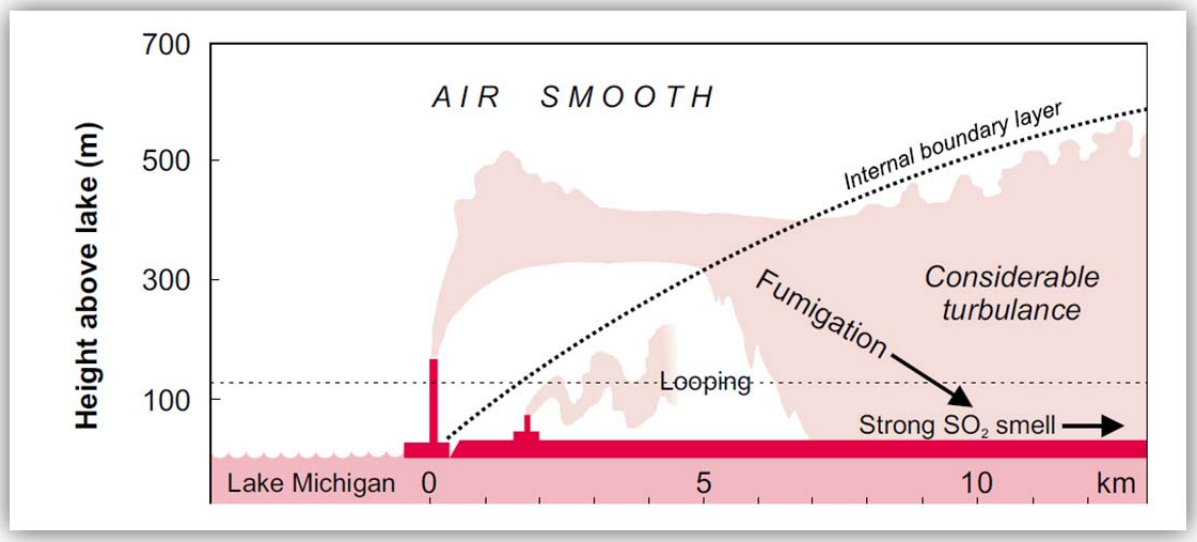
All of these occur, although the first process (cooling from below) is more common.

In air pollution, fumigation is defined as the appearance at ground level of pollutants that were previously

- in a poorly dispersed smoke plume
- trapped in a temperature inversion
- trapped between two inversion layers as a result of turbulence

Video 23. Wildfire smoke and aerosol dispersion
http://www.youtube.com/watch?v=hOX1Mr0LHhc

An example is the turbulence arising from early morning heating of the earth's surface by the sun. The subsequent gradual warming of the atmosphere from the ground upward "burns off" the inversion that builds up during nighttime. As the strong convective mixing reaches a fanned plume, it immediately mixes the large pollutant concentrations of the plume towards ground level "fumigating" that area. This process leads to high but short-term ground level concentrations. The effect is particularly strong if the plume from a shoreline source is carried inland by a stable onshore breeze. As the breeze passes inland, it encounters warmed air, increasing the convectional flow. A strong breeze will prevent the plume from mixing upward; and the convectional flow will drive air pollutants to the ground further inward. (See figure 15.)



Source: Stockholm Environment Institute, Modelling

Figure 15. Fumigation

Local phenomena such as mountain-valley winds, sea breezes, and other processes occurring at a particular place, all affect the transport of the pollutants. High concentration can accumulate within a city basin due to these phenomena. There are three reasons why the local circulation systems are not good pollution ventilators. First, the speed of these local circulations is usually rather low. Second, they are closed circulation systems. Third, they exhibit a diurnal reversal in direction of flow. The latter two factors mean that there is little true air exchange. Instead of the flow replacing dirty air with clean air, there is a back-and-forth slurrying movement involving a rather limited volume of contaminated air.

The city thermal wind system that is thought to exist with large scale stagnation, is particularly dangerous because the system is totally self-contained over an area of densely packed sources. The low-level flow converges on the city center from all directions, rises, diverges aloft, and then moves outward to subside on the urban/rural fringe and rejoins the inflow.

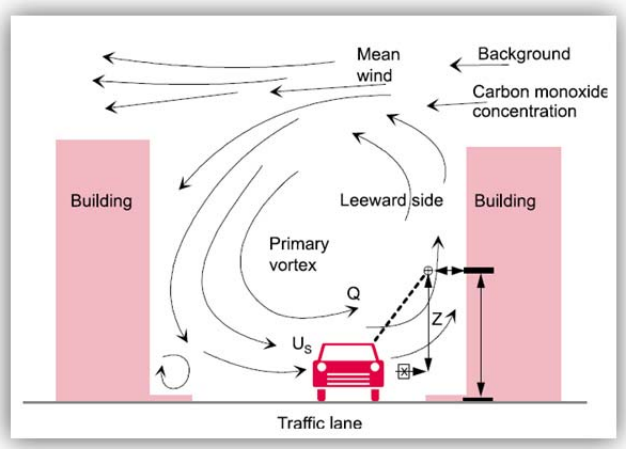
On a wider scale, regional phenomena such as dust storms and haze from fires used to clear forests add to the local air pollution in many countries. Dust storms contain large amounts of mineral dust aerosols in most parts of the Asian region and are referred to as yellow sand events in East Asia. Dust aerosols interact with urban air particulate pollutants; thus, black carbon particles are mixed with dust during a yellow sand event and are transported across the Pacific Ocean. Dust storms impair visibility, pose health hazards, and cause a strong perturbation to the atmospheric radiation budget.

Influence of Topography on Wind Speed and Direction

Topographical characteristics such as mountains, valleys, and urban areas all influence the diffusion of stack plumes and releases from low-level sources. For a plume, the centerline may become distorted and have directions completely different from the main wind direction above the topographical influences. In mountainous regions and valleys wind speed and direction may change substantially from one location to another.

In urban areas, wind speed and direction may be quite complicated in a street canyon. In addition to the topographical characteristics of the area, atmospheric and surface thermal characteristics influence air motions, particularly at low wind velocities.

Local wind velocities may be greater or lesser than would otherwise occur in the absence of heat emissions from buildings. It can easily be inferred that the airflow around urban structures will be completely different from the airflow in rural areas. Correspondingly, the dispersion of air pollutants is a complex phenomenon and concentrations within a street canyon vary for different wind directions above the urban structures and different shapes of the buildings. (See figure 16.)



Source: Stockholm Environment Institute, Modelling

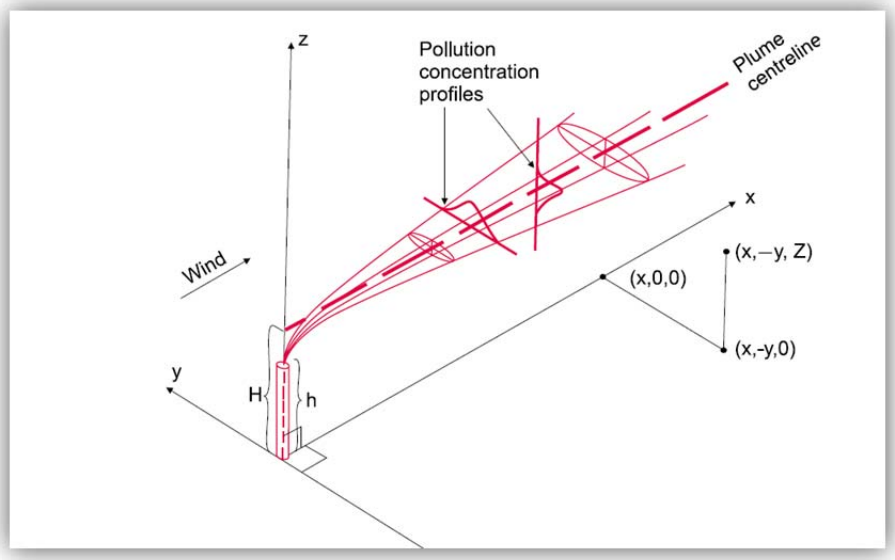
Figure 16. Wind pattern in a street canyon

Video 24. Wind engineering, wind field and pollution dispersion
http://www.youtube.com/watch?v=OqRUtRytanI

Gaussian Dispersion Models

The Gaussian model is based on the following assumptions:

- Continuous emissions (mass/time unit, usually g/s).
- Conservation of mass.
- Steady-state meteorological conditions for the travel time of pollutant from source to receptor.
- Concentration profiles in the crosswind direction and in the vertical direction (both perpendicular to the path of transport) are represented by Gaussian or normal distribution. (See figure 17.)



Source: Stockholm Environment Institute, Modelling

Figure 17. The Gaussian plume model

A Gaussian model is the solution of the basic equations for transport and diffusion in the atmosphere assuming stationary in time and complete homogeneity in space. A Gaussian dispersion model is normally used for considering a point source such as a factory smoke stack. It attempts to compute the downward concentration resulting from the point source. The origin of the coordinate system is placed at the base of the stack, with the x axis aligned in the downwind direction. The contaminated gas stream that is normally called "plume," is shown rising from the smokestack and then leveling off to travel in the x direction and spreading in the y and z directions as it travels.

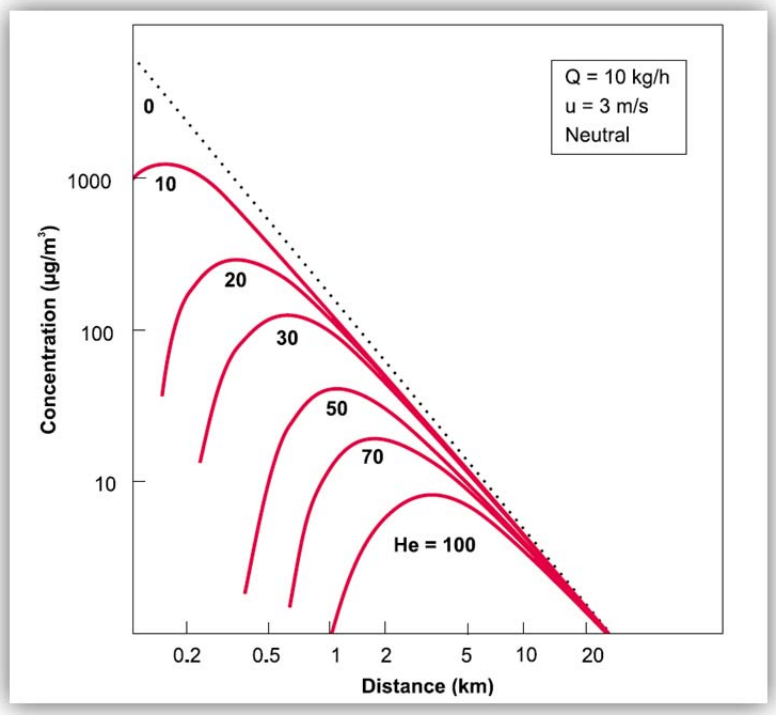
The plume normally rises to a considerable height above the stack because it is emitted at a temperature higher than that of ambient air and with a vertical velocity component. For Gaussian plume calculation, the plume is assumed to start from a point with coordinates (0,0,H), where H is called the effective stack height and is the sum of physical stack height (h) and the plume rise (dh).

It should be kept in mind that the Gaussian plume approach tries to calculate only the average values without making any statement about instantaneous values. The results obtained by Gaussian plume calculations should be considered only as averages over periods of at least ten minutes, and preferably one-half to one hour. The Gaussian plume model so far allows one to estimate the concentration at a receptor point due to a single emission source for a specific meteorology. In this form, the model is frequently used to estimate maximum concentrations to be expected from single isolated sources:

$$\chi^{(t)} = Q/(2\pi \cdot u \cdot \sigma_y \cdot \sigma_z) \cdot \exp\{-y^2/(2 \cdot \sigma_y^2)\} \cdot [\exp\{-H-z)^2/(2 \cdot \sigma_z^2)\} + \exp\{-(H+z)^2/(2 \cdot \sigma_z^2)\}]$$

In this equation $\sigma_y = a \cdot x^p$ is the standard deviation of the concentration distribution in the crosswind direction, in [m] at the downward distance x; and $\sigma_z = b \cdot x^q$ is the standard deviation of the concentration distribution in the vertical direction, in [m], at the downward distance x. a, b, p, and q are constants depending on the stability of the atmosphere.

Figure 18 presents a typical result of a concentration simulation with the Gaussian model.



Source: Stockholm Environment Institute, Modelling

Figure 18. Simulated concentrations of a point source at ground level (effective stack height H in meters)

Gaussian plume models are applied to estimate multi-source urban concentrations. The procedure is to estimate the concentration at various locations for each of the point, area, and line sources in the city for each meteorological condition and then sum up over all sources, all wind directions, all wind speeds, and all stability classes, weighted by the frequency of their occurrence.

Video 25. Contaminant plume modeling
http://www.youtube.com/watch?v=1nbOnKHaPbg

j. Discuss the mechanisms involved in the damage caused by extreme natural phenomena including hurricanes, tornadoes, ice storms, wind, flood, earthquakes, and wild fires.

The following is taken from Wikipedia, *Severe Weather*.

Organized severe weather occurs from the same conditions that generate ordinary thunderstorms: atmospheric moisture, lift, and instability. A wide variety of conditions cause severe weather. Floods, hurricanes, tornadoes, and thunderstorms are considered to be the most destructive weather-related natural disasters. Although these weather phenomena are all related to cumulonimbus clouds, they form and develop under different conditions and geographic locations.

Thunderstorms

Severe thunderstorms can be assessed in three different categories: approaching severe, severe, and significantly severe. These categories include the following attributes:

- Approaching severe—Hail between ½ to 1 inch (13 to 25 millimeter [mm]) diameter or winds between 50 and 58 mph (50 knots). In the United States (U.S.), such storms will usually warrant a significant weather alert.
- Severe—Hail 1 inch (25 mm) diameter or larger, winds 58 mph or stronger, or a tornado.
- Significant severe—Hail 2 inches (51 mm) in diameter or larger, winds 75 mph (65 knots) or stronger, a tornado of strength Enhanced Fujita 2 or stronger, the occurrence of flash flood phenomena by heavy precipitation, or extreme temperatures.

Severe and significant severe events warrant a severe thunderstorm warning from the U.S. National Weather Service (excluding flash floods), the Environment Canada, the Australian Bureau of Meteorology, or the Meteorological Service of New Zealand if the event occurs in those countries. If a tornado is occurring or is imminent, the severe thunderstorm warning will be superseded by a tornado warning in the U.S. and Canada.

Video 26. How do thunderstorms form?
http://www.youtube.com/watch?v=Xou6-iwfeNk

High Winds

High winds are known to cause damage, depending on their strength. Wind speeds as low as 23 knots (43 km/hour [km/h]) may lead to power outages when tree branches fall and disrupt power lines.

Wind gusts may cause poorly designed suspension bridges to sway. When wind gusts harmonize with the frequency of the swaying bridge, the bridge may fail as occurred with the Tacoma Narrows Bridge in 1940.

Hurricane-force winds, caused by individual thunderstorms, thunderstorm complexes, tornadoes, extra tropical cyclones, or tropical cyclones can destroy mobile homes and

structurally damage buildings with foundations. Winds of this strength due to downslope winds off terrain have been known to shatter windows and sandblast paint from cars.

Once winds exceed 135 knots (250 km/h) within strong tropical cyclones and tornadoes, homes completely collapse, and significant damage is done to larger buildings. Total Saffir-Simpson scale for cyclones and Enhanced Fujita scale for tornadoes were developed to help estimate wind speed from the damage they cause.

Tornado

A tornado is a dangerous rotating column of air in contact with the surface of the earth and the base of a cumulonimbus cloud or a cumulus cloud, in rare cases. Tornadoes come in many sizes but typically form a visible condensation funnel whose narrowest end reaches the earth and are surrounded by a cloud of debris and dust.

Tornadoes wind speeds generally average between 40 mph (64 km/h) and 110 mph (180 km/h). They are approximately 250 feet (76 m) across and travel a few miles before dissipating. Some attain wind speeds in excess of 300 mph (480 km/h), may stretch more than a mile (1.6 km) across, and maintain contact with the ground for dozens of miles (more than 100 km).

Tornadoes, despite being one of the most destructive weather phenomena are generally short-lived. A long-lived tornado generally lasts no more than an hour, but some have been known to last for two hours or longer. Due to their relatively short duration, less information is known about the development and formation of tornadoes.

Video 27. Understanding how tornadoes form
http://www.youtube.com/watch?v=2ajShQpzBus

Downburst

Downbursts are created within thunderstorms by significantly rain-cooled air that, on reaching ground level, spreads out in all directions and produces strong winds. Unlike winds in a tornado, winds in a downburst are not rotational but are directed outward from the point where they strike land or water. Dry downbursts are associated with thunderstorms with very little precipitation, while wet downbursts are generated by thunderstorms with large amounts. Microbursts are very small and macrobursts are large-scale downbursts. The heat burst is created by vertical currents on the backside of old outflow boundaries and squall lines where rainfall is lacking. Heat bursts generate significantly higher temperatures due to the lack of rain-cooled air in their formation. Derecho are longer, usually stronger, forms of downburst winds characterized by straight-lined windstorms.

Video 28. Thunderstorm downburst winds
http://www.youtube.com/watch?v=GtrJs90N0f8

Squall Line

A squall line is an elongated line of severe thunderstorms that can form along or ahead of a cold front. The squall line typically contains heavy precipitation, hail, frequent lightning, strong straight-line winds, and possibly tornadoes or waterspouts. Severe weather in the form of strong straight-line winds can be expected in areas where the squall line forms a bow echo,

in the farthest portion of the bow. Tornadoes can be found along waves within a line echo wave pattern where mesoscale low pressure areas are present. Some summer bow echoes are called derechos, and move quickly over large territories. A wake low or a mesoscale low pressure area forms behind the rain shield (a high pressure system under the rain canopy) of a mature squall line and is sometimes associated with a heat burst.

Squall lines often cause severe straight-line wind damage, and most non-tornadic wind damage is caused from squall lines. Although the primary danger from squall lines is straight-line winds, some squall lines contain weak tornadoes.

Video 29. Squall lines
http://www.youtube.com/watch?v=2bJyKOGiljc

Tropical Cyclone (Hurricanes)

A tropical cyclone (called hurricanes in the U.S. and Canada) is a storm system characterized by a low pressure center and numerous thunderstorms that produce strong winds and flooding rain. A tropical cyclone feeds on heat released when moist air rises, resulting in condensation of water vapor contained in the moist air. Tropical cyclones may produce torrential rain, high waves, and damaging storm surge. Heavy rains produce significant inland flooding. Storm surges may produce extensive coastal flooding up to 25 miles from the coastline.

Very high winds can be caused by mature tropical cyclones. A tropical cyclone's heavy surf created by such winds may cause harm to marine life either close to or on the surface of the water, such as coral reefs. Coastal regions may receive significant damage from a tropical cyclone while inland regions are relatively safe from the strong winds, due to their rapid dissipation over land. However, severe flooding can occur even far inland because of high amounts of rain from tropical cyclones and their remnants.

Although cyclones take an enormous toll in lives and personal property, they are important factors in the precipitation regimes of areas they impact. They bring much-needed precipitation to otherwise dry regions. Areas in their path can receive a year's worth of rainfall from a tropical cyclone passage. Tropical cyclones can relieve drought conditions. They can carry heat and energy away from the tropics and transport them toward temperature latitudes, making them an important part of the global atmospheric circulation mechanism. As a result, tropical cyclones help to maintain equilibrium in the earth's troposphere.

Video 30. Hurricanes
http://www.bing.com/videos/search?q=Science+of+hurricanes&qs=n&form=QBVR&pq=science+of+hurricanes&sc=8-19&sp=-1&sk=#view=detail&mid=043744F6FC285AE47B2A043744F6FC285AE47B2A

Wildfires

The most common cause of wildfires varies throughout the world. In the U.S. and Canada lightning is the major source of ignition. In other parts of world, human involvement is a major contributor. Wildfires have a rapid, forward rate of spread when burning through dense, uninterrupted fuels. They can move as fast as 6.7 mph in forests and 14 mph in

grasslands. Wildfires can advance tangential to the main front to form a flanking front, or burn in the opposite direction of the main front by backing.

Wildfires may spread by jumping or spotting as winds and vertical convection columns carry firebrands (hot wood embers) and other burning materials through the air over roads, rivers, and other barriers that may otherwise act as firebreaks. Torching and fires in tree canopies encourage spotting, and dry ground fuels that surround a wildfire are especially vulnerable to ignition from firebrands. Spotting can create spot fires as hot embers and firebrands ignite fuels downwind from the fire. In Australian bushfires, spot fires are known to occur as far as 6 miles from the fire front. Since the mid-1980s, earlier snowmelt and associated warming has been associated with an increase in length and severity of the wildfire season in the western United States.

Heavy Rainfall and Flooding

Heavy rainfall can lead to a number of hazards, most of which are floods or hazards resulting from floods. Flooding is the inundation of areas that are not normally under water. Flooding is typically divided into three classes: river flooding that relates to rivers rising outside their normal banks; flash flooding that is the process where a landscape, often in urban and arid environments, is subjected to rapid floods; and coastal flooding that can be caused by strong winds from tropical or non-tropical cyclones.

Meteorologically, excessive rains occur within a plume of air with high amounts of moisture that is directed around an upper level cold-core low or a tropical cyclone. Flash flooding can frequently occur in slow-moving thunderstorms and are usually caused by the heavy liquid precipitation that accompanies it. Flash floods are most common in dense populated urban environments, where less plants and bodies of water are present to absorb and contain the extra water. Flash flooding can be hazardous to small infrastructure, such as bridges, and weakly constructed buildings. Plants and crops in agricultural areas can be destroyed and devastated by the force of raging water. Automobiles parked within experiencing areas can be displaced. Soil erosion can occur as well, exposing risks of landslide phenomena. Like all forms of flooding phenomenon, flash flooding can spread and produce waterborne and insect-borne diseases caused by microorganisms. Flash flooding can be caused by extensive rainfall released by tropical cyclones of any strength or the sudden thawing effect of ice dams.

Video 31. How floods work

http://www.bing.com/videos/search?q=the+science+of+floods&qs=n&form=QBVR&pq=the+science+of+floods&sc=8-13&sp=-1&sk=#view=detail&mid=4CDD6CFB9108F77916914CDD6CFB9108F7791691

Heavy Snowfall

When extratropical cyclones deposit heavy, wet snow with a snow-water equivalent ratio of between 6:1 and 12:1 and a weight in excess of 10 lbs/sq. ft. piles onto trees or electricity lines, significant damage may occur on a scale usually associated with strong tropical cyclones. An avalanche can occur with a sudden thermal or mechanical impact on snow that has accumulated on a mountain, causing the snow to rush downhill suddenly. Preceding an avalanche is a phenomenon known as an avalanche wind caused by the approaching

avalanche itself that adds to its destructive potential. Large amounts of snow that accumulate on top of man-made structures can lead to structural failure. During snowmelt, acidic precipitation that previously fell in the snow pack is released and harms marine life.

Lake-effect snow is produced in the winter in the shape of one or more elongated bands. This occurs when cold winds move across long expanses of warmer lake water, providing energy and picking up water vapor that freezes and is deposited on the lee shores.

Conditions within blizzards often include large quantities of blowing snow and strong winds that may significantly reduce visibility. Reduced viability of personnel on foot may result in extended exposure to the blizzard and increase the chance of becoming lost. The strong winds associated with blizzards create wind chill that can result in frostbite and hypothermia. The strong winds present in blizzards are capable of damaging plants and may cause power outages, frozen pipes, and cut-off fuel lines.

Ice Storms

Ice storms are known as a silver storm, referring to the color of the freezing precipitation. Ice storms are caused by liquid precipitation that freezes on cold surfaces and leads to the gradual development of a thickening layer of ice. The accumulations of ice during the storm can be extremely destructive. Trees and vegetation can be destroyed and in turn may bring down power lines, causing the loss of heat and communication lines. Roofs of buildings and automobiles may be severely damaged. Gas pipes can become frozen or damaged, causing gas leaks. Avalanches may develop due to the extra weight of the ice. Visibility can be reduced dramatically. The aftermath of an ice storm may result in severe flooding due to sudden thawing, with large quantities of displaced water, especially near lakes, rivers, and bodies of water.

Earthquakes

The following is taken from the Science Daily, *Earthquake*.

An earthquake is the result of a sudden release of stored energy in the earth's crust that creates seismic waves. The magnitude of an earthquake is conventionally reported using the Richter scale or a related moment scale. At the earth's surface, earthquakes may manifest themselves by a shaking or displacement of the ground. Sometimes they cause tsunamis that may lead to loss of life and destruction of property.

An earthquake is caused by tectonic plates getting stuck and putting a strain on the ground. The strain becomes so great that rocks give way by breaking and sliding along fault planes. Earthquakes may occur naturally or as a result of human activities. Smaller earthquakes can be caused by volcanic activity, landslides, mine blasts, and nuclear experiments.

The earth's lithosphere is a patchwork of plates in slow but constant motion caused by the release to space of the heat in the earth's mantle and core. The heat causes the rock in the earth to become flow on geological timescales, so that the plates move slowly but surely. Plate boundaries lock as the plates move past each other, creating frictional stress. When the frictional stress exceeds a critical value, called local strength, a sudden failure occurs. The boundary of tectonic plates along which failure occurs is called the fault plane. When the

failure at the fault plane results in a violent displacement of the earth's crust, the elastic strain energy is released and seismic waves are radiated, causing an earthquake.

In subduction zones, where older and colder oceanic crust descends beneath another tectonic plate, deep focus earthquakes may occur at much greater depths. These are earthquakes that occur at a depth where the subducted lithosphere should no longer be brittle, due to the high temperature and pressure. A possible mechanism for the generation of deep focus earthquakes is faulting caused by olivine undergoing a phase transition into a spinel structure. Earthquakes may occur in volcanic regions and are caused by tectonic faults and by the movement of magma in volcanoes.

Such earthquakes can be an early warning of volcanic eruptions. A recently proposed theory suggests that some earthquakes may occur in a sort of earthquake storm where one earthquake will trigger a series of earthquakes each triggered by the previous shifts on the fault lines, similar to aftershocks, but occurring years later, and with some of the later earthquakes as damaging as the early ones.

k. Define and discuss the following terms:

- **Chi/Q**
- **Dose conversion**
- **Breathing rate**
- **Aerodynamic equivalent diameter**
- **Solubility class**
- **Population dose**

Chi/Q

The following is taken from the Rocky Mountain Arsenal, Woodward Clyde Consultants, *Public Health Risk Assessment Report, Submerged Quench Incinerator Task IRA-2 Basin F Liquids Treatment Design*.

The maximum concentrations/emission values (denoted by the symbol Chi/Q) are used to calculate contaminant concentrations. The Chi/Q value occurring at the exposure point is multiplied by the emission rate to determine the contaminant concentration at the exposure point.

$$\text{Chi/Q} = \text{Ambient air dispersion modeling factor} \left(\frac{mg/m^3}{g/sec}\right)$$

Dose Conversion Factor

The following is taken from the Berkeley University, Department of Nuclear Engineering, *Berkeley Radiological Air and Water Dose Calculation*.

The dose conversion factor relates the activity of a radionuclide (in Bq or uCI) to the TEDE received by the person (in millirem or uXv).

Dose conversion factors can be obtained from tabulated data in the EPA Federal Guidance Report No. 11, in ICRP 30, in the supplement to ICRP 30, part 1, or calculated directly using computer programs.

Breathing Rate

The following is taken from Wikipedia, *Respiratory Rate*.

The respiratory rate (breathing rate) is the frequency of ventilation, that is, the number of breaths (inhalation—exhalation cycles) taken within a set amount of time, typically 60 seconds. A normal respiratory rate is termed eupnoea, an increased respiratory rate is termed tachypnea and a lower than normal respiratory rate is termed bradypnea.

Human respiration rate is measured when a person is at rest and involves counting the number of breaths for one minute by counting how many times the chest rises. An optical breath rate sensor can be used for monitoring patients during a magnetic resonance imaging scan. Respiration rates may increase with fever, illness, or other medical conditions. When checking respiration, it is important to note whether a person has any difficulty breathing.

The average respiratory rate reported in a healthy adult at rest is usually given as 12–18 breaths per minute but estimates do vary between sources (e.g., 12–20 breaths per minutes, 10–14, between 16–18, etc.). With such a slow rate, more accurate readings are obtained by counting the number of breaths over a full minute. Average resting respiratory rates per minute by age are as follows:

- Birth to 6 weeks: 30–60
- 6 months: 25–40
- 3 years: 20–30
- 6 years: 18–25
- 10 years: 15–20
- Adults: 12–20

Respiratory minute volume is the volume of air that is inhaled or exhaled from the lungs in one minute.

Video 32. Respiratory rate
http://www.youtube.com/watch?v=d_kTJSuqeco

Aerodynamic Equivalent Diameter

The following is taken from DOE-HDBK-3010-94.

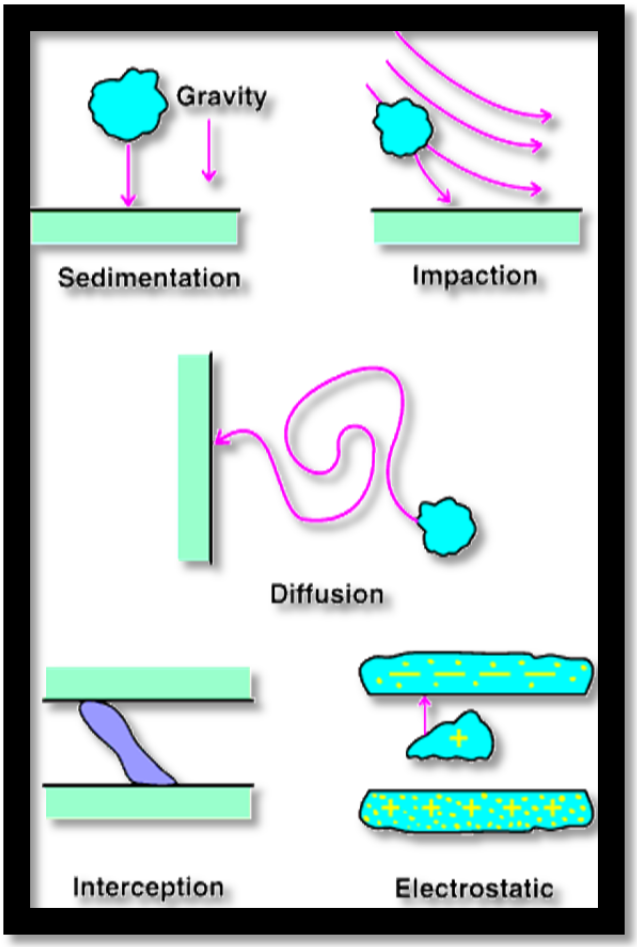
Aerodynamic equivalent diameter (AED) is the diameter of a sphere of density $1 g/cm^3$ that exhibits the same terminal velocity as the particle in question.

The following is taken from Radiation Resources, *How Particles Deposit in the Respiratory Tract*.

Where inhaled particles are likely to deposit in the respiratory tract depends on the aerodynamic properties of the particles and the geometry of the respiratory tract of the person of interest. A key aerodynamic property is the aerodynamic size. For airborne particles, the aerodynamic size depends on the shape and density of the particle. The shapes and densities of airborne radioactive particles depend on the exposure scenario considered.

The aerodynamic size is different from the actual size and accounts for the aerodynamic behavior of an aerosol. Usually AED is used to indicate the aerodynamic size. The AED is the diameter of a sphere, with density=1 g/cm^3, that has the same terminal settling velocity under gravity as the airborne particle considered.

There are different ways by which airborne particles can deposit in the respiratory tract. The five most important ways are sedimentation, impaction, Brownian diffusion, interception, and electrostatic precipitation. The five ways of deposition are summarized in figure 19.



Source: Radiation Resources, How Particles Deposit in the Respiratory Tract

Figure 19. The five ways of airborne particle deposition

Sedimentation—Represents deposition caused by gravity. The chance of particle deposition in the respiratory tract by sedimentation increases as the particle size, particle density, and length of time spent in the airway increase. Airway resident time increases as the breathing rate slows. Respiratory tract deposition by sedimentation is important for particles with an aerodynamic diameter greater than 0.5 µm that reach the medium-sized to small bronchi and bronchioles, where air velocity is relative low.

Impaction—When the aerodynamic diameter is larger than 1 µm, inhaled particles can deposit in the nose, pharynx, and mouth. Deposition by impaction can occur when the particle momentum is too large for it to change directions in an area where there is a rapid change in the direction of the bulk airflow. The chance of impaction increases as the air velocity, particle size, and particle density increase. Air velocity increases as the breathing rate increases. The breathing rate increases as the level of physical activity increases.

Brownian diffusion—For particles with an aerodynamic diameter less than 1 µm, Brownian diffusion is a major way for deposition in airways where the bulk flow is very low or absent. With Brownian diffusion, airborne particles acquire a random motion because of their bombardment by surrounding air molecules; this motion can result in particle contact with an airway wall. The displacement sustained by a particle depends on a parameter called the diffusion coefficient that increases as the particle size decreases. Deposition by Brownian diffusion is especially important for particles with aerodynamic diameters less than 0.2 µm. Particles in this size range may also deposit by diffusion in the upper respiratory tract, trachea, and larger bronchi. Deposition in the respiratory tract by Brownian diffusion is unimportant for relatively large particles.

Video 33. Brownian motion and diffusion
http://gsis.mediacore.tv/media/brownian-motion-and-diffusion

Interception—Particle deposition in the respiratory tract can occur when the edge of the particle contacts the airway wall. For elongated particles, interception is an important respiratory tract deposition mechanism. The chance of particle interception increases as the airway diameter becomes smaller.

Electrically charged particles—Inhaled particles can be electrically charged. If so, they can exhibit greater regional deposition over what would be expected based on their size, shape, and density.

Video 34. Industrial hygiene air sampling for uranium dust
http://www.youtube.com/watch?v=MZOBzog3vw8

Solubility Class

The following is taken from Los Alamos National Laboratory (LANL), BEAR Conference, *Uranium Lung Solubility Class Selection at Bechtel Jacobs Company LLC-Operated Facilities*.

DOE regulations are currently based on the use of the ICRP 30 dosimetric model for the respiratory system. Under this model, materials are classified as D, W, or Y to describe the

clearance of inhaled radioactive materials from the lung. These designations refer to the length of time particles from inhaled aerosols are retained in the pulmonary region: D representing days, W representing weeks, and Y representing years. The model takes account of particle sizes within the range of 0.2 µm to 10 µm, although 1 µm activity median aerodynamic diameter is recommended as the default size.

Population Dose

The following is taken from DOE-HDBK-1122-99 (Archived).

ALARA dictates awareness of changes in radiation exposure to the general population that results from nuclear operations. Issuing thermoluminescent dosimeters (TLDs) to the population is not practical. In addition the TLDs are not sensitive enough to detect changes in environmental radiation levels. The only practical way to determine population exposure is by measurement of environmental radiation levels:

- External radiation level
- Radioactivity present in the air
- Radioactivity present in food
- Radioactivity present in water

Population exposure can then be determined by using these values combined with knowledge of the drinking water sources and the types of food consumed in the region.

l. Given a source term, determine dose consequences applying Chi/Q, dose conversion factor, breathing rate, and specific activity as applicable.

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

m. Given a simple accident scenario, demonstrate knowledge by constructing a simple neutral gas dispersion and heavy gas dispersion. Estimate consequences using an accident modeling code including hand calculations, and explain the assumptions, inputs, and results.

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

The following taken from Dai, Weiping, *Applying Proper Dispersion Models for Industrial Accidental Releases*, may be helpful.

The Dense Gas Dispersion Model (DEGADIS) was developed specifically to model heavier-than-air gaseous releases from a single source over flat terrain. A single meteorological condition is specified for the duration of the release. DEGADIS was developed by Tom Spicer and Jerry Havens of the University of Arkansas, on commission by the U.S. Coast Guard in the 1980s. It incorporates the vertical jet plume model developed by Ooms, Mahiu, and Zelis. This is useful for chemical processes requiring the storage of pressurized substances that, if released, produce high velocity emissions. The Ooms jet plume model

provides for the prediction of the trajectory and dilution of these types of vertically oriented gas or aerosol jets. It also accounts for ground reflection when the plume's lower boundary reaches the ground. Overall, DEGADIS models the following release types:

- Continuous release—A steady-state release of dense gas at a constant rate in the atmosphere over a long period of time. The output from modeling a steady-state release is concentration estimates at various downwind distances determined by the model.
- Finite duration release—A steady-state release of dense gas at a constant rate in the atmosphere over a short period of time. Finite duration model output is organized either by time or distance, depending on which parameter is of greater interest.
- Transient release—Release rates vary over time, for example, if a liquid pool boils off or a container of gas depressurizes. As the pool decreases in size, the emission rate and radius change. Other transient releases include near-instantaneous releases such as container ruptures. Transient modeling output is organized either by time or distance, depending on which parameter is of most interest.
- Jet release—A vertical release of a dense gas or aerosol by using the Ooms mathematical model. The jet plume model requires that the jet be vertical, with a definable exit velocity. If the jet release is such that the plume centerline does not reach the ground before dispersing, the jet plume model is run alone. If this is unclear, or if the plume centerline does reach the ground, the jet plume model is run in conjunction with the regular DEGADIS model as either a continuous or finite duration release.
- Liquid spill—A release of a chemical in its liquid state. The liquid is assumed to form a pool at ground level, with the evaporation rate calculated using one of three different evaporation models incorporated in DEGADIS. The results from the evaporation model are run as either a continuous or finite duration release. Note that the liquid spill option will only be available if the chemical's normal boiling point is greater than the ambient temperature.

Furthermore, the DEGADIS model has the following assumptions and limitations that govern the model execution:

- Use of the model is restricted to dense gas releases or liquid spills that evaporate to a dense gas.
- The DEGADIS and the jet plume models assume a flat atmospheric flow field with no obstructions, such as buildings or trees. The model does not consider sloping terrain either. Use of the model is restricted to conditions in which the depth of the dispersing gas layer is much greater than the surface roughness of the surrounding area.
- The jet model is strictly for vertical releases. No horizontal jet release velocity is incorporated in the model. If the jet release is not perpendicular to the ground, the modeling results will not be accurate.

Video 35. What is air dispersion modeling?
http://www.youtube.com/watch?v=nzH1-8fDxys

n. Discuss the processes for evaluating assumptions made for scenarios being modeled.

The following is taken from DOE-STD-3009-94, Chg 3.

Accident analysis typically starts with formal descriptions of accident scenarios. Basic ETs may support such descriptions. All major assumptions in scenarios must be identified. The next step is determination of accident source terms. Source terms for accidents are obtained through phenomenological and system response calculations. Once a source term has been determined, consequences due to atmospheric dispersion or other relevant pathways of concern are determined. If the source term is small, a simple dispersion hand calculation for consequences would be sufficient. If source terms are large, computer modeling to determine consequences may be required.

All assumptions made in the accident analysis (i.e., defining points in scenario progression) are to be validated as part of the accident analysis activity. For example, if an operator is supposed to push button Z to stop an accident progression, the accident analysis needs to make it clear that the operator can actually do so. Making it clear may simply involve noting there are no physical phenomena associated with the accident that would preclude them from doing so. Likewise, basic assurance must be provided that equipment relied on in unusual or severe environments will function. This assurance does not constitute the need for, or expectation of full, formal environmental qualification.

This guidance is not meant to imply that the DSA must contain detailed validations for all assumptions; however, the DSA needs to present information at a level that is considered sufficient for review and approval of the DSA. Referencing an auditable trail of information as part of the controlled supporting documentation is acceptable.

o. Discuss the methods used in the calculation of criticality accidents.

The following is taken from Ralph Fullwood, *Lecture Notes for Criticality Safety*.

Determining criticality begins by showing that four of the factors in the six-factor formula are equal to the neutron multiplication factor for an infinite reactor. The other two factors are related to the Fermi age, the thermal diffusion length, and buckling. Buckling, related to the size of the reactor, can be used to determine if an assembly or process is critical.

Unfortunately, there are many complex geometries and material combinations that occur in process and experimental situations than can be accurately represented by multi-parameter critical graphs. The criticality aspects of these complex geometries can be calculated with computers, but several methods have been developed that are simple enough for hand calculations.

Buckling/Shape Conversion

The buckling/shape conversion method determines the equivalent geometry for which criticality limits are available to the geometry being assessed for criticality by relating the buckling of one to the other. If the buckling is greater than the subcriticality limited buckling

shape, the unknown is deemed to be subcritical; it is deemed to be critical if its buckling is less. However, to use these formulae, the extrapolation distance is needed.

Surface Density Method

Consider a fissile inventory storeroom stacked with shipping containers and if a catastrophe occurred and the material were deposited on the floor, forming a slab, and this slab became flooded with water, the configuration should still be subcritical. This requirement that an inventory must be subcritical even in flooded slab form is the basis for the surface density method. A well-verified formulation is: $SD=0.54*SD_o*(1-1.37*f)$, where SD_o is the surface density in gm/cm^2 of a critical water reflected slab, and f the mass of a unit in the array to the critical mass of an unreflected sphere of the same material.

A storeroom stacked with such containers has a uniform spacing of material such that the surface floor density is: $SD=n*m/d^2$, where n is the number of units stacked on each other, m is the mass of fissile material contained in each container, and d^2 is the length of a side of the square base of the shipping container.

Combining these two equations, the allowable center to center spacing for the storage of fissile materials is: $d=1.37*\sqrt{\{n*m/[SD_o*(1-1.37*f)]\}}$.

Density Analog Method

The density analog method is not restricted to two-tier stacking; but, it is restricted to N being the cube of an integer. The surface density method, as presented is restricted to two-tier stacking, but may be very large in the other dimensions. For an array of 20,000 containers, the density analog method gives a minimum spacing of 123 cm, while the surface density method still limits the spacing to 66 cm provided that ACs limit the stacking to two tiers.

Solid Angle Method

Solid angle method assumes there is no neutron absorption between the fissile containers. In such a case, the key aspect of the method is the calculation of the solid angle. The solid angle method, when applied to an array of fissile packages refers to the maximum solid angle, Ω, if all of the packages have the same k (neutron multiplication factor), otherwise to the maximum product of Ω*k.

Computer Simulation of the Chain Reaction: Monte Carlo

Computer simulation begins by assuming a neutron is emitted from a weighted randomly selected location in the source region; the weighting should correspond to the spatial emission distribution. The neutron is assigned a randomly selected direction that is weighted by the angular emission distribution. The neutron's energy is randomly selected from a distribution corresponding to the emission energy spectrum. It travels through various materials corresponding to the problem for a distance randomly weighted by the transmission in the medium through which it is traveling. The type of interaction is selected randomly from the relative amounts of the competing cross sections. If the neutron is captured and only neutron history is being calculated, this trial ends and the process is repeated by selecting another source neutron.

If the neutron scatters, the scattering angle and energy are calculated from the kinematics, and the neutron travels in this new direction at this new energy until it interacts in a distance randomly weighted by the transmission in the medium through which it is traveling. The type of process is selected as before, and this procedure continues until the neutron is captured. If it is captured in a fission process, the number of neutrons emitted is randomly selected from the fission neutron emission distribution and these neutrons must be traced. A neutron may come to the boundary of the reactor and be lost in space that also terminates a history. By repeating this neutron life-death process a very large number of times, and combining the trials, the neutron flux is calculated. The process described is called an un-weighted Monte Carlo calculation. Its disadvantage is the very large number of trials needed to obtain an accurate answer. More sophisticated methods accelerate the convergence to give an accurate answer with fewer calculations.

Mandatory Performance Activities:

a. Demonstrate by participation on at least five safety basis document or amendment reviews whose major focus deals with hazard or accident analysis for the determination of adequacy of the analyses.

Mandatory performance activities are performance-based. The Qualifying Official will evaluate the completion of this activity.

6. Nuclear safety specialists must demonstrate a familiarity level knowledge of terminology associated with PRA techniques.

a. Identify the strengths and weaknesses of PRA for safety design and regulatory decision-making.

The following is taken from the Environmental Protection Agency, "Probabilistic Risk Assessment," chapter 31.

The primary advantage of PRA is that it can provide a quantitative description of the degree of variability or uncertainty (or both) in risk estimates. The quantitative analysis of uncertainty and variability can provide a more comprehensive characterization of risk than is possible in the point estimate approach.

Another significant advantage of PRA is the additional information and potential flexibility it affords the risk manager. Risk management decisions are often based on an evaluation of high-end risk to an individual—for deterministic analyses, this is generally developed by the combination of a mix of central tendency and high-end point values for various exposure parameters. When using PRA, the risk manager can select a specific upper-bound level from the high-end range of percentiles of risk, generally between the 90th and 99.9^{th} percentiles.

PRA may not be appropriate for every analysis. The primary disadvantages of PRA are that it generally requires more time, resources, and expertise on the part of the assessor, reviewer, and risk manager than a point estimate approach. The chief obstacle to using PRA in air toxics risk assessments is usually the lack of well-documented frequency distributions for many input variables.

b. Define the following terms with respect to reliability engineering and PRAs:

- **Probability**
- **Reliability**
- **Availability**
- **Unavailability**
- **Uncertainty**
- **Risk**
- **Safety**
- **Accident sequence**
- **Dominant contributors**
- **Minimal cut set**

Probability

The following is taken from DOE-HDBK-1014/2-92.

If E_1 is the number of heads, and E_2 is the number of tails, $E_1/(E_1 + E_2)$ is an experimental determination of the probability of heads resulting when a coin is flipped.

$$P(E_1) = n/N$$

By definition, the probability of an event must be greater than or equal to 0, and less than or equal to 1. In addition, the sum of the probabilities of all outcomes over the entire "event" must add to equal 1. For example, the probability of heads in a flip of a coin is 50 percent and the probability of tails is 50 percent. If we assume these are the only two possible outcomes, 50% + 50%, the two outcomes, equals 100 percent or 1.

The concept of probability is used in statistics when considering the reliability of the data or the measuring device, or in the correctness of a decision. To have confidence in the values measured or decisions made, one must have an assurance that the probability is high of the measurement being true, or the decision being correct.

To calculate the probability of an event, the number of successes (s), and failures (f), must be determined. Once this is determined, the probability of the success can be calculated by:

$$p = \frac{s}{s + f}$$

where:
s + f = n = number of tries

Reliability

The following is taken from NASA, *Reliability and Data Analytics—Viewpoints, Tips, Examples*.

The classic definition of reliability is the probability (likelihood) an item will perform its intended or required functions with no downtime during a given period of time under specified operating conditions. The emerging definition of reliability is the ability of an item to perform its intended or required functions with no downtime during a given period of time under specified operations conditions. The focus is on achievement rather than probability.

The following are taken from NASA/SP-2011-3421, *Probabilistic Risk Assessment Procedures Guide for NASA Managers and Practitioners*, unless otherwise stated.

Availability

Another area worthy of comment is the quantification of reliability and availability metrics. In commercial nuclear applications, relatively little effort is invested in time-dependent quantification of expressions; point values are used for basic event probabilities independently of possible correlation between basic event probabilities. In commercial nuclear power applications, this is arguably acceptable in many contexts, because the point of the analysis is to distinguish scenarios, and low precision will suffice. In other applications, arguably including certain applications, the actual reliability of certain systems is of some interest, and better numerical evaluations of failure probability are warranted.

Unavailability

The unavailable state includes two distinct sub-states: failed and functionally unavailable, depending on whether the cause of the unavailability is damage to the component or lack of necessary support such as motive power. The state classification recognizes that even when a component may be capable of performing its function, an incipient or degraded condition could exist in that component, or in a supporting component. These failure situations are termed potentially failed and potentially functionally unavailable, respectively. These concepts have proven useful in many PRA data applications.

Uncertainty

Estimation of model parameters involves uncertainties that need to be identified and quantified. A broad classification of the types and sources of uncertainty and potential variabilities in the parameter estimates is as follows:

- Uncertainty in statistical inference based on limited sample size.
- Uncertainty due to estimation model assumptions. Some of the most important assumptions are
  - assumption about applicable testing scheme (i.e., staggered versus non-staggered testing methods);
  - assumption of homogeneity of the data generated through specializing generic data to a specific system.
- Uncertainty in data gathering and database development. These include
  - uncertainty because of lack of sufficient information in the event reports, including incompleteness of data sources with respect to number of failure events, number of system demands, and operating hours;
  - uncertainty in translating event characteristics to numerical parameters for impact vector assessment (creation of generic database); and
  - uncertainty in determining the applicability of an event to a specific system design and operational characteristics (specializing generic database for system-specific application).

The role of uncertainty analysis is to produce an epistemic probability distribution of the common cause failure (CCF) frequency of interest in a particular application, covering all relevant sources of uncertainty from the above list. Clearly, some of the sources or types of uncertainty may be inapplicable, depending on the intended use of the CCF parameter and

the form and content of the available database. Methods for handling various types of uncertainty vary in complexity and accuracy.

The following techniques have been used for propagation of uncertainties:

- Sampling—The distributions for input parameters are mapped using crude Monte Carlo or Latin hypercube sampling techniques to obtain an empirical distribution for the output function.
- Moment propagation—First and second moments of the input parameters are mapped to obtain the mean and variance of the output function using variance/covariance propagation.
- Discrete probability distribution—The distributions for input parameters are converted to discrete probability distribution before mapping. The resulting distribution for the output function is empirical.

Risk

A very common definition of risk represents it as a set of triplets: scenarios, likelihoods, and consequences. Determining risk generally amounts to answering the following questions:

- What can go wrong?
- How likely is it?
- What are the associated consequences?

The answer to the first question is a set of accident scenarios. The second question requires the evaluation of the probabilities of these scenarios, while the third estimates their consequences. Implicit within each question is that there are uncertainties. The uncertainties pertain to whether all the significant accident scenarios have been identified, and whether the probabilities of the scenarios and associated consequence estimates have properly taken into account the sources of variability and the limitations of the available information.

Safety

In many NASA contexts, decisions regarding design features, especially safety features, are faced with competing objectives. For example, if a candidate safety system performs well but has a large mass, the decision to include it must be made carefully. Once design decisions are made, they need to be reflected in the risk-informed safety class (RISC). Not only do the features need to be modeled, in addition, the trade process itself needs to be presented in the RISC. There are good reasons for this; it shows not only the decision-makers but the risk-takers that the best possible job has been done in trading safety, and documentation of the process creates a better starting point for future design exercises.

Accident Sequence

Accident progression can be modeled using an event sequence diagram (ESD) or its derivative, an ET. Both are inductive logic models used in PRAs to provide organized displays of sequences of system failures or successes, and human errors or successes that can lead to specific end states. An ESD is inductive because it starts with the premise that some IE has occurred and then maps out what could occur in the future if the systems, or humans, fail or succeed. The ESD identifies accident sequences leading to different end states. The accident sequences form part of the Boolean logic that allows the systematic quantification of risk.

A traditional accident progression analysis begins with an ESD, refines it, and then transforms it into an ET format. The advantage of this process is that the morphology of an ESD is less rigidly structured than an ET. Hence, ESDs permit the complex relationships among IEs and subsequent responses to be displayed more readily.

Typically, one ESD is developed for each IE. The objective is to illustrate all possible paths from the IE to the end state. An ESD is a success-oriented graphic in that it is developed by considering how human actions and system responses (including software) can prevent an accident or mitigate its severity.

Dominant Contributors

The description of each dominant risk scenario along with its likelihood should be provided. The narrative should discuss the nature of initiator and system failures involved in the scenarios. The dominant contributors to each system end state should be presented. A consistent presentation scheme needs to be adapted to systematically delineate the progression of the accident starting from the initiator and all system failures and interactions that are captured in the definition of the accident scenario. The method of presentation should permit detailed technical review, including recalculation.

Minimal Cut Set

A collection of basic events whose simultaneous occurrence engenders the top event is called a cut set. Minimal cut sets (MCSs) are cut sets containing the minimum subset of basic events whose simultaneous occurrence causes the top event to occur. Boolean reduction of a FT has the objective of reducing the FT to an equivalent form that contains only MCSs. This is accomplished by sequential application of the basic laws of Boolean algebra to the original logic embodied in the FT until the simplest logical expression emerges. Quantification of the FT is the evaluation of the probability of the top event in terms of the probabilities of the basic events using the reduced Boolean expression of MCSs.

c. Define the following terms and differentiate between the associated processes:

- **Event tree**
- **Fault tree**
- **Failure modes and effects analysis**

The following is taken from DOE-STD-1027-92, Chg 1.

Event Tree

ET analysis is a simple approach to delineating sequences of events that 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 IE, 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 that succeed or fail to prevent the occurrence of the undesired event or to achieve a final safe condition. ET 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. Examples of

such sequences include fire scenarios or seismic events. In such cases, the combination of various barrier successes and failures is best represented by using ET analysis.

Fault Tree

Large, moderately complex processes include solid handling activities that include rather simple movement of materials from one discrete step to another. FTA and ET analysis techniques are appropriate for such facilities. FTA 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 FTA starts from the undesirable event and logically identifies basic fault conditions that can contribute to its occurrence, only those faults contributing to the occurrence of undesired event are modeled. This process is much more efficient than the bottom-up approach such as FMEA and is the main reason for its widespread use. FTA is most suitable for analysis of large, moderately complex systems, or processes where multiple component failures including human errors can contribute to the failure of the system or process.

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 examples of such devices. FMEA 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. FMEA 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.

FMEA has some limitations that 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. It looks at failure of equipment in different nodes and assesses their consequences but does not look at failures of a process that by its very nature, may have complexities and instabilities far beyond those that can be assessed only by examining the failure of individual components.

d. Discuss how PRA methods can help in understanding accident scenarios.

The following is taken from DOE G 421.1-1 (Archived).

A set of accidents may be postulated based on the designer/analyst knowledge of previous operating experience, incident data, previously conducted safety assessments, and engineering judgment. This technique often involved the generation of a series of “what if” questions. These postulated accidents may be quantified if accident frequency data are

available. In many cases, accident frequencies are estimated using engineering judgment. This approach offers the advantage of simplicity; but, its success is highly dependent on the experience of the designer/analyst. The results of such analyses are difficult to reproduce and defend.

The maximum credible accident approach and the DBA approach are two related techniques that may be useful in identifying scenarios and in distinguishing between those that are credible and those that are incredible. The maximum credible accident approach uses engineering judgment to identify accidents. Based on an intuitive estimate of their probabilities, the accident scenarios are divided into credible and incredible accident scenarios. The incredible accidents are not analyzed in detail. Accidents having a probability of occurrence greater than the maximum credible accident can then be analyzed in detail. This approach is typically used only to estimate the upper bound of the accident consequence potential of the particular operation and to design specific protective systems only for the maximum credible accident. As noted previously, it is important to identify as many accident scenarios as possible that potentially could lead to a criticality accident. Designer/analysts should not subjectively dismiss potential criticality scenarios as incredible when it may be possible through a design change to eliminate the scenario completely. The advantage of the maximum credible accident approach is its simplicity, while its weakness is the subjective nature of the division between credible and incredible accident scenarios and the typical treatment of only the maximum credible accident.

The DBA approach is an extension of the maximum credible accident approach. A series of accidents, including low-probability accidents with major consequences, are postulated based on various accident initiators and used as the explicit basis for design or analysis. Accidents having a lower probability of occurrence than the DBA in each accident initiator area are generally not analyzed. The DBA approach is more comprehensive than the maximum credible accident approach but the weakness remains—the subjective nature of the selection of accidents.

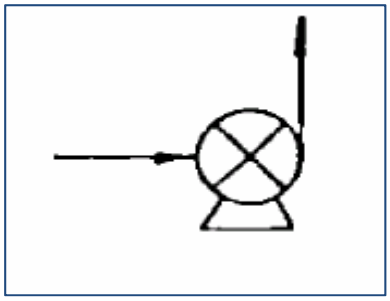
As applied to NCS, the terms “maximum credible accident” and “design basis accident” are not particularly useful except as a means to aid in distinguishing between credible and incredible accidents. Any potential criticality event, regardless of the magnitude of the initial fission burst, should be carefully analyzed and appropriate design changes made if necessary.

7. **Nuclear safety specialists must demonstrate a familiarity level knowledge of basic heating, ventilation, air conditioning (HVAC) system and filtration system construction, operation, and application.**

   a. **Given engineering diagrams of an HVAC system, identify the following components and discuss their purposes:**
      - **Blowers**
      - **Fans**
      - **Dampers**
      - **Chillers**
      - **Filters**
      - **HEPA filters**
      - **Heat exchangers**
      - **Scrubbers**
      - **Hoods**
      - **Gloveboxes**
      - **Flow, pressure, temperature, current, level, voltage and position indicators, recorders, and controllers**

Blowers and Fans

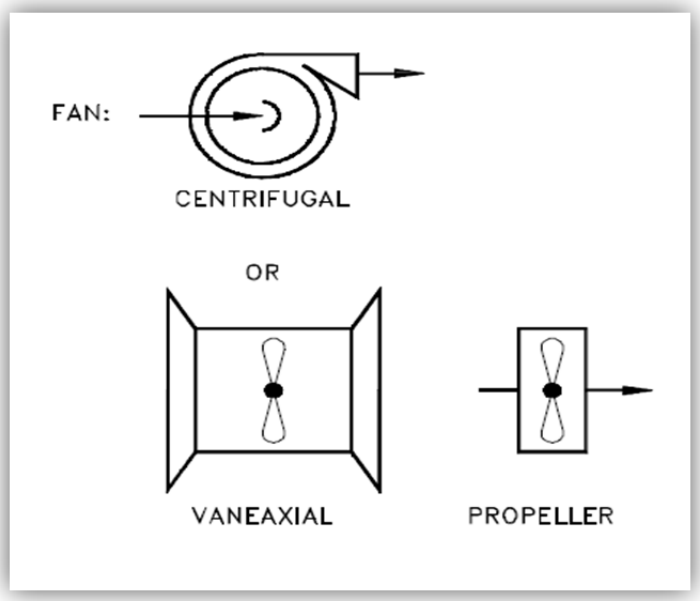
The following is taken from GlobalSpec, *HVAC Fans and Blowers Information*.



HVAC fans and blowers are used to move air through heating, ventilating, and air conditioning systems. They are an integral part of the air handling systems that bring or draw fresh air into buildings.

Source: IPS-E-PR-170

Figure 20. Blower



Source: DOE-HDBK-1016/1-93

Figure 21. Fans

Dampers

The following is taken from eHow, *How Does a HVAC Damper Work?*

A damper is an adjustable plate for controlling the draft in an HVAC system. Dampers open and close based on the portion of the HVAC system in use, and they control airflow and venting of the system.

Source: ANSI/ASHRAE Standard 134-2005

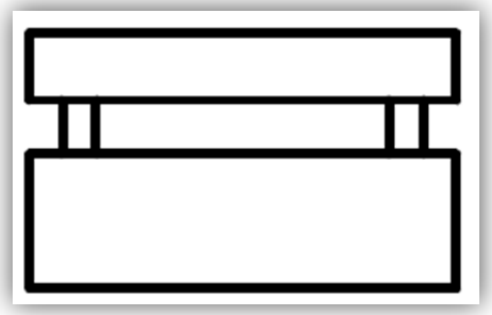
Figure 22. Dampers

Chiller

The following is taken from Wikipedia, *Glossary of HVAC Terms*.

A chiller is a device that removes heat from a liquid via a vapor-compression or absorption refrigeration cycle. Chillers are of two types: air-cooled or water-cooled. Air-cooled chillers are usually outside and consist of condenser coils cooled by fan-driven air. Water-cooled

chillers are usually inside a building, and heat from these chillers is carried by recirculating water to a heat sink such as an outdoor cooling tower.



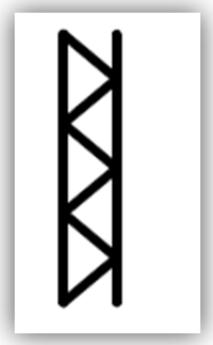
Source: ANSI/ASHRAE Standard 134-2005

Figure 23. Chiller

Filters

The following is taken from Environmental Protection Agency, *Indoor Air Quality in Large Buildings*, appendix B, "HVAC Systems and Indoor Air Quality."

Filters are primarily used to remove particles from the air. The type and design of filters determine the efficiency at removing particles of a given size and the amount of energy needed to pull or push air through the filter.



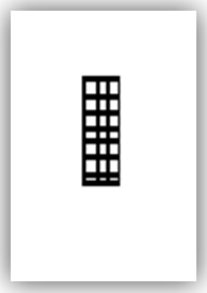
Source: ANSI/ASHRAE Standard 134-2005

Figure 24. Filter

HEPA Filter

The following is taken from Wikipedia, *HEPA*.

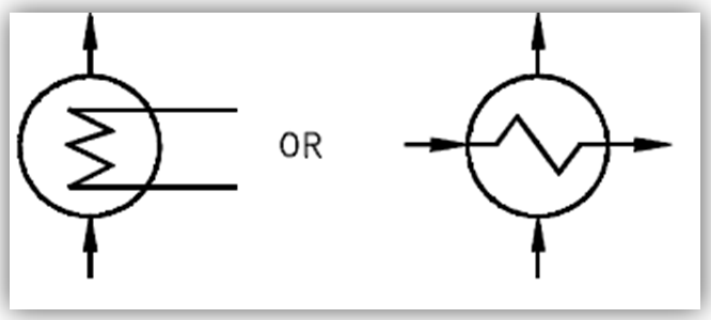
High-efficiency particulate absorption (HEPA) is a type of air filter. Filters meeting the HEPA standard have many applications, including use in medical facilities, automobiles, aircraft, and homes. The filter must satisfy certain standards of efficiency such as those set by DOE. To qualify as HEPA by government standards, an air filter must remove 99.97 percent of particles that have a size of 0.3 µm or larger. A filter that is qualified as HEPA is subject to interior classifications.



Source: DOE-HDBK-1013/1-93
Figure 25. HEPA filter

Heat Exchanger
The following is taken from Wikipedia, *Heat Exchanger*.

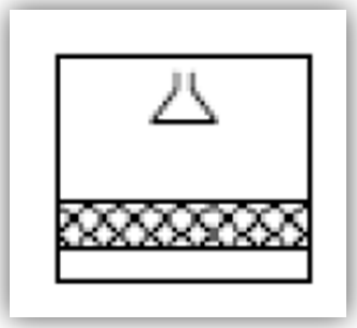
A heat exchanger is a piece of equipment built for efficient heat transfer from one medium to another. They are widely used in space heating, refrigeration, air conditioning, power plants, chemical plants, petrochemical plants, petroleum refineries, natural gas processing, and sewage treatment.



Source: DOE-HDBK-1013/1-92
Figure 26. Heat exchangers

Scrubbers
The following is taken from GlobalSpec, *Air Scrubbers, Wet Scrubbers, and Gas Scrubbers Information*.

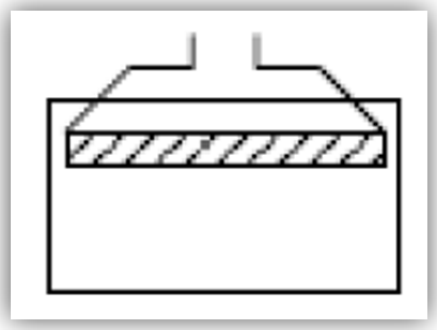
Scrubbers are air pollution control devices that use liquid to remove particulate matter (PM) or gases from an industrial exhaust or flue gas stream. Scrubbers use a liquid (often water) to capture and remove the pollutants.



Source: DOE-HDBK-1013/1-92
Figure 27. Scrubber

Hoods
The following is taken from Wikipedia, *Fume Hood*.

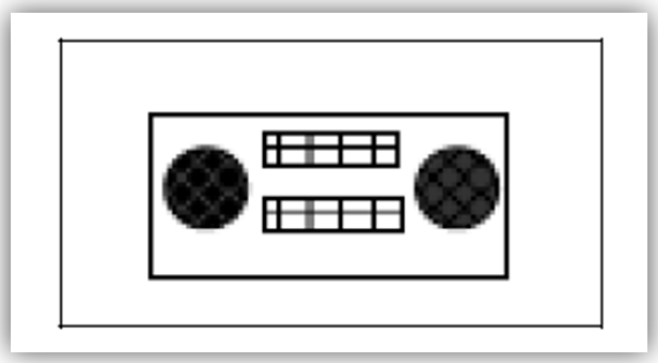
A fume hood is a type of local ventilation device that is designed to limit exposure to hazardous or toxic fumes, vapors, or dusts. A fume hood is typically a large piece of equipment enclosing five sides of a work area, the bottom of which is most commonly located at a standing work height.



Source: DOE-HDBK-1013/1-92
Figure 28. Hood

Gloveboxes
The following is taken from Laboratory Network.com, *Determining the Right Glovebox for Accurate Level of Containment*.

A glovebox is a sealed enclosure that allows materials handling through long, relatively impermeable gloves secured to ports in the walls of the enclosure. The purpose is the protection or isolation that is provided by the physical barrier.



Source: DOE-HDBK-1013/1-92

Figure 29. Glovebox

Flow, Pressure, Temperature, Current, Level, Voltage and Position Indicators, Recorders, and Controllers

FLOW

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

Flow measurement is an important process measurement to be considered in operating a facility's fluid systems. For efficient and economic operation of these fluid systems, flow measurement is necessary. The differential pressure (DP) that is caused by the head is measured and converted to a flow measurement.

PRESSURE

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

Pressure measurement is a necessary function in the safe and efficient operation of DOE nuclear facilities. All pressure detectors are used to provide up to three basic functions: indication, alarm, and control. Since the fluid system may operate at saturation and subcooled conditions, accurate pressure indication must be available to maintain proper cooling.

TEMPERATURE

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

The hotness or coldness of a piece of plastic, wood, metal, or other material depends on the molecular activity of the material. Kinetic energy is a measure of the activity of the atoms that make up the molecules of any material. Therefore, temperature is a measure of the kinetic energy of the material in question. The resistance of certain medals will change as temperature changes.

CURRENT

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

Thermocouples (TCs) 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.

LEVEL

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

Liquid level measuring devices are classified in two groups: direct method, and inferred method. An example of the direct method is the dipstick in a car that measures the height of the oil in the oil pan. An example of the inferred method is a pressure gauge at the bottom of a tank that measures the hydrostatic head pressure from the height of the liquid.

VOLTAGE AND POSITION INDICATORS

Recorders

The following is taken from Wikipedia, *Data Logger*.

The data logger or data recorder, is an electronic device that records data over time or in relation to location either with a built-in instrument or sensor or via external instruments and sensors. They generally are small, battery-powered, portable, and equipped with a microprocessor, internal memory for data storage, and sensors.

Controllers

The following is taken from Wikipedia, *HVAC Control System*.

Central controllers and most terminal unit controllers are programmable, meaning the direct digital control program code may be customized for the intended use. Groups of controllers, networked or not, form a layer of the system by themselves. This subsystem is vital to the performance and basic operation of the overall HVAC system; it is the brain of the HVAC system. It dictates the position of every damper and valve in the system. It determines which fans, pumps, and chillers run and at what speed or capacity.

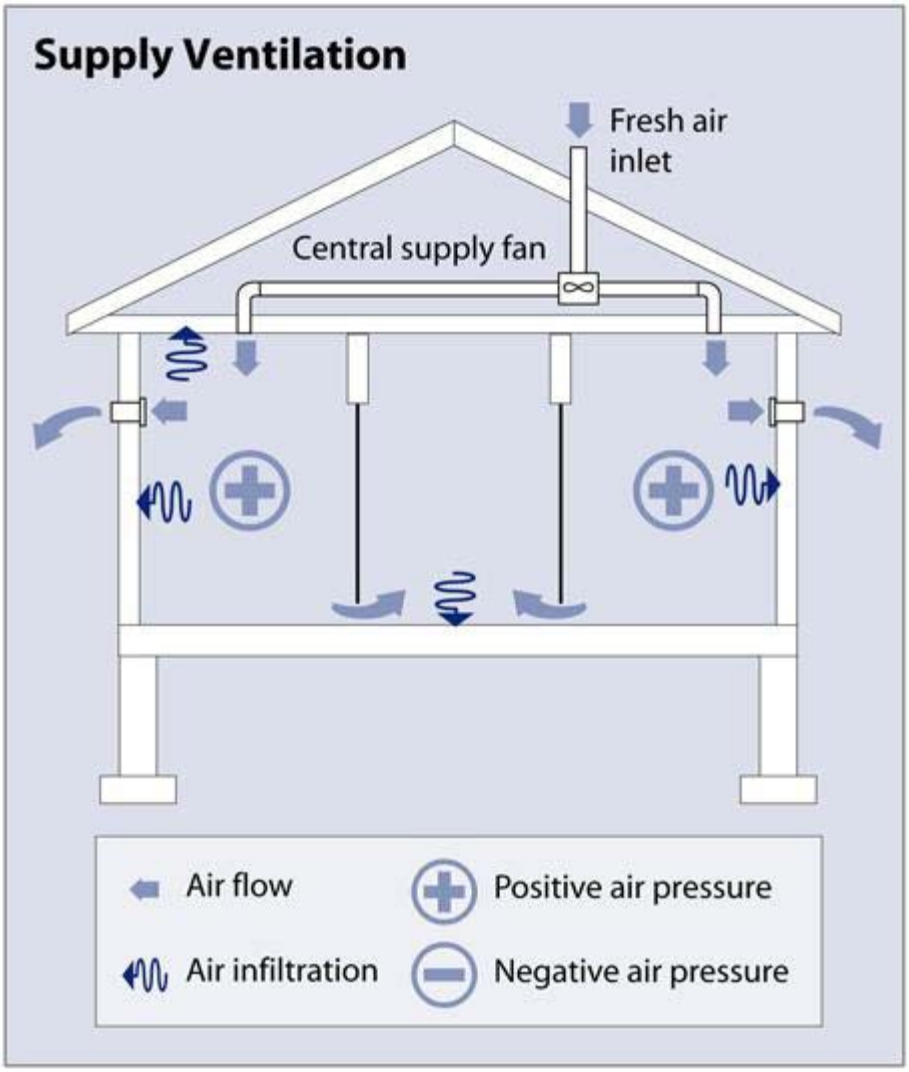
b. Discuss the relationships between the following in HVAC systems:

- **Supply ventilation**
- **Flow**
- **Exhaust ventilation**

Supply Ventilation

The following is taken from Cambridge Energy Alliance, *Whole House Ventilation Systems*.

Supply ventilation systems work by pressurizing the building. They use a fan to force air into the building while air leaks out through cracks and strategically placed vents. They discourage the entry of pollutants and prevent back-drafting of combustion gases. They allow outside air to be filtered before entering the house. However, they do not allow outside air to be conditioned or dehumidified before entering and do not work well in cold climates. They could contribute to higher cooling costs. They are recommended for warm climates.



Source: Cambridge Energy Alliance, Whole House Ventilation Systems

Figure 30. Supply ventilation

Flow

The following is taken from *Energy Consumption Characteristics of Commercial Building HVAC Systems*, volume II: "Thermal Distribution, Auxiliary Equipment, and Ventilation."

Airflow to the conditioned space may be controlled, as in the case of a variable air volume system, with a terminal valve box. The air is finally delivered to the space through a diffuser, whose purpose is to mix the supply air and the room air. The terminal box may or may not have a reheat coil that provides additional heat when the space does not need to be cooled or needs less cooling than would be delivered by supply air at the terminal box's minimum air quantity setting. Constant air volume systems that are not allowed by energy codes in many applications do not reduce air delivery rates and are dependent on reheat coils to control the delivered cooling.

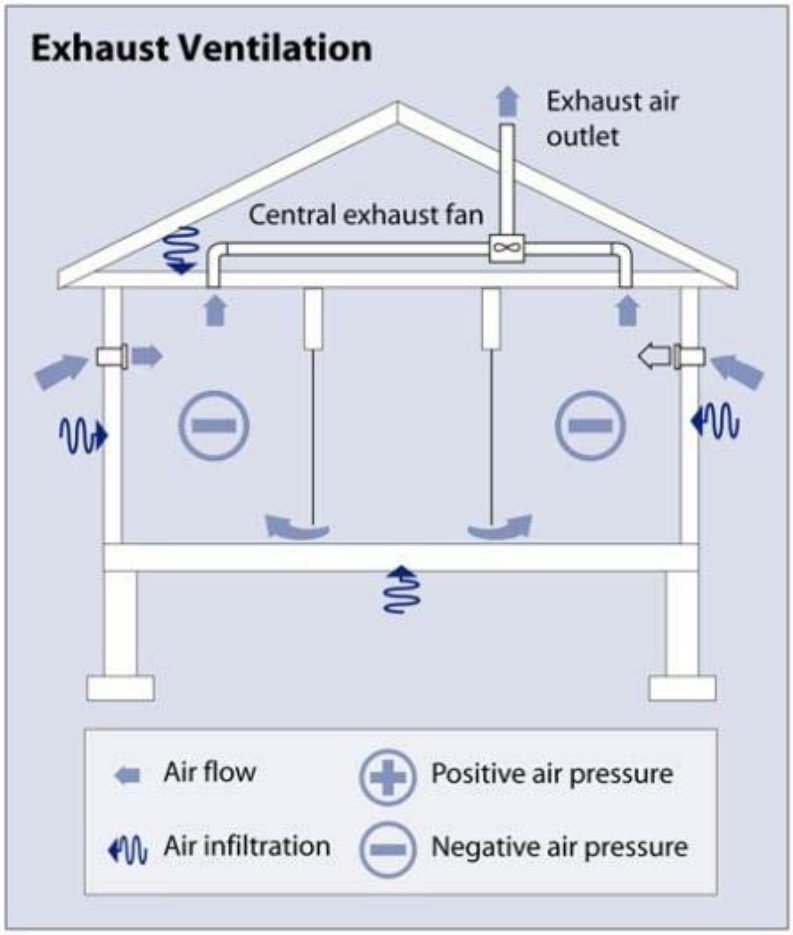
Air leaves the conditioned space either through the return system, or through the exhaust system. In many installations, the ceiling plenum space is used as part of the return ducting to save the cost of return ductwork.

Video 36. Airflow and ventilation in a condo
http://www.youtube.com/watch?v=6392z75_4iU

Exhaust Ventilation

The following is taken from Cambridge Energy Alliance, *Whole House Ventilation Systems*.

Exhaust ventilation systems work by depressurizing the building; they reduce the indoor air pressure to below the outdoor air pressure and extract indoor air from inside a building while fresh air infiltrates through the cracks and strategically placed passive vents. Some concerns with the exhaust ventilation system are that they might draw in pollutants if the infiltration is not controlled and it might lead to higher heating and cooling costs, since the incoming air is not conditioned. The exhaust ventilation system is recommended for cold climates.



Source: Cambridge Energy Alliance, Whole House Ventilation Systems

Figure 31. Exhaust ventilation

Video 37. Supply versus exhaust ventilation
http://www.youtube.com/watch?v=6N8n8sv3mY8

c. Describe the purpose of the HVAC system in the following applications:

- **Hoods**
- **Gloveboxes**
- **Hot cells**
- **Confinement systems**
- **HEPA filtration**

Hoods

The following is taken from the ESCO, *A Guide to Laboratory Fume Hoods*.

Laboratory fume hoods are a type of ventilation system with the primary function to exhaust chemical fumes, vapors, gases, dust, mist, and aerosol. Fume hoods serve as physical barriers between reactions and the laboratory, offering a measure of protection against inhalation exposure, chemical spills, run-away reactions, and fires.

The hood functions by maintaining a relatively negative pressure in the interior to prevent any contaminant from escaping while drawing air in through the hood opening at a consistent rate. A suitable hood face velocity (the speed at which air is drawn into the hood) is of importance to the safe and effective operation of a fume hood. While excessive face velocities can often result in turbulence and reduce containment, insufficient velocities can compromise hood performance.

Gloveboxes

The following is taken from Savannah River Site, *DNFSB Recommendation 2004-2, Ventilation System Evaluation*.

The primary function of the active confinement ventilation system is to minimize the spread of potentially radioactive airborne contaminants, maintain personnel radiation exposure ALARA, provide dilution airflow to prevent flammable gas buildup, and prevent the release of radioactive contaminants to the environment. This system provides confinement by maintaining an airflow gradient that moves air from areas of less contamination potential to areas of higher contaminant potential before being exhausted through HEPA filers and the exhaust stack.

Hot Cells

The following is taken from DOE-HDBK-1132-99.

Hot cells should meet the following criteria:

- Space and equipment should be provided as needed to support accountability, process monitoring, and material control requirements.
- Exhaust prefilters and HEPA filters should be installed to facilitate filter replacement and repair.
- Standby filters should be incorporated for backup protection during filter changes so that filters can be changed without shutting down the exhaust fans. Standby filters should be installed outside the cell and sealed in an acceptable enclosure for direct maintenance.
- Exhaust systems should have alarms that will annunciate if the concentration of radioactive material in the exhaust exceeds the limits specified in the facility TSR.

In facilities where plutonium or enriched uranium is processed, the following are additional considerations:

- Wherever possible, the designer should provide enclosures for confining process work on plutonium and enriched uranium. When these confinement enclosures are specified and designed, consideration should be given to whether room ventilation air for either a secondary or tertiary confinement can be recirculated. If a recirculation ventilation system is provided, the design should provide a suitable means for switching from recirculation to once-through ventilation.
- If advantageous to operations, maintenance, or emergency personnel, the ventilation system should provide for independent shutdown. Such a shutdown should be considered in light of its effect on the airflow in other interfacing ventilation systems. When a system is shut down, positive means of controlling backflow of air to uncontaminated spaces should be provided by positive shutoff dampers, blind flanges, or other devices.
- Equipment to continuously monitor oxygen levels should be provided for occupied working areas of facilities equipped with significant quantities of inert or oxygen-deficient process glovebox lines. Allowable leakage rates for ductwork systems should be taken into consideration.
- The supply of air to primary confinement, such as enclosures that confine the processing of plutonium and enriched uranium, should be filtered by HEPA filters at the ventilation inlets to the enclosures and area confinement barriers to prevent the transport of radioactive contamination in the event of a flow reversal.
- If room air is recirculated, the recirculation circuit should provide at least one stage of HEPA filtration. The design should include redundant filter banks and fans. If recirculation systems are used, contaminated process enclosure air should be prevented from exhausting into the working area rooms. Process enclosure air (from hoods, gloveboxes, etc.) should be treated and exhausted without any potential for recirculation to occupied areas.
- The designer should specify and locate components in the exhaust systems to remove radioactive materials and noxious chemicals before the air is discharged to the environment. These components should be capable of handling combustion products safely. Exhaust system design should safely direct effluents through the appropriate ventilation ducts and prevent spread beyond the physical boundary of the ventilation system until treated.
- HEPA filters should be installed at the interface ventilation system to minimize the contamination of exhaust ductwork. Prefilters should be installed ahead of HEPA filters to reduce HEPA filter loading. The filtration system should be designed to allow reliable in-place testing of the HEPA filter and to simplify filter replacement.
- Separate exhaust ventilation system ductwork and the initial two stages of filtration should be designed for exhaust air from enclosures that confine the process. These systems should maintain a negative pressure inside the enclosure with respect to the operating area. These systems should be designed to remove moisture, heat, explosive and corrosive gases, and other contaminants. These systems should be designed to automatically provide adequate inflow of air through a credible breach in the enclosure confinement.

- Enclosures that confine the process and are supplied with gases at positive pressure should have positive-acting pressure-relief valves that relieve the exhaust system to prevent over-pressurization of the process confinement system.

The following is taken from the International Thermonuclear Experimental Reactor, *G A0 General Design Requirements Document 3 01-17-19 R1.0*, chapter 21, "Design Requirements and Guidelines Level 2."

The hot cell building HVAC systems will provide air quality (temperature, humidity, purity, freshness) sufficient to meet the requirements of the personnel and equipment located in the building. HVAC systems provide heat removal for the equipment and lighting. The hot cell building HVAC systems shall be designed as independent subsystems for each of the major process areas, to limit the possibility of transfer of contamination between areas.

Confinement Systems

The following is taken from Lawrence Livermore National Laboratory, UCRL-AR-133352 Rev 1, *HVAC Systems in Nonreactor Nuclear Facilities*.

The design of a confinement ventilation system shall ensure the ability to maintain desired airflow characteristics when personnel access doors or hatches are open. When necessary, air locks or enclosed vestibules shall be used to minimize the impact of this on the ventilation system and to prevent the spread of airborne contamination within the facility. The ventilation system design shall provide the required confinement capability under all credible circumstances including a single-point failure in the system.

Air clean-up systems shall be provided in confinement ventilation exhaust systems to limit the release of radioactive or other hazardous material to the environment and to minimize the spread of contamination within the facility as determined by the safety analysis. The following general clean-up system requirements shall be met, as appropriate, for ventilation system design:

- The level of radioactive material in confinement exhaust systems shall be continuously monitored. Alarms shall be provided that will annunciate in the event that activity levels above specified limits are detected in the exhaust stream. Appropriate manuals or automatic protective features that prevent an uncontrolled release of radioactive material to the environment or workplace shall be provided.
- To limit onsite doses and reduce offsite doses by enhancing atmospheric dispersion, elevated confinement exhaust discharge locations are required. The height of the exhaust discharge location shall ensure that the calculated consequences of normal or accidental releases shall not exceed the radiological guidance contained in the applicable SAR. In addition, to the extent practical, all normal and accidental releases shall be maintained at ALARA levels.
- An elevated stack shall be used for confinement exhaust discharge. Provisions shall be made to ensure an adequate ventilation exhaust discharge path in the event of stack failure.

HEPA Filtration

The following is taken from Wikipedia, *Air Purifier*.

An air purifier is a device that removes contaminants from the air. Several different processes of varying effectiveness can be used to purify air, one of which is the HEPA filter. Air is forced through a filter and particles are physically captured by the filter.

HEPA filters remove at most 99.97 percent of 0.3 micrometer particles, and are usually more effective for particles that are larger. Filters for HVAC at minimum efficiency reporting value (MERV) 14 or above are rated to remove airborne particles of 0.3 micrometers or larger. A high-efficiency MERV 14 filter has a capture rate of at least 75 percent for particles between 0.3 to 1.0 micrometers.

d. Discuss the reason for, and safety significance of, the following system parameters:

- **Positive versus negative system pressure**
- **Differential pressure across filters**
- **Differential pressure across components**
- **Adequacy of flow across filters versus differential pressure**

Positive Versus Negative System Pressure

The following is taken from the Environmental Protection Agency, *Control of Gaseous* Emissions, chapter *3*: "Air Pollution Control Systems."

When fans used in industrial systems create gas static pressures above the prevailing atmospheric pressure, the condition is termed positive pressure. When the fans create a gas static pressure below the prevailing atmospheric pressure, negative pressure exists. Positive and negative pressures are considered relative terms since the static pressure is being described in a form that is compared to the atmospheric pressure. The atmospheric pressure is an absolute term since it is directly related to the number of molecules and their kinetic energy.

$$SP\ (absolute) = SP\ (barometric) + SP\ (relative)$$

Video 38. HVAC negative and positive pressure, part 2
http://www.youtube.com/watch?v=4MdK-hjCdyQ

Differential Pressure Across Filters

The following is taken from Controlled Environments, *Filtration Monitoring Lags Behind Advances in HVAC Technology*.

The common means of monitoring air filters continues to be by measuring the DP across the filter bank.

Looking at the dynamics of air filtration, we are really talking about the changing airflow as a filter progresses from a clean to an obstructed state. Air filters are defined by their relative airflow. When we use DP as the means to monitor air filters, we introduce several problems. Pressure across the filter bank is influenced by environmental factors such as changes in atmospheric temperature. Systems using variable speed fans change the relative pressure

according to the demand of the building. When monitoring air filters with DP, practitioners must be aware that relatively small changes in air pressure result in large changes in airflow. HVAC systems can appear to have adequate airflow as measured by pressure, yet the system has reached a point where a small change in pressure can result in loss of flow and equipment failure.

The commonly accepted industry "rule of thumb" is that a filter is ready for replacement once it has reached twice the initial pressure drop across the filter bank.

Differential Pressure Across Components

The following is taken from the OneMine.org, National Institute for Occupational Safety and Health, *A Comparative Evaluation of the Differential-Pressure-Based Respirable Dust Dosimeter with the Personal Gravimetric Respirable Dust Sampler in Underground Coal Mines*.

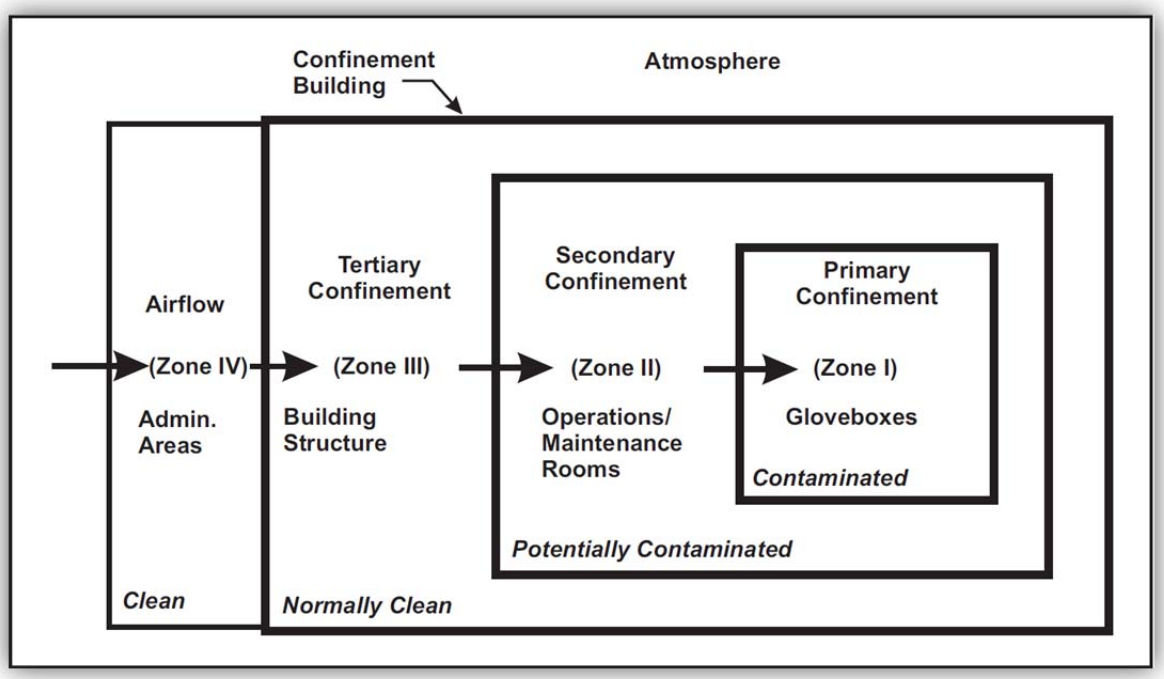
As DP across a filter or a component increases, it indicates an increase in the amount of material contained in the filters or that settles on the component. For radioactive systems, this may lead to increased radiation levels above what is prescribed in the DSA.

Video 39. Combustion analyzer training, part 8
http://www.youtube.com/watch?v=Ojz3I5Vp5iA

Adequacy of Flow Across Filters Versus Differential Pressure

The following is taken from DOE-HDBK-1169-2003.

As shown in figure 32, the general approach to ventilation is a three-tiered relationship. Multi-zoned buildings are usually ventilated to achieve airflows from the less contaminated zone to the more contaminated zone. Thus, airflow must be sufficient to provide the necessary degree of contaminant dilution and cooling, and to maintain sufficient pressure differentials between zones where there can be no backflow of air spaces of lower contamination, even under abnormal conditions.



Source: DOE-HDBK-1169-2003.

Figure 32. Typical process facility confinement zones

PRIMARY CONFINEMENT ZONE

The following is taken from DOE-HDBK-1169-2003.

The primary confinement zone comprises those areas where high levels of airborne contamination are anticipated during normal operations. Facility personnel do not normally enter primary confinement zones. When entry is necessary, it is done under tightly controlled conditions. This zone includes the interior of a hot cell, glovebox, piping, vessels, tanks, exhaust ductwork, primary confinement HEPA filter plenums, or other confinement for handling highly radiotoxic material. Confinement features must prevent the spread of radioactive material within the building under normal operating and upset conditions up to and including the DSA for the facility. Complete isolation (physical separation) from neighboring facilities, laboratories, shop areas, and operating areas is necessary. Unavoidable breaches in the primary confinement barrier must be sized to ensure an adequate inflow of air or safe collection of the spilled material. The exhaust system must be sized to ensure an adequate inflow of air in the event of a credible confinement breach. An air exhaust system that is independent of those serving surrounding areas is required. High-efficiency filters, preferably HEPA type, are typically required in air inlets, and two independently testable stages of HEPA filters are required in the exhaust. The exact number of testable stages is determined by safety analysis.

SECONDARY CONFINEMENT ZONE
The following is taken from DOE-HDBK-1169-2003.

The secondary confinement zone comprises those areas where airborne contamination could be generated during normal operations or as a result of a breach of a primary confinement barrier. This zone consists of the walls, floors, ceilings, and associated ventilation systems that confine any potential release of hazardous materials from primary confinement. Related areas include glovebox operating areas, hot cell service or maintenance areas, and the ventilation system servicing the operating areas. Pressure differentials must be available to produce inward airflow into the primary confinement should a breach occur. Penetrations of the secondary confinement barrier typically require positive seals to prevent migration of contamination out of the secondary confinement zone. Air locks or a personnel clothing-change facility are recommended at the entrance to the zone. Restricted access areas are generally included in the secondary confinement zone.

TERTIARY CONFINEMENT ZONE
The following is taken from DOE-HDBK-1169-2003.

The tertiary confinement zone comprises those areas where airborne contamination is not expected during normal facility operations. This zone consists of the walls, floors, ceilings, and associated exhaust system of the process facility. It is the final barrier against release of hazardous material to the environment. This level of confinement should never become contaminated under normal operating conditions. The secondary and tertiary boundaries may exist in common, as in a single-structure envelope.

e. Discuss the failure modes and potential hazards (to equipment and personnel) associated with the use of HVAC systems and components within nuclear safety-related systems.

Failure Modes
The following is taken from DOE-HDBK-1132-99.

Premature component failure and poor performance are often the result of materials being selected without an adequate evaluation of all the failure modes applicable to a component and the materials of construction. Mechanical design codes have been excellent in preventing failure due to mechanical design. Component failures due to rupture, overload, or cyclic loading have steadily decreased and in some applications have been totally designed out of the component. The majority of the failures and poor component performance that occur are due to age-related degradation. The effects of thermal aging, radiation, corrosion, erosion, cyclic fatigue, and instability of the material should be taken into account. Cable products should be manufactured in accordance with insulated cable engineering association requirements.

Potential Hazards
The following is taken from DOE-HDBK-1169-2003, chapter 11.

The HVAC systems of a building where radioactive materials are handled or processed are integral parts of the building's confinement. In some cases, these systems may be shut down

in the event of an operational upset, power outage, accident, fire, or other emergency. In other cases, they must remain operational to maintain the airflows and pressure differentials between building spaces and between the building and the atmosphere required to maintain confinement. In some of these cases, airborne radioactive material may not be a problem until an emergency occurs. In all cases, however, a particular danger is damage to or failure of the final HEPA filters (and absorbers in those facilities where radiolytic particulates could be released) that constitute the final barrier between the contained space (hot cell, glovebox, room, or building) and the atmosphere or adjacent building spaces. Even if the system can be shut down in the event of an emergency, protection of the final filters is essential to prevent the escape of contaminated air to the atmosphere or to allow personnel to occupy spaces of the building.

Considerations must be given to the following:

- The possible effects of operational upsets, power outages, accidents, fires, and other emergencies on the HVAC systems, including damage to filters and adsorbers from shock, overpressure, heat, fire, and high sensible-moisture loading
- The design and arrangement of ducts and air cleaning components to alleviate these conditions
- The means of switching to a redundant air-cleaning unit, fan, or alternate power supply
- The methods of controlling or isolating the exhaust system during failure conditions

To provide the necessary protection to the public and plant personnel, the HVAC system components that confinement leakage controls depends on must remain essentially intact and serviceable under these upset conditions. These components must be capable of withstanding the DP, heat, moisture, and stress of the most serious accident predicted for the facility, with minimum damage and loss of integrity, and they must remain operable long enough to satisfy system objectives.

The following is taken from the Illinois Department of Public, Health Environmental Health Fact Sheet, *Air Quality in the Home*.

Poor indoor air quality (IAQ) is generally caused by the lack of adequate ventilation that results in a buildup of contaminants from sources in the home. The National Institute for Occupational Safety and Health identified inadequate ventilation as the primary problem in more than half the workplace indoor air investigations it has conducted. These investigations revealed that proper ventilation is important in maintaining good IAQ.

Most residential HVAC systems recirculate indoor air to conserve energy. The current trend in home construction is to reduce air leakage through cracks and other openings in walls, floors, and roofs. The combination of these construction practices and the recirculation of indoor air have led to an increase in indoor air problems. The American Society of Heating, Refrigeration, and Air Conditioning Engineers (ASHRAE) recommends that a minimum of 15 cubic feet of outdoor make-up air per minute per person be brought into the indoor living areas of residential buildings. The ASHRAE guidelines for make-up air for kitchens and bathrooms in homes are much higher. This fresh make-up air dilutes indoor contaminants and helps exhaust them from the home.

Inadequate maintenance of the HVAC system may contribute to IAQ problems. Ventilation system filters that are not replaced on a regular basis may become clogged, reducing airflow volume, quality, and distribution. They may become places for bacteria and molds to grow and be distributed throughout the home. Humidifier systems on HVAC systems must be properly maintained to prevent bacteria and mold contamination.

A bioaerosol is an airborne product of a biological contaminant. Biological contaminants in the home may include mold, bacteria, viruses, mites, and pollen. Bioaerosols released into the air can be distributed throughout the home by the HVAC system. Bioaerosols can cause infectious diseases such as Legionnaires' disease. They can cause allergic reactions, including hypersensitivity pneumonitis, allergic rhinitis, and some types of asthma.

Factors that can contribute to the growth of biological contaminants are wet or moist building materials and poorly maintained humidifiers, dehumidifiers, and air conditioners. "Humidifier fever" is a common illness caused by improper maintenance of humidifiers that can produce fever, chills, headaches, and persistent coughs. Various fungi and bacteria grow quickly in the warm, stagnant water inside humidifiers. During operation, these contaminants can be distributed in the air and inhaled by the building occupants.

The following is taken from Cleaningpro.com, *Cleaning Products From A to Z: M*.

HVAC systems accumulate, and become a breeding ground for, a variety of contaminants that have adverse health effects. Most people spend more than half their time inside, and yet the EPA has found that indoor air can be 70 percent more polluted than outdoor air. In fact, one out of six people suffering from allergies does so as a direct result of the fungi and bacteria in their HVAC system and air duct system.

Today, IAQ has become a major health concern. Indoor air pollution can lead to the serious problem of "sick building syndrome" that result in a wide range of health concerns, including respiratory disease, dizziness, eye irritation, and general fatigue. Contamination of air associated with bacteria, fungi, and mold is believed to be the primary cause of the IAQ problems. According to the EPA and the U.S. Department of Health and Human Services, "contaminated central air handling systems can become breeding grounds for mold, mildew, and other sources of biological contaminants and can then distribute these contaminants," and "the HVAC systems must be well-maintained, inspected, and cleaned on a prescheduled and periodic basis and repaired as needed. If maintenance is inadequate, problems that may arise again include growth and dissemination of microbial agents."

HVAC system surfaces have always been point sources for microbial contamination that may degrade the indoor environment's IAQ and the interiors of these systems provide an especially favorable environment where microbial growth can expand uncontrollably into colony-forming units. Cooling coils, drain pans, and the fan housing assemblies are the most contaminant-prone components of an air-handling unit. When mold colonies grow, they produce a sticky substance called enzyme mycelium that traps dust particles from the air and breaks down the trapped particles into food. By preventing the growth and establishment of colonies, the various effects caused by biological contamination in the air handler can effectively be controlled.

All air delivered to the interior of a mechanically ventilated building passes through an air handler. Investigative reports of indoor environmental problems have long noted that air handlers in problem buildings are almost always contaminated, often with evidence of extensive microbial growth. In most systems, air is recirculated many times rather than being exhausted after one pass through the air handler.

Air that passes through a contaminated air handler will pick up loose particles from the surfaces within the air handler. The more times the same volume of air passes over a contaminated surface, the greater the possibility that it will pick up contamination. The tight design of modern buildings results in reduced exchange of air with the outdoors. Thus, the concentration of contaminated air will increase as long as air is allowed to pass over contaminated surfaces.

In light of these facts, a fundamental requirement for achieving and maintaining an acceptable level of interior air quality is the cleanest possible interior for all air handlers in the facility. Air handlers must be maintained in a clean condition and not be allowed to become contaminated. Maintaining an acceptable clean air handler requires three steps: 1) cleaning an air handler that has become contaminated; 2) preventing the air handler from becoming contaminated again; and 3) identifying when contamination has begun to reoccur.

8. Nuclear safety specialists must demonstrate a familiarity level knowledge of process instrumentation principles of operation as applied to nuclear safety-related systems.

a. Explain the process-related reasons for measuring temperature, pressure, flow, and fluid level.

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

Temperature

The hotness or coldness of a piece of plastic, wood, metal, or other material depends on the molecular activity of the material. Kinetic energy is a measure of the activity of the atoms that make up the molecules of any material. Therefore, temperature is a measure of the kinetic energy of the material in question.

Most temperature measuring devices use the energy of the material or system they are monitoring to raise, or lower, the kinetic energy of the device. A normal household thermometer is one example. The mercury, or other liquid, in the bulb of the thermometer expands as its kinetic energy is raised. The temperature of the measured object can be told by observing how far the liquid rises in the tube.

Pressure

Although the pressures that are monitored vary slightly depending on the details of facility design, all pressure detectors are used to provide up to three basic functions: indication, alarm, and control. Since the fluid system may operate at saturation and subcooled conditions, accurate pressure indication must be available to maintain proper cooling. Some pressure detectors have audible and visual alarms associated with them when specified preset limits are exceeded. Some pressure detector applications are used as inputs to protective features and control functions.

Flow
Flow measurement is an important process measurement to be considered in operating a facility's fluid systems. For efficient and economic operation of these fluid systems, flow measurement is necessary.

Fluid Level
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.

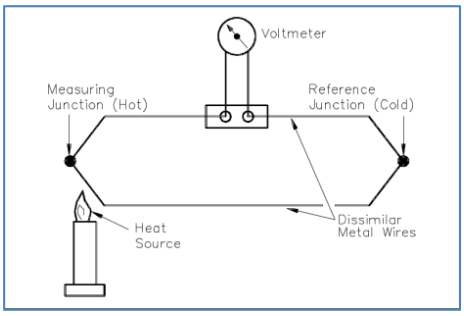
There are three major reasons for utilizing remote level indication:

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.

b. For the temperature detection devices listed, explain how the instrument provides an output representative of the temperature being measured:
- **Thermocouple (TC)**
- **Resistance temperature detector (RTD)**

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

Thermocouple (TC)
TCs 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. Heating the measuring junction of the TC produces a voltage that is greater than the voltage across the reference junction. The difference between the two voltages is proportional to the difference in temperature and can be measured on the voltmeter (in millivolts). For ease of operator use, some voltmeters are set up to read out directly in temperature through use of electronic circuitry. Other applications provide only the millivolt readout. To convert the millivolt reading to its corresponding temperature, the applicable conversion chart supplied by the manufacturer needs to be used.



Source: DOE-HDBK-1013/1-92

Figure 33. Thermocouple

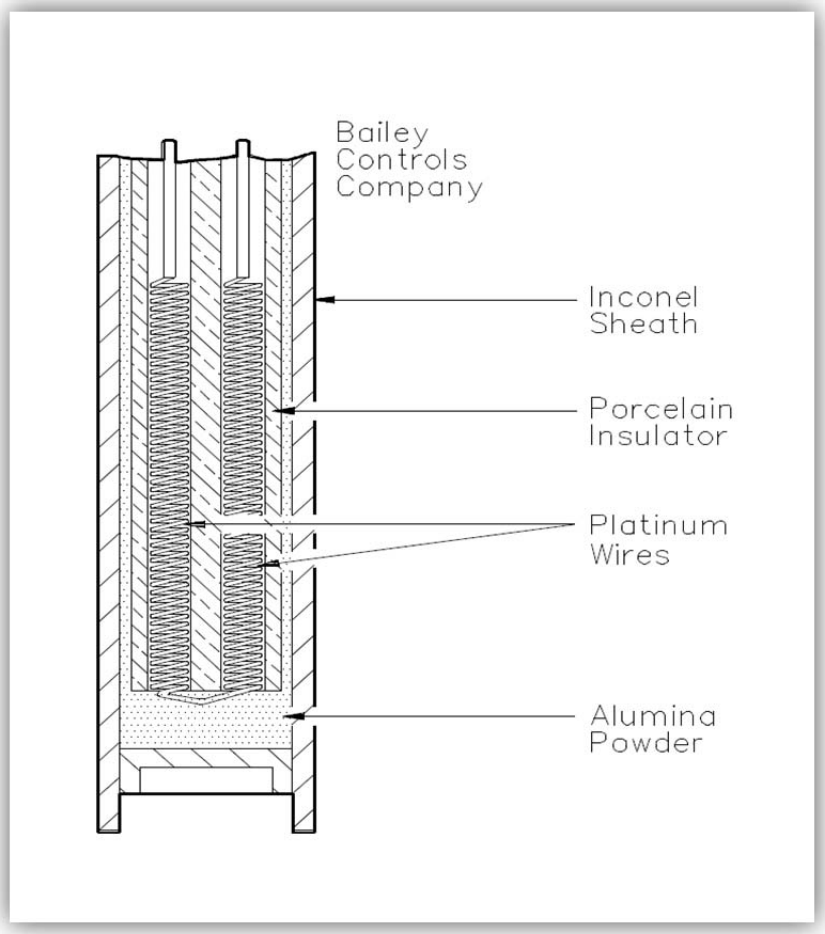
Video 40. Thermocouple tutorial

http://www.youtube.com/watch?v=AubXDi6AD2M

Resistance Temperature Detector (RTD)
The RTD incorporates pure metals or certain alloys that increase in resistance as temperature increases and, conversely, decrease in resistance as temperature decreases. RTDs act somewhat like an electrical transducer, converting changes in temperature to voltage signals

by the measurement of resistance. The metals that are best suited for use as RTD sensors are pure, of uniform quality, stable within a given range of temperature, and able to give reproducible resistance-temperature readings. Only a few metals have the properties necessary to use in RTD elements.

RTD elements are normally constructed of platinum, copper, or nickel. These metals are best suited for RTD applications because of their linear resistance-temperature characteristics, their high coefficient of resistance, and their ability to withstand repeated temperature cycles. The coefficient of resistance is the change in resistance per degree change in temperature, usually expressed as a percentage per degree of temperature. The material used must be capable of being drawn into fine wire so that the element can be easily constructed.



Source: DOE-HDBK-1013/1-92

Figure 33. RTD

RTD elements are usually long, spring-like wires surrounded by an insulator and enclosed in a sheath of metal. One design has a platinum element that is surrounded by a porcelain insulator. The insulator prevents a short circuit between the wire and the metal sheath. Inconel, a nickel-iron-chromium alloy, is normally used in manufacturing the RTD sheath because of its inherent corrosion resistance. When placed in a liquid or gas medium, the Inconel sheath quickly reaches the temperature of the medium. The change in temperature

will cause the platinum wire to heat or cool, resulting in a proportional change in resistance. This change in resistance is then measured by a precision resistance measuring device that is calibrated to give the proper temperature reading.

Video 41. RTD simulation using resister network
http://www.youtube.com/watch?v=BWPW7yB9bGY

c. For the pressure detection devices listed, explain how the instrument provides an output representative of the pressure being measured:
- **Magnehelic differential pressure device**
- **Photohelic differential pressure device**

Magnehelic Differential Pressure Device

The following is taken from ncidavid.blogspot.com, *Magnehelic Basics*.

The magnehelic is a test instrument that has been around the HVAC industry for decades. It is still the test instrument of choice when measuring the static pressure of HVAC systems. This instrument can handle daily field use by a technician and is cost-effective. The magnehelic gauge has a very sensitive diaphragm that responds to changes in pressure. When pressure is exerted on the diaphragm, the dial of the magnehelic responds accordingly and moves to the appropriate pressure. For this diaphragm to function correctly, positioning of the instrument is important. A magnehelic must be level and in the vertical position when used to measure pressure. If the magnehelic is in any position other than level and vertical, the diaphragm can sag and cause a shift that leads to inaccurate readings.

Video 42. How does a magnehelic gauge work?
http://www.ask.com/youtube?q=How+Does+a+Magnehelic+Gauge+Work&qsrc=167&qo=channelNavigation&o=102140&l=dir

Photohelic Differential Pressure Device

The following is taken from Terrauniversal.com, *Photohelic Pressure Control*.

Photohelic gauges use a precise pressure switch to control low and high gas pressures. Knob controls adjust dual set points that allow for variable dead band control. These set points contain photocells that actuate double-pull double-throw relays when pressures reach a pre-set upper or lower limit, and they can be set up to control corrective damper action. As the room pressure changes in response to damper motion, and the indicator returns to the null band, damper motion is halted.

Source: Terrauniversal.com

Figure 34. Photohelic pressure gauge

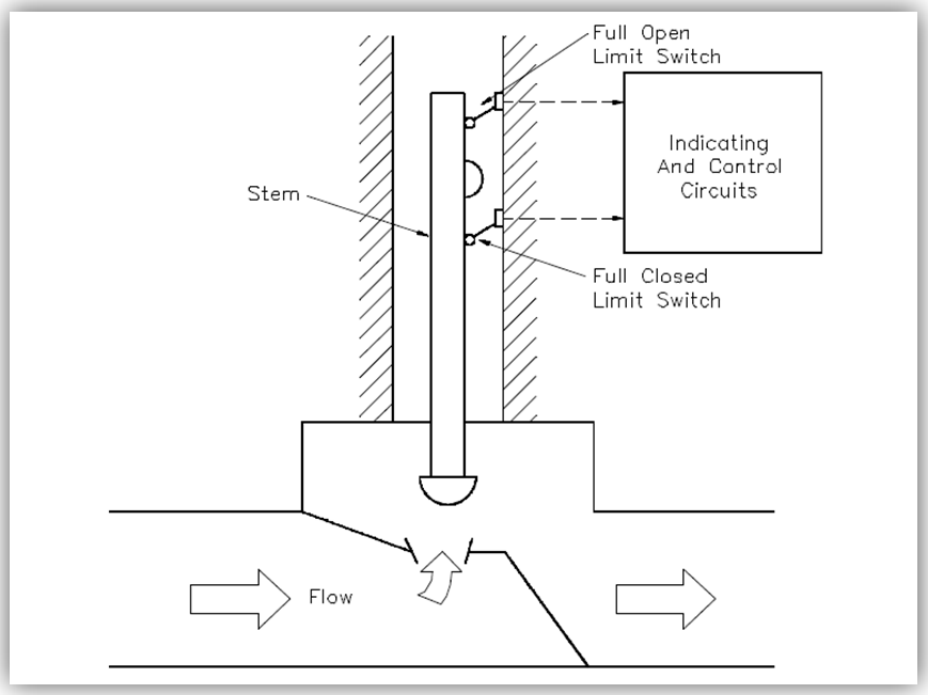
Video 43. Differential pressure transmitter response
http://www.youtube.com/watch?v=2p74l0E6D8g

d. For the position detection devices listed, explain how the detector provides an output representative of the position being represented:
- **Limit switches**
- **Potentiometer**
- **Linear variable differential transformer types**

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

Limit Switches

A limit switch is a mechanical device that can be used to determine the physical position of equipment. For example, an extension on a valve shaft mechanically trips a limit switch as it moves from open to shut or shut to open. The limit switch gives on/off output that corresponds to valve position. Normally, limit switches are used to provide full open or full shut indications as shown in figure 36.



Source:DOE-HDBK-1013/1-92

Figure 35. Limit switches

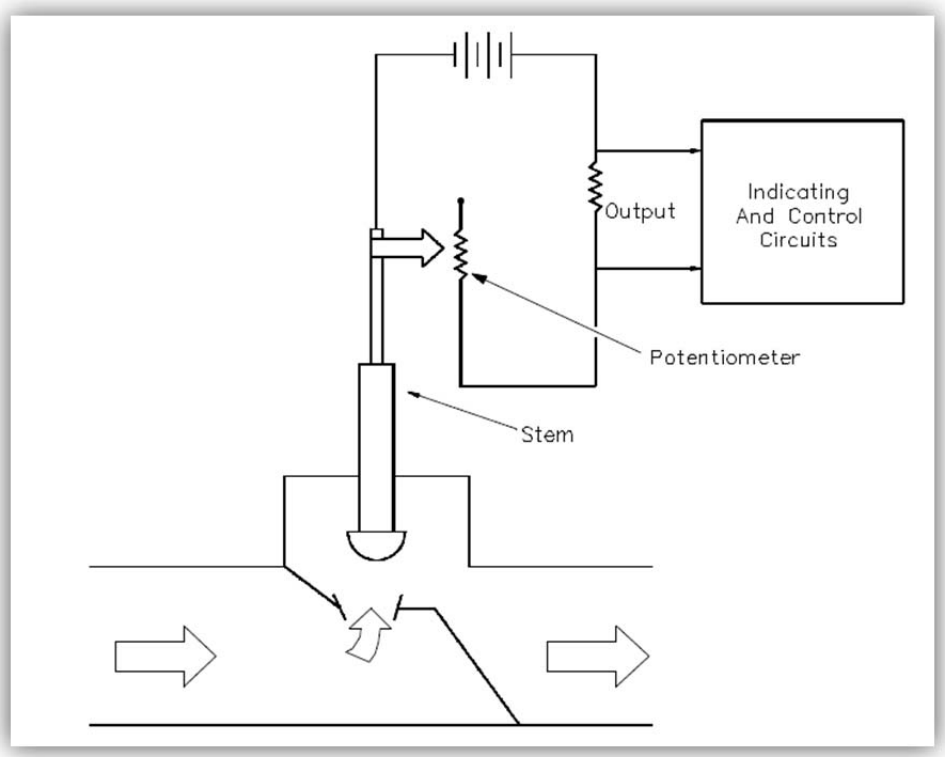
Many limit switches are the push-button variety. When the valve extension comes in contact with the limit switch, the switch depresses to complete, or turn on, the electrical circuit. As the valve extension moves away from the limit switches, spring pressure opens the switch, turning off the circuit.

Limit switch failures are normally mechanical in nature. If the proper indication or control function is not achieved, the limit switch is probably faulty. In this case, local position indication should be used to verify equipment position.

Video 44. Limit switch operation
http://www.youtube.com/watch?v=85oepWJAVbU

Potentiometer

Potentiometer valve position indicators, figure 37, provide an accurate indication of position throughout the travel of a valve or control rod. The extension is physically attached to a variable resistor. As the extension moves up or down, the resistance of the attached circuit changes, changing the amount of current flow in the circuit. The amount of current is proportional to the valve position.



Source: DOE-HDBK-1013/1-92

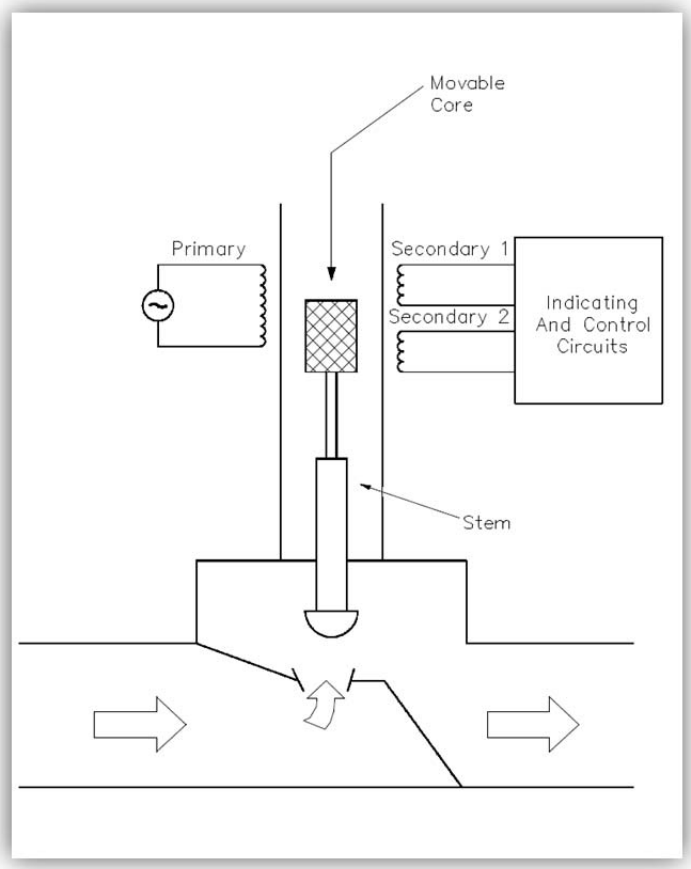
Figure 36. Potentiometer valve position indicator

Potentiometer valve position indicator failures are normally electrical in nature. An electrical short or open will cause the indication to fail at one extreme or the other. If an increase or decrease in the potentiometer resistance occurs, erratic indicated valve position occurs.

Video 45. What is a potentiometer?
http://www.youtube.com/watch?v=MXFvWLrpVSk

Linear Variable Differential Transformer Types

A linear variable differential transformer (LVDT) is a device that provides accurate position indication throughout the range of valve or control rod travel. (See figure 38.) Unlike the potentiometer position indicator, no physical connection to the extension is required.



Source: DOE-HDBK-1013/1-92

Figure 37. Linear variable differential transformer

The extension valve shaft, or control rod, is made of a metal suitable for acting as the movable core of a transformer. Moving the extension between the primary and secondary windings of a transformer causes the inductance between the two windings to vary; thereby, varying the output voltage proportional to the position of the valve or control rod extension. Figure 38 illustrates a valve whose position is indicated by an LVDT. If the open and shut position is all that is desired, two small secondary coils could be utilized at each end of the extension's travel.

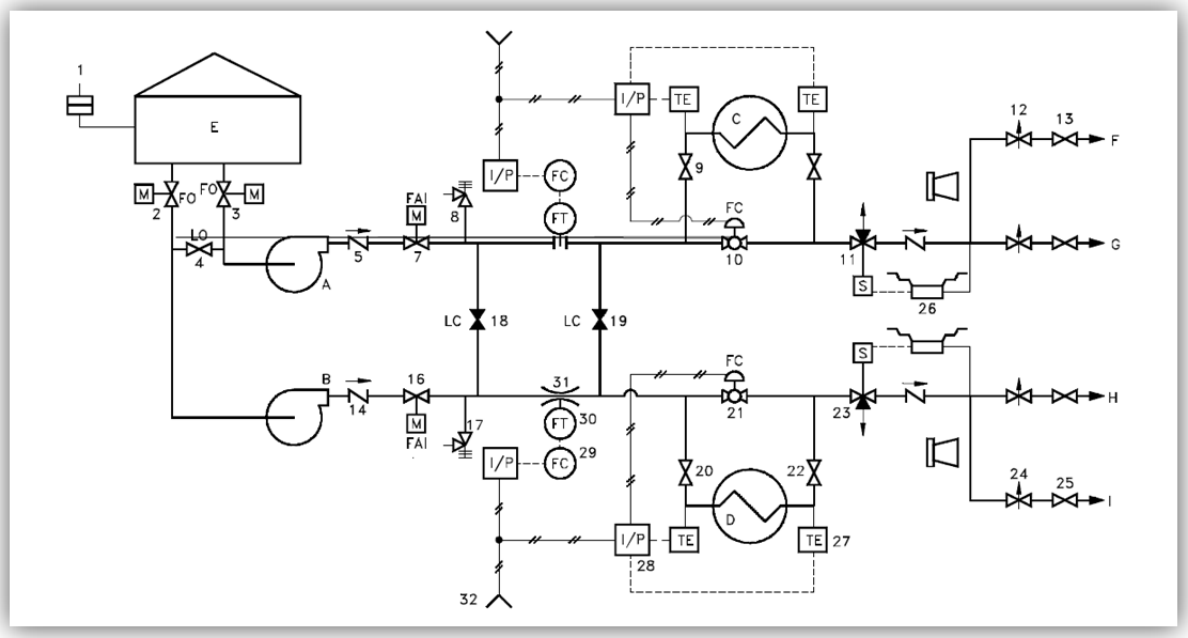
LVDTs are extremely reliable. As a rule, failures are limited to rare electrical faults that cause erratic or erroneous indications. An open primary winding will cause the indication to fail to some predetermined value equal to zero differential voltage. This normally corresponds to mid-stroke of the valve. A failure of either secondary winding will cause the output to indicate either full open or full closed.

Video 46. LVDT—simple operation and demo
http://www.youtube.com/watch?v=N_aXvUV1vMs

e. Referring to a piping and instrumentation drawing (P&ID) containing temperature, pressure, level, flow, or position detection components, explain their function in the designated system and relationship to system safety.

This is a performance-based KSA. The Qualifying Official will evaluate its completion. The following information from DOE-HDBK-1016/1-93 may be helpful.

P&IDs are usually designed to present functional information about a system or component. Examples are piping layout, flow paths, pumps, valves, instruments, signal modifiers, and controllers, as illustrated in figure 39.



Source: DOE-HDBK-1016/1-93

Figure 38. Piping and instrument drawings (P&IDs)

As a rule, P&IDs do not have a drawing scale and present only the relationship or sequence between components. Just because two pieces of equipment are drawn next to each other does not indicate that in the plant the equipment is even in the same building; it is just the next part or piece of the system. These drawings only present information on how a system functions, not the actual physical relationships.

Because P&IDs provide the most concise format for how a system should function, they are used extensively in the operation, repair, and modification of the plant.

Additional information related to P&IDs is available in DOE-HDBK-1016/1-93.

A three-part series of videos on P&IDs is available starting with the introduction as follows.

Video 47. Introduction to P&IDs
http://www.youtube.com/watch?v=dqInt1AoDV8&list=PL20028C3C0A1CB2D7

f. Discuss the importance of safety and process instrumentation to nuclear safety including redundancy and calibration requirements.

Redundancy

The following is taken from DOE Order 5480.30, Chg 1.

Design techniques, such as redundancy, physical separation, functional diversity, or diversity in component design and principles of operation, shall be used to prevent loss of the protection function. The protection shall be sufficient to ensure no single failure results in loss of protection and capability exists to test channels independently to determine failure and loss of redundancy.

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel that is common to the control and protection system leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system.

The following is taken from International Atomic Energy Agency, Safety Series No. NS-G-1.3, *Instrumentation and Control Systems Important to Safety in Nuclear Power Plants*.

Higher reliability is achieved by the use of redundancy or diversity. For example, it may be possible to monitor reactor power with multiple channels or by diverse means such as measurement of neutron flux or temperature and fluid flow or pressure. The use of redundancy provides protection against random failures. Use of diversity provides protection against certain CCFs.

The single failure criterion is a deterministic approach to ensuring that a minimal redundancy of a system or of a group of equipment items is obtained. It is based on the general experience that even components and equipment that are manufactured to high standards of quality may sometimes fail to function, in a way and at a time that is random and unpredictable.

Redundancy is commonly used in I&C systems important to safety to achieve system reliability goals and/or conformity with the single failure criterion. For redundancy to be fully effective there should be independence. Taken alone, redundancy increases the probability of false operation. Coincidence of redundant signals for equipment or a rejection scheme for bogus signals that is based on inter-comparisons of the redundant signals is commonly used to obtain an appropriate balance of reliability and freedom from false operation.

Calibration

The following is taken from DOE G 414.1-2B, Admin Chg 2.

Measuring and test equipment (M&TE) used for inspections, tests, monitoring, and data collection should be calibrated, maintained, and controlled using a documented process. M&TE should be checked before use to ensure that it is of the proper type, range, accuracy, and precision, and that it is uniquely identified and traceable to its calibration data. Procedures should be established for testing, retesting, adjusting, and recalibrating M&TE. M&TE should be calibrated to standards traceable to the National Institute of Standards and Technology (NIST) or other nationally recognized standards when appropriate. When calibrating and/or checking M&TE for use, computer programs/software that are part of M&TE should be checked to ensure verification and validation (V&V) have been performed for the computer programs/software, and that the V&V is current.

The use of M&TE should be traceable to the item inspected because measurements and tests performed with the M&TE may need to be reevaluated if the M&TE is subsequently found to be out of its acceptable calibration range. Systems that rely on recording the identity of the M&TE in work packages are ineffective because review of all work packages to identify each use of a particular M&TE is almost impossible.

The following is taken from International Atomic Energy Agency, Safety Series No. NS-G-1.3, *Instrumentation and Control Systems Important to Safety in Nuclear Power Plants*.

Two areas of concern in relation to access control are set points adjustments and calibration adjustments, because of their importance in preventing degraded system performance due to potential errors in operation or maintenance.

The allowable limit is used for instruments that require periodic testing and surveillance. The margin between the allowable limit and the nominal set point comprises random uncertainties in instrument calibration, random instrument errors, and errors due to instrument drift. If a set point is found to be beyond the allowable limit, immediate corrective action should be taken.

Equipment selected for systems important to safety should be of a proven design whenever possible, should be consistent with the reliability goals, and should facilitate meeting the requirements for calibration, testing, maintenance, and repair.

9. **Nuclear safety specialists must demonstrate a familiarity level knowledge of P&ID.**

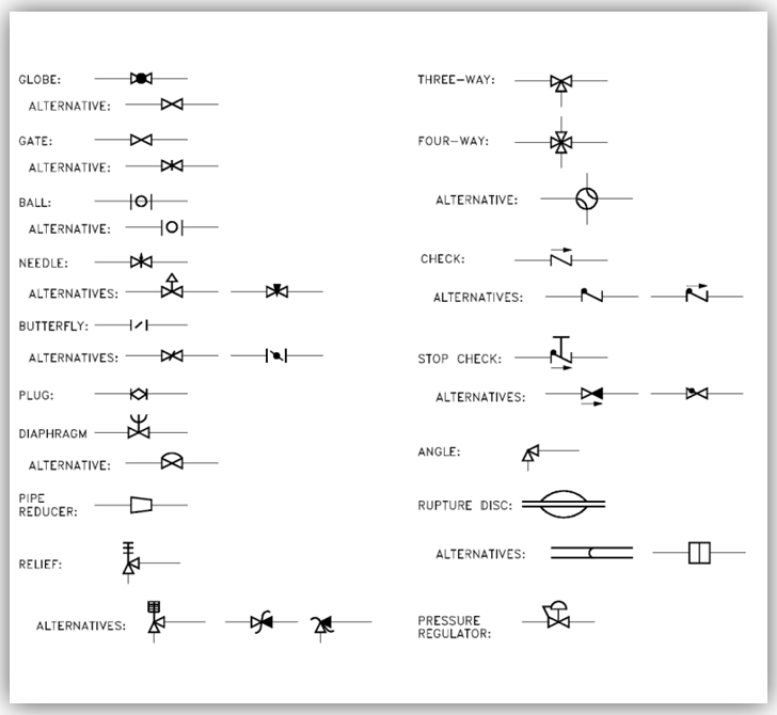
   a. **Given a P&ID, identify/interpret the symbols used for system components including the following at a minimum:**
      - **Valves**
      - **Pumps**
      - **Heat exchangers**
      - **Filters/strainers**
      - **Fans**
      - **Compressors**
      - **Instruments**
      - **Indicators**
      - **Controllers**

The following is taken from DOE-HDBK-1016/1-93.

Valves

Valves are used to control the direction, flow rate, and pressure of fluids. Figure 40 shows the symbols that depict the major valve types.

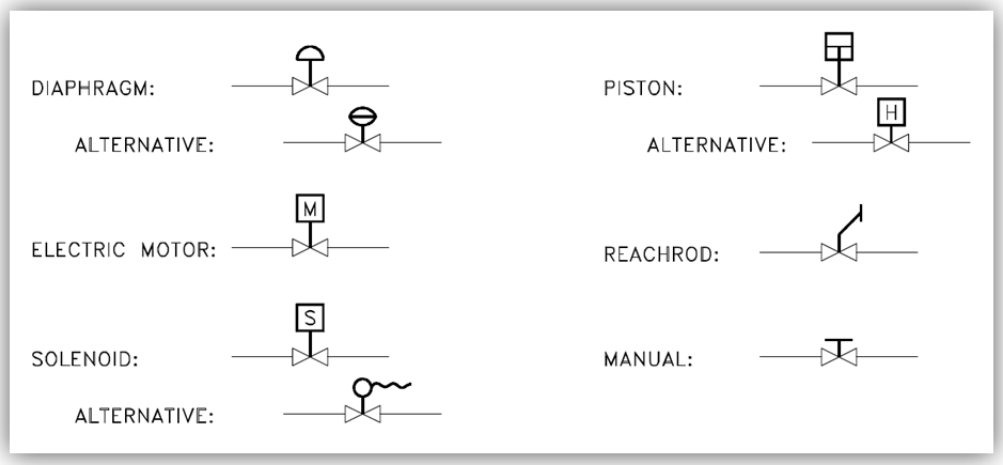
It should be noted that globe and gate valves will often be depicted by the same valve symbol. In such cases, information concerning the valve type may be conveyed by the component identification number or by the notes and legend section of the drawing; however, in many instances even that may not hold true.



Source: DOE-HDBK-1016/1-93

Figure 39. Valve symbols

Some valves are provided with actuators to allow remote operation, to increase mechanical advantages, or both. Figure 41 shows the symbols for the common valve actuators. Note that although each is shown attached to a gate valve, an actuator can be attached to any type of valve body. If no actuator is shown on a valve symbol, it may be assumed the valve is equipped only with a hand wheel for manual operation.



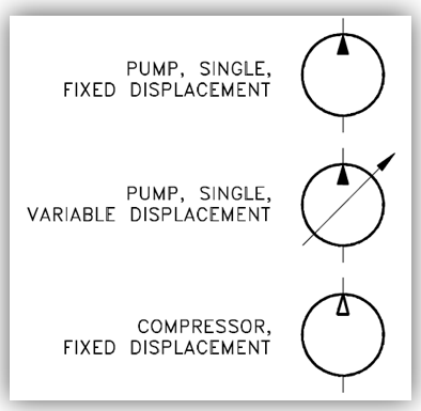
Source: DOE-HDBK-1016/1-93

Figure 40. Valve actuator symbols

The combination of a valve and an actuator is commonly called a control valve. Control valves are symbolized by combining the appropriate valve symbol and actuator symbol, as illustrated in figure 41. Control valves can be configured in many different ways. The most commonly found configurations are to manually control the actuator from a remote operating station, to automatically control the actuator from an instrument, or both.

Pumps

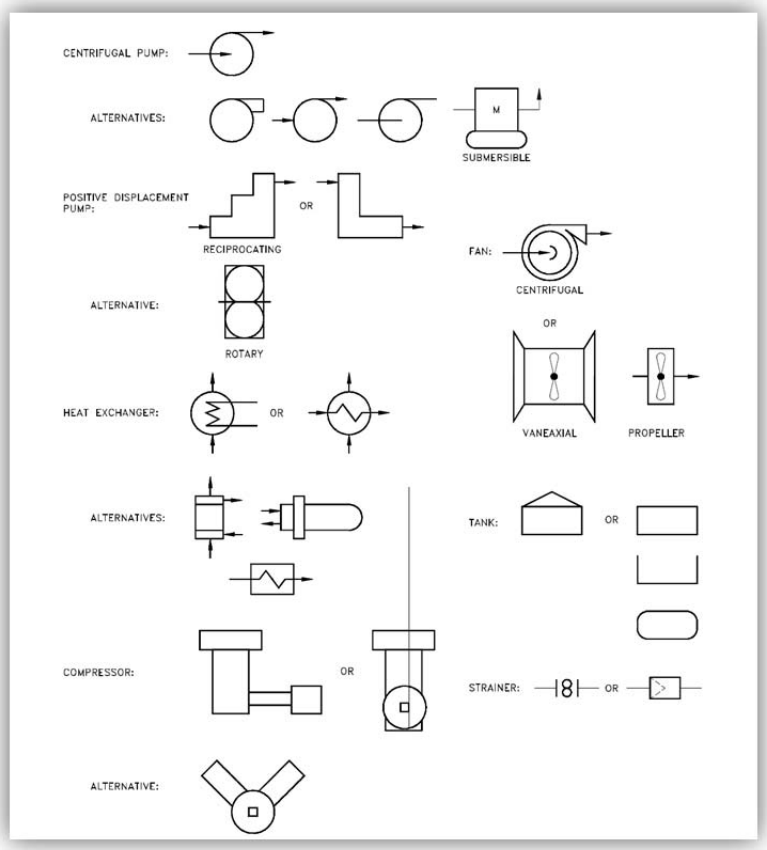
In the broad area of fluid power, two categories of pump symbols are used, depending on the motive media being used (i.e., hydraulic or pneumatic). The basic symbol for the pump is a circle containing one or more arrow heads indicating the direction(s) of flow with the points of the arrows in contact with the circle. Hydraulic pumps are shown by solid arrow heads. Pneumatic compressors are represented by hollow arrow heads. Figure 42 provides common symbols used for pumps (hydraulic) and compressors (pneumatic) in fluid power diagrams.



Source: DOE-HDBK-1016/1-93

Figure 41. Fluid power pump and compressor symbols

Figure 43 depicts the engineering symbols for the most common major components, including heat exchanger, strainers, fans, and compressors.



Source: DOE-HDBK-1016/1-93

Figure 42. Symbols for major components

Instruments

One of the main purposes of a P&ID is to provide functional information about how instrumentation in a system or piece of equipment interfaces with the system or piece of equipment. Because of this, a large amount of the symbology appearing on P&IDs depicts instrumentation and instrument loops.

The symbols used to represent instruments and their loops can be divided into four categories. Generally each of these four categories uses the component identifying (labeling) scheme identified in table 6. The first column of table 6 lists the letters used to identify the parameter being sensed or monitored by the loop or instrument. The second column lists the letters used to indicate the type of indicator or controller. The third column lists the letters used to indicate the type of component. The fourth column lists the letters used to indicate the type of signals that are being modified by a modifier.

Table 6. Instrument identifiers

Sensed Parameter	Type of Indicator or Controller	Type of Components	Type of Signal
F = flow T = temperature P = pressure I = current L = level V = voltage Z = position	R = recorder I = indicator C = controller	T = transmitter M = modifier E = element	I = current V = voltage P = pneumatic

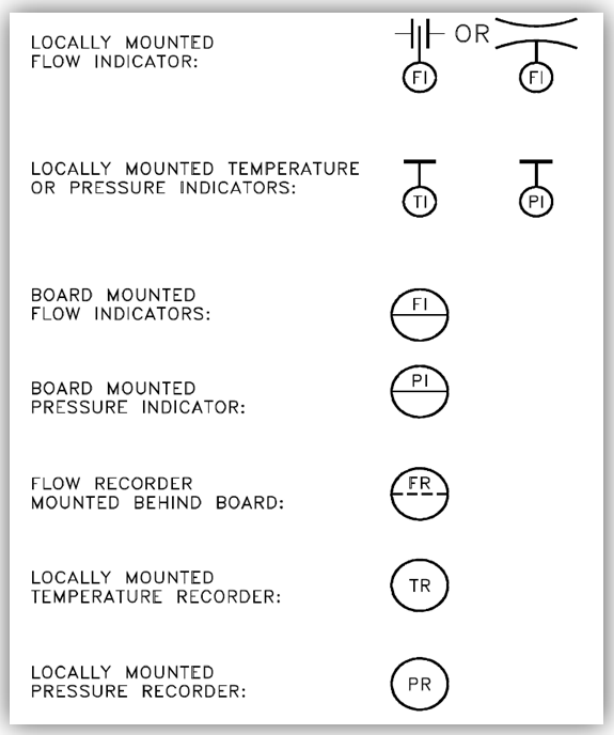
Source: DOE-HDBK-1016/1-93

The first three columns of table 6 are combined such that the resulting instrument identifier indicates its sensed parameter, the function of the instrument, and the type of instrument. The fourth column is used only in the case of an instrument modifier and is used to indicate the types of signals being modified. The following is a list of example instrument identifiers constructed from table 6.

FIC = flow indicating controller
FM = flow modifier
PM = pressure modifier
TE = temperature element
TR = temperature recorder
LIC = level indicating controller
TT = temperature transmitter
PT = pressure transmitter
FE = flow element
FI = flow indicator
TI = temperature indicator
FC = flow controller

Indicators

Indicators and recorders are instruments that convert the signal generated by an instrument loop into a readable form. The indicator or recorder may be locally or board mounted, and like modifiers and transmitters this information is indicated by the type of symbol used. Figure 44 provides examples of the symbols used for indicators and recorders and how their location is denoted.

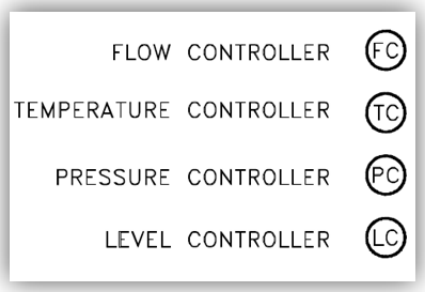


Source: DOE-HDBK-1016/1-93

Figure 43. Indicators and recorders

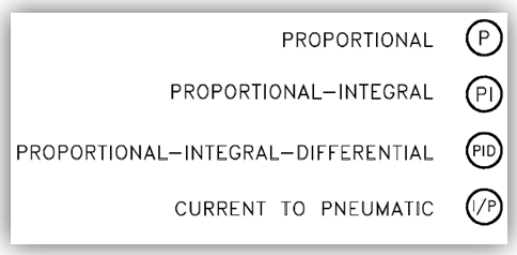
Controllers

Controllers process the signal from an instrument loop and use it to position or manipulate some other system component. Generally they are denoted by placing a "C" in the balloon after the controlling parameter as shown in figure 45. There are controllers that serve to process a signal and create a new signal. These include proportional controllers, proportional-integral controllers, and proportional-integral-differential controllers. The symbols for these controllers are illustrated in figure 46. Note that these types of controllers are also called signal conditioners.



Source: DOE-HDBK-1016/1-93

Figure 44. Controllers



Source: DOE-HDBK-1016/1-93

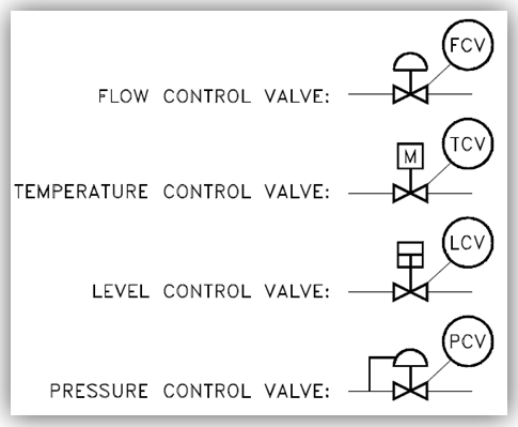
Figure 45. Signal conditioners

b. Identify how valve conditions (open/closed) are depicted.

The following is taken from DOE-HDBK-1016/1-93.

A control valve may serve any number of functions within a fluid system. To differentiate between valve uses, a balloon labeling system is used to identify the function of a control valve, as shown in figure 47. The common convention is that the first letter used in the valve designator indicates the parameter to be controlled by the valve. For example:

F = flow
T = temperature
L = level
P = pressure
H = hand (manually operated valve)

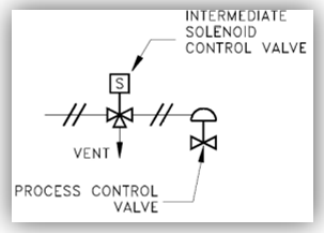


Source: DOE-HDBK-1016/1-93

Figure 46. Control valve designations

The second letter is usually a “C” and identifies the valve as a controller, or active component, as opposed to a hand-operated valve. The third letter is a “V” to indicate that the piece of equipment is a valve.

In many cases, remote control of a valve is accomplished by using an intermediate, small control valve to operate the actuator of the process control valve. The intermediate control valve is placed in the line supplying motive force to the process control valve, as shown in figure 48. In this example, air to the process air-operated control valve is controlled by the solenoid-operated, three-way valve in the air supply line. The three-way valve may supply air to the control valve's diaphragm or vent the diaphragm to the atmosphere.

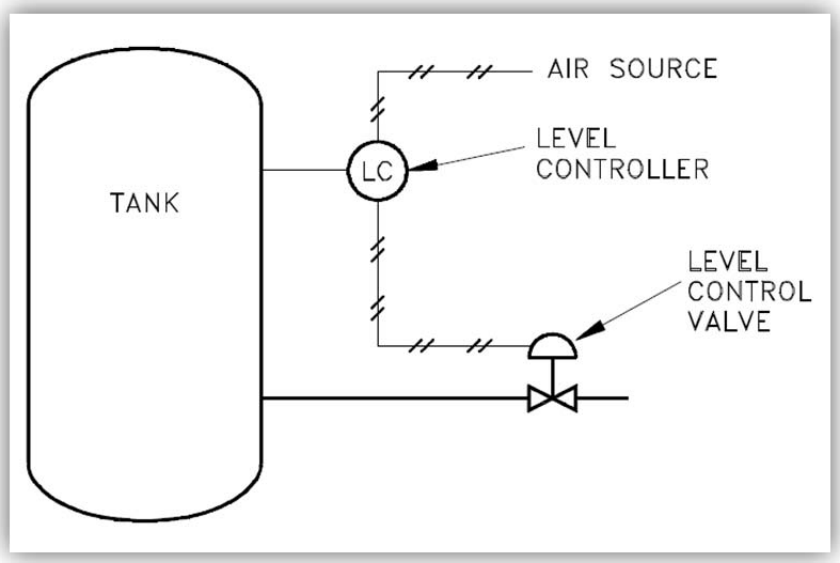


Source:DOE-HDBK-1016/1-93

Figure 47. Remotely controlled valve

Note that the symbols in figure 48 do not provide the reader with enough information to determine whether applying air pressure to the diaphragm opens or closes the process control valve, or whether energizing the solenoid pressurizes or vents the diaphragm. Figure 48 is incomplete in that it does not show the electrical portion of the valve control system nor does it identify the source of the motive force (compressed air). Although figure 48 informs the reader of the types of mechanical components in the control system and how they interconnect, it does not provide enough information to determine how those components react to a control signal.

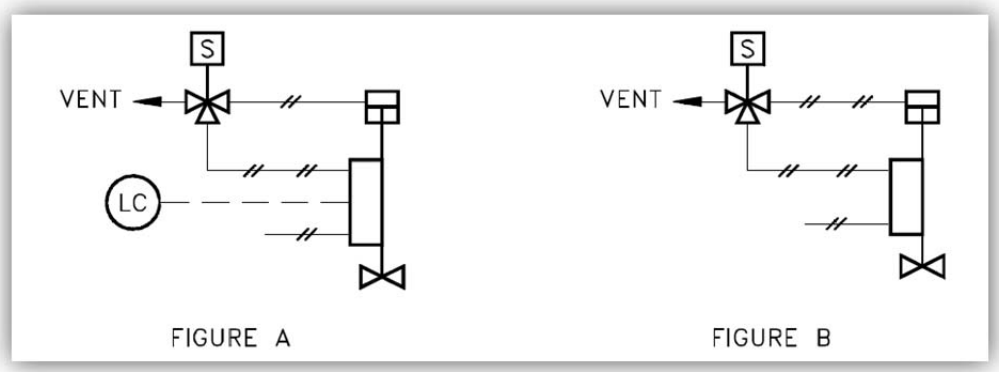
Control valves operated by an instrument signal are symbolized in the same manner as those shown previously, except the output of the controlling instrument goes to the valve actuator. Figure 49 shows a level instrument (designated LC) that controls the level in the tank by positioning an air-operated diaphragm control valve. Again, note that figure 49 does not contain enough information to enable the reader to determine how the control valve responds to a change in level.



Source: DOE-HDBK-1016/1-93

Figure 48. Level control valve

An additional aspect of some control valves is a valve positioner that allows more precise control of the valve. This is especially useful when instrument signals are used to control the valve. An example of a valve positioner is a set of limit switches operated by the motion of the valve. A positioner is symbolized by a square box on the stem of the control valve actuator. The positioner may have lines attached for motive force, instrument signals, or both. Figure 50 shows two examples of valves equipped with positioners. Note that, although these examples are more detailed than those of figure 48 and 49, the reader still does not have sufficient information to fully determine response of the control valve to a change in control signal.



Source: DOE-HDBK-1016/1-93

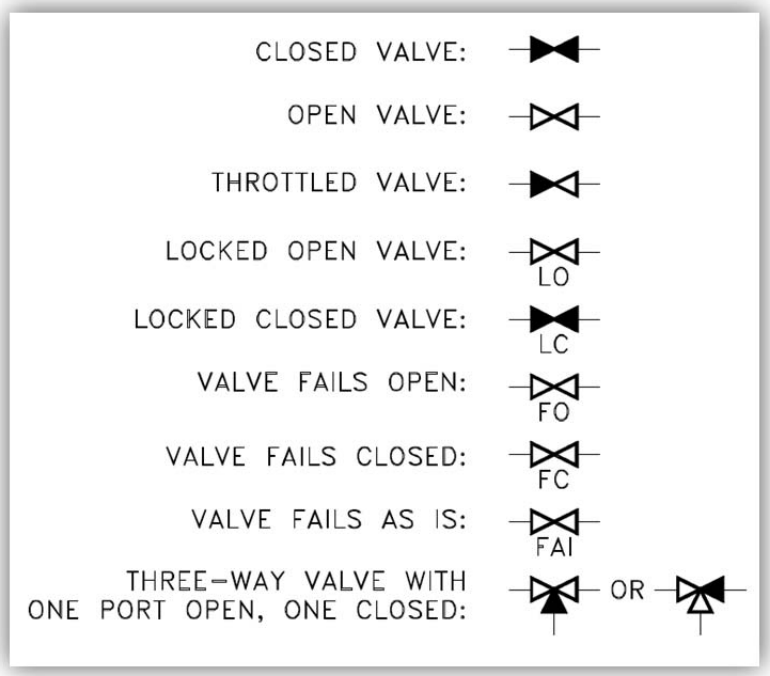
Figure 49. Control valve with valve positioners

In example A of figure 50, the reader can reasonably assume that opening of the control valve is in some way proportional to the level it controls and that the solenoid valve provides an override of the automatic control signals. However, the reader cannot ascertain whether it opens or closes the control valve. The reader cannot determine in which direction the valve moves in response to a change in the control parameter. In example B of figure 50, the reader can make the same general assumptions as in example A, except the control signal is unknown. Without additional information, the reader can only assume the air supply provides the control signal and motive force for positioning the control valve. Even when valves are equipped with positioners, the positioner symbol may appear only on detailed system diagrams. Larger, overall system diagrams usually do not show this much detail and may show the examples of figure 50 as air-operated valves with no special features.

Before a diagram or print can be properly read and understood, the basic conventions used by P&IDs to denote valve positions and failure modes must be understood. The reader must be able to determine the valve position, know if this position is normal, know how the valve will fail, and in some cases know if the valve is normally locked in that position.

Figure 51 illustrates the symbols used to indicate valve status. Unless otherwise stated, P&IDs indicate valves in their “normal” position. This is usually interpreted as the normal or primary flowpath for the system. An exception is safety systems that are normally shown in their standby or non-accident condition. Three-way valves are sometimes drawn in the position they will fail to instead of always being drawn in their normal position. This will

either be defined as the standard by the system of drawings or noted in some manner on the individual drawings.



Source: DOE-HDBK-1016/1-93

Figure 50. Valve status symbols

c. Determine and follow system flowpath(s).

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

d. Discuss the role of P&IDs relative to identification of failure modes and mapping fault propagation through networks to support the identification of accident vulnerabilities.

Refer to KSA b of this competency for a discussion regarding identification of failure modes and the role of P&IDs.

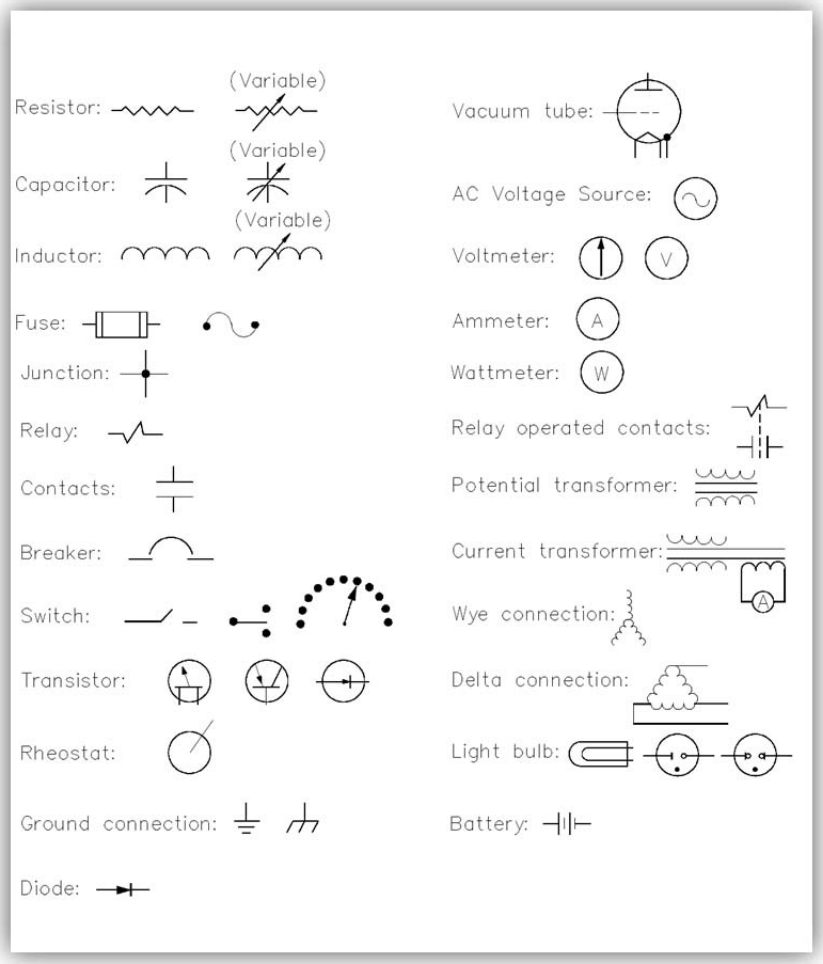
10. Nuclear safety specialists must demonstrate a familiarity level knowledge of electrical diagrams and schematics.

a. Given a system diagram, identify/interpret the following symbols:
- **Motors**
- **Transformers**
- **Breakers**
- **Generators**
- **Batteries**

This is a performance-based KSA. The Qualifying Official will evaluate its completion. The following information may be helpful.

The following is taken from DOE-HDBK-1011/1-92.

The symbols for the various electrical components that will appear on electrical diagrams and schematics are shown in figure 52.



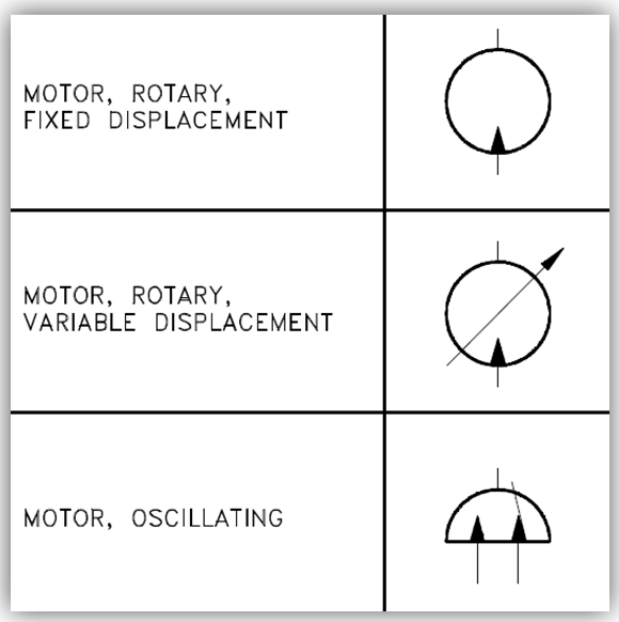
Source: DOE-HDBK-1011/1-92

Figure 51. Electrical symbols

Motors

The following is taken from DOE-HDBK-1016/1-93.

Rotary actuators are generally called motors and may be fixed or variable. Several of the more common rotary symbols are shown in figure 53.



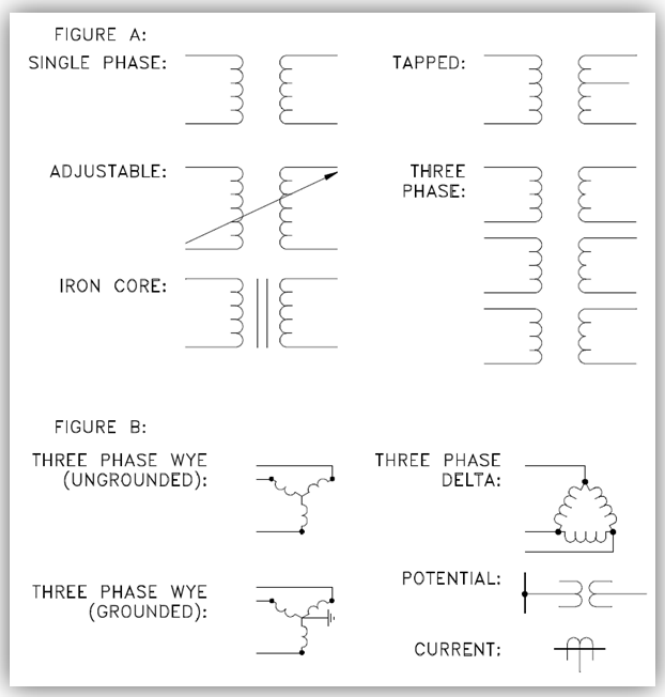
Source: DOE-HDBK-1016/1-93

Figure 52. Symbols for motors

Transformers

The following is taken from DOE-HDBK-1016/1-93.

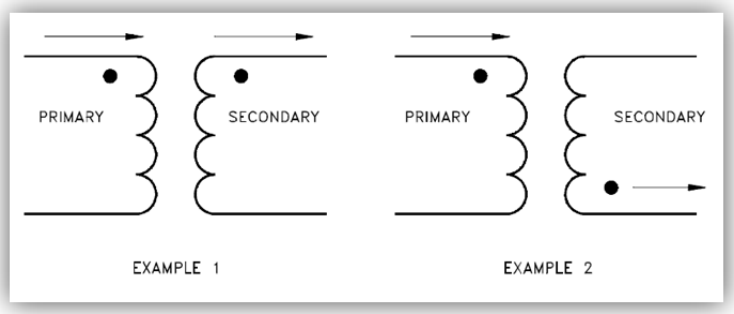
The basic symbols for the various types of transformers are shown in figure 54A. Figure 54B shows how the basic symbol for the transformers is modified to represent specific types and transformer applications.



Source: DOE-HDBK-1016/1-93

Figure 53. Basic transformer symbols

In addition to the transformer symbol, polarity marks are sometimes used to indicate current flow in the circuit. This information can be used to determine the phase relationship (polarity) between the input and output terminals of a transformer. The marks usually appear as dots on a transformer symbol, as shown in figure 55.



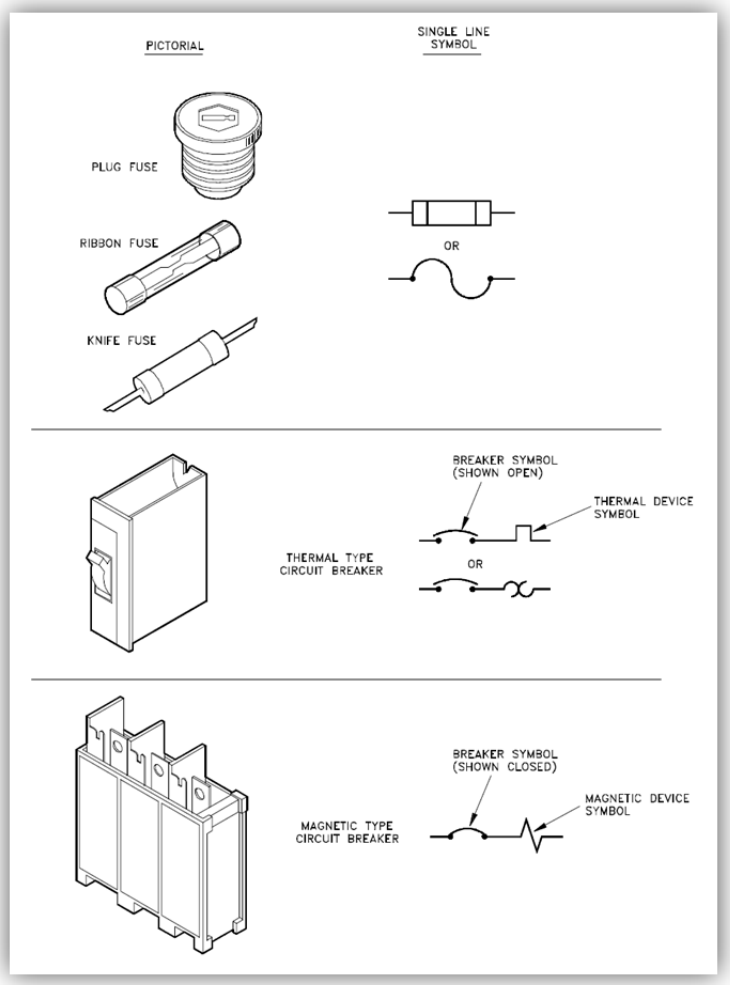
Source: DOE-HDBK-1016/1-93

Figure 54. Transformer polarity

Breakers

The following is taken from DOE-HDBK-1016/1-93.

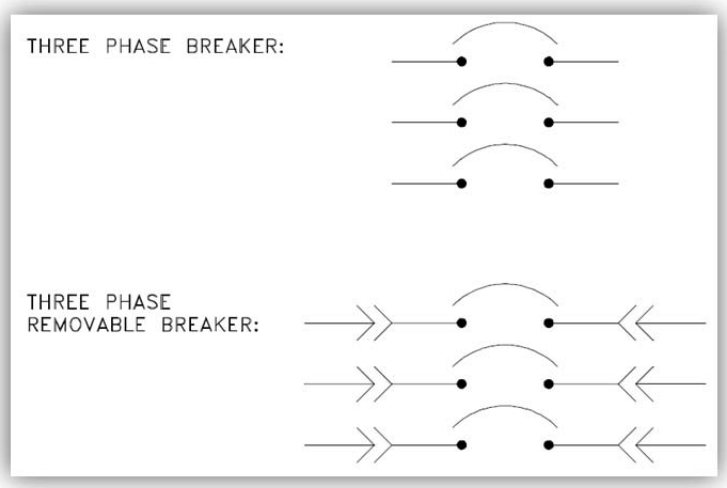
Figure 56 depicts basic fuse and circuit breaker symbols for single-phase applications. In addition to the graphic symbol, most drawings will provide the rating of the fuse next to the symbol. The rating is usually in amperes.



Source: DOE-HDBK-1016/1-93

Figure 55. Fuse and circuit breaker symbols

When fuses, breakers, or switches are used in three-phase systems, the three-phase symbol combines the single-phase symbol in triplicate as shown in figure 57. Also shown is the symbol for a removable breaker that is a standard breaker symbol placed between a set of chevrons. The chevrons represent the point at which the breaker disconnects from the circuit when removed.



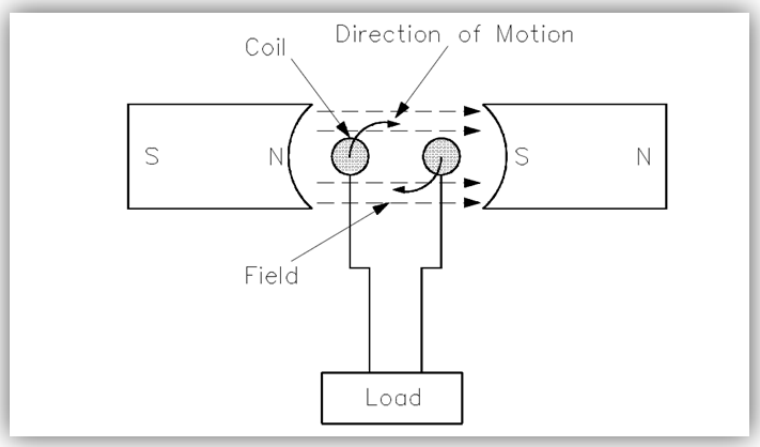
Source: DOE-HDBK-1016/1-93

Figure 56. Three-phase and removable breaker symbols

Generators

The following is taken from DOE-HDBK-1011/1-92.

A generator is a machine that converts mechanical energy into electrical energy by using the principle of magnetic induction. Magnetic induction is used to produce a voltage by rotating coils of wire through a stationary magnetic field, as shown in figure 58, or by rotating a magnetic field through stationary coils or wire. This is one of the most useful and widely employed applications of producing vast quantities of electric power.



Source: DOE-HDBK-1011/1-92

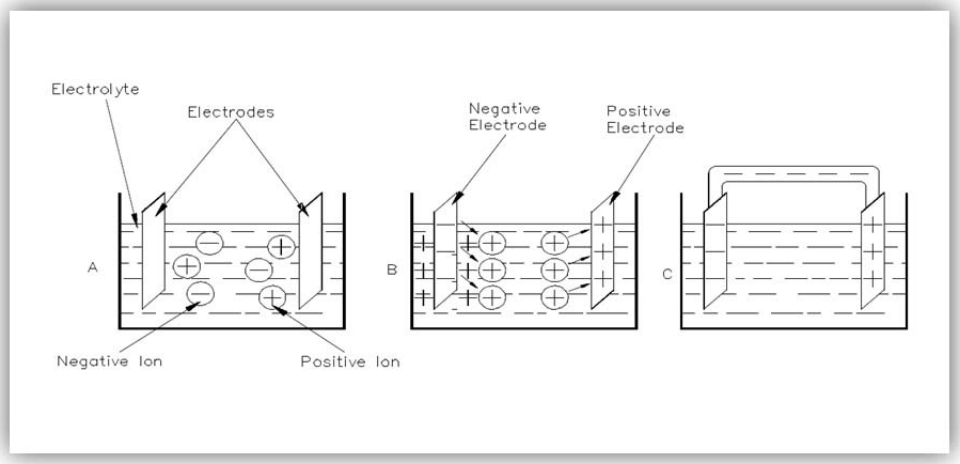
Figure 57. Generator—electromagnetic induction

Batteries

The following is taken from DOE-HDBK-1011/1-93.

A battery consists of two or more chemical cells connected in series. The combination of materials within a battery is used for the purpose of converting chemical energy into electrical energy. To understand how a battery works, we must first discuss the chemical cell.

The chemical cell is composed of two electrodes made of different types of metal or metallic compounds that are immersed in an electrolyte solution. The chemical actions that result are complicated, and they vary with the type of material used in cell construction. Some knowledge of the basic action of a simple cell will be helpful in understanding the operation of a chemical cell in general. In the cell, electrolyte ionizes to produce positive and negative ions. (See figure 59.) Simultaneously, chemical action causes the atoms within one of the electrodes to ionize.



Source: DOE-HDBK-1011/1-92

Figure 58. Basic chemical battery

Video 48. Symbols used for electrical diagrams
http://www.youtube.com/watch?v=h3ObQurHibk

b. Given the appropriate diagram, state the condition (energized/de-energized) in which all electrical devices are shown, unless otherwise noted in the diagram.

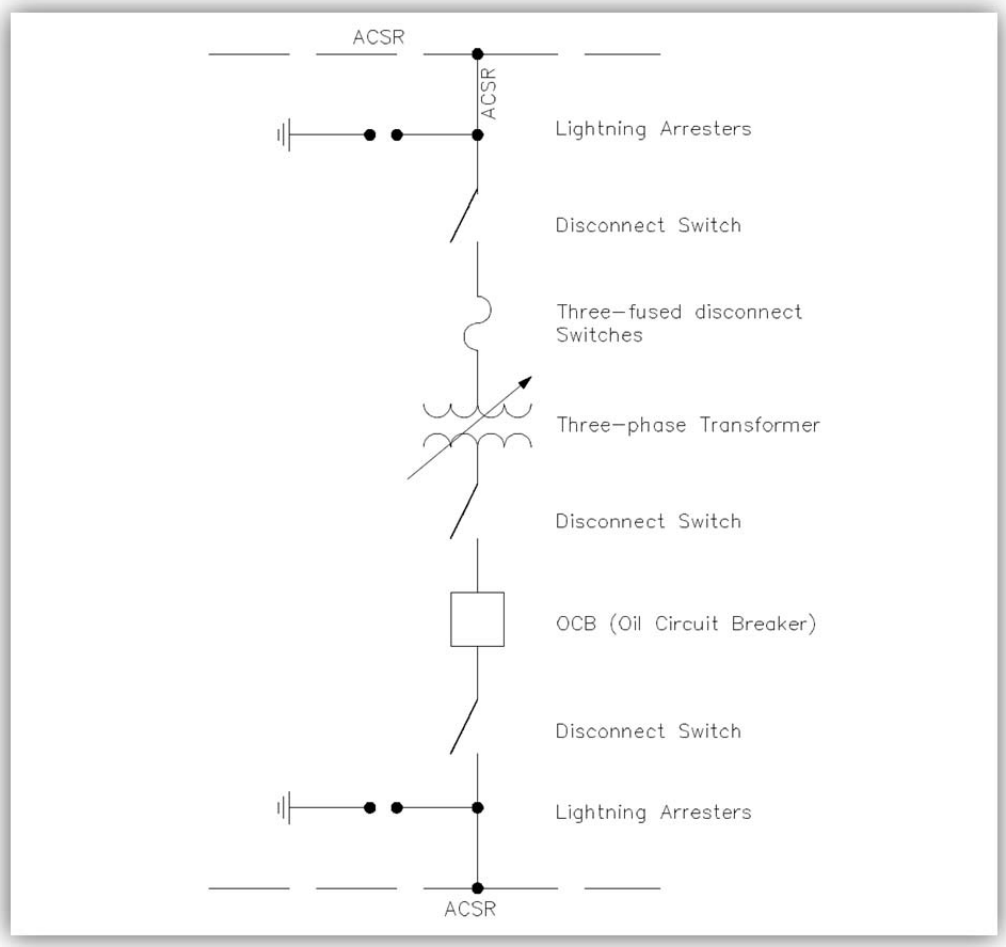
c. Given a system diagram, identify the power sources and/or loads and their status.

KSAs b and c are performance-based. The Qualifying Official will evaluate their completion.

d. Discuss the role of electrical one-line diagrams for identifying failure modes and for mapping fault propagation through networks to support the identification of accident vulnerabilities.

The following is taken from DOE-HDBK-1011/1-92.

The one-line, or single-line, diagram shows the components of a circuit by means of single lines and the appropriate graphic symbols. One-line diagrams show two or more conductors that are connected between components in the actual circuit. The one-line diagram shows all pertinent information about the sequence of the circuit, but does not give as much detail as a schematic diagram. Normally, the one-line diagram is used to show highly complex systems without showing the actual physical connections between components and individual conductors. As an example, figure 60 shows a typical one-line diagram of an electrical substation.



Source: DOE-HDBK-1011/1-92

Figure 59. One-line diagram

The following is taken from BrightPlanIT, Inc., *Optimize Your Data Center*.

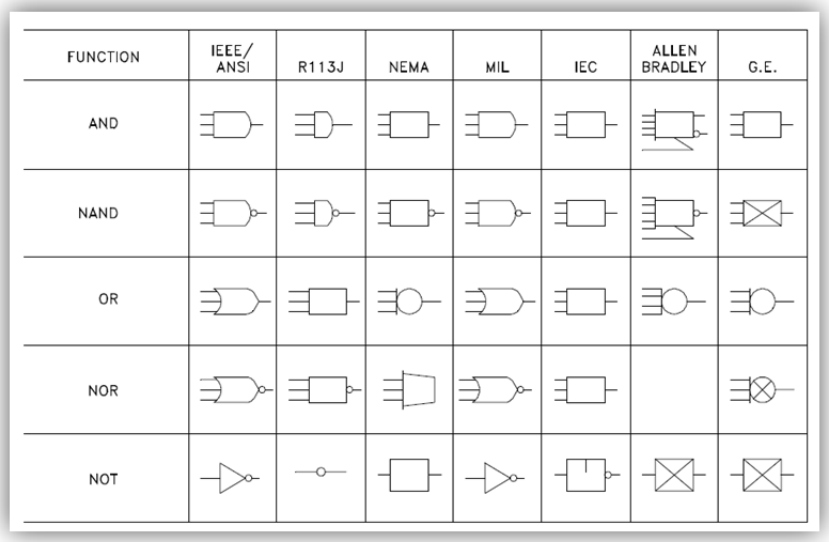
One-line diagrams help identify fault locations, simplify troubleshooting, identify potential sources of electric energy during lock-out/tag-out procedures, and help ensure safe operation.

11. Nuclear safety specialists must demonstrate a familiarity level knowledge of electrical logic diagrams.

a. Given a logic diagram, identify/interpret the symbols used on logic diagrams to represent the components.

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

The following information from DOE-HDBK-1016/2-93 may be helpful.



Source: DOE-HDBK-1016/2-93

Figure 60. Basic logic symbols

The following is taken from DOE-HDBK-1016/2-93.

Basic Logic Circuits

AND gate—Provides an output (on) when all its inputs are on. When any one of the inputs is off, the gate's output is off.

OR gate—Provides an output (on) when any one or more of its inputs is on. The gate is off only when all of its inputs are off.

NOT gate—Provides a reversal of the input. If the input is on, the output will be off. If the input is off, the output will be on.

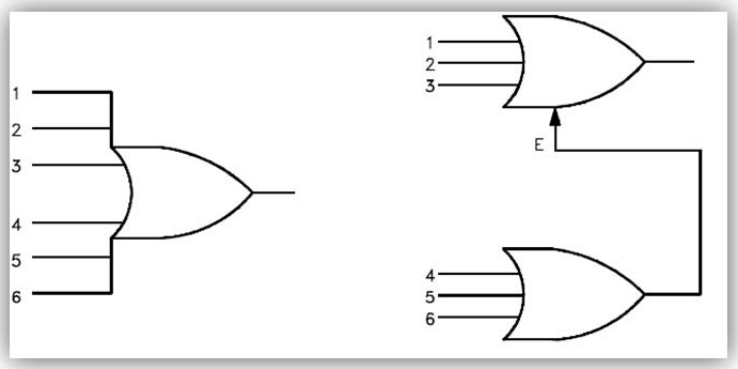
Special "Conjunction" Logic Circuit

NAND gate—Is the opposite (NOT) of an AND gate's output. It provides an output (on) except when all the inputs are on.

NOR gate—Is the opposite (NOT) of an OR gate's output. It provides an output only when all inputs are off.

COINCIDENCE Gate

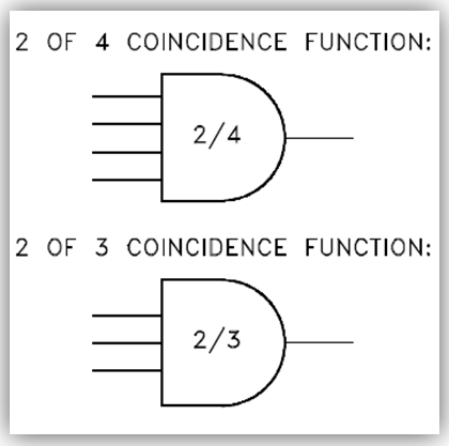
Figure 62 demonstrates two basic ways to show multiple inputs using an OR gate as an example.



Source: DOE-HSBK-1016/2-93

Figure 61. Conventions for depicting multiple inputs

A common variation of the AND gate is the COINCIDENCE gate where there are theoretically no limits to the number of inputs a gate can have. The COINCIDENCE gate behaves as an AND gate except only a specific number of the total number of inputs needs to be on for the gate's output to be on. (See figure 63.)



Source: DOE-HDBK-1016/2-93

Figure 62. COINCIDENCE gate

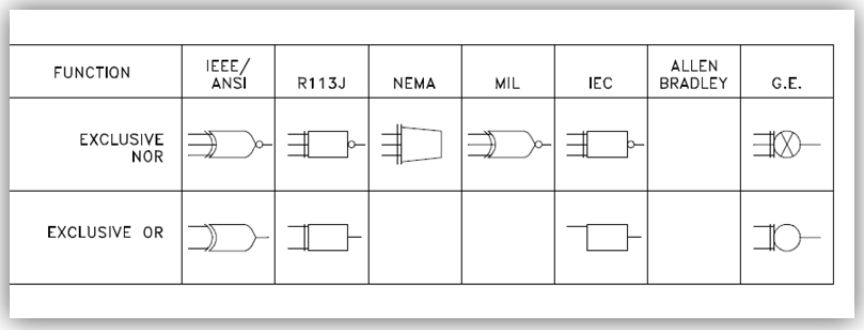
The fraction in the logic symbol indicates that the AND gate is a COINCIDENCE gate. The numerator of the fraction indicates the number of inputs that must be on for the gate to be on. The denominator states the total number of inputs to the gate.

EXCLUSIVE Gate

Two variations of the OR gate are as follows:

- EXCLUSIVE OR—Provides an output (on) only when one of the inputs is on. Any other combination results in no output (off).
- EXCLUSIVE NOR—Is the opposite (NOT) of an EXCLUSIVE OR gate's output. It provides an output (on) only when all inputs are on or when all inputs are off.

Figure 64 shows EXCLUSIVE OR and EXCLUSIVE NOR gates.



Source: DOE-HDBK-1016/2-93

Figure 63. EXCLUSIVE OR and EXCLUSIVE NOR gates

Video 49. Basic logic gates
http://www.youtube.com/watch?v=eJHmVlzH_9Q

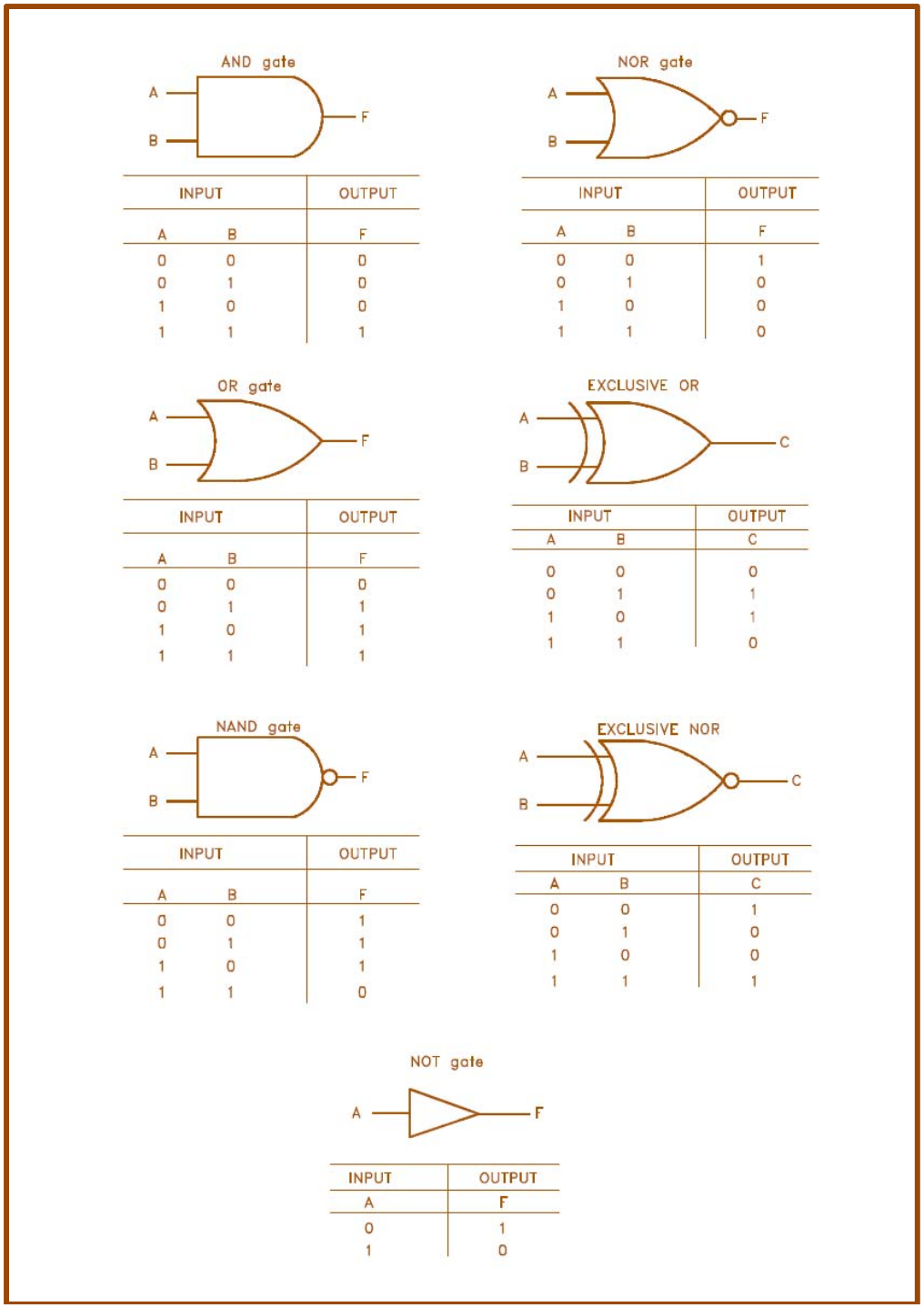
b. Given a logic diagram and appropriate information, determine the output of each component and the logic circuit.

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

The following information from DOE-HDBK-1016/2-93 may be helpful.

Truth Tables

When a logic gate has only two inputs, or the logic circuit to be analyzed has only one or two gates, it is fairly easy to remember how a specific gate responds and determine the output of the gate or circuit. But, as the number of inputs and/or the complexity of the circuit grows, it become more difficult to determine the output of the gate or circuit. Truth tables, as illustrated in figure 65, are tools designed to help solve this problem. A truth table has a column for input of each gate and column for the output of each gate. The number of rows needed is based on the number of inputs, so that every combination of input signal is listed (mathematically the number of rows is 2^n, where n = number of inputs). In truth tables, the on and off status of the inputs and outputs is represented using 0s and 1s. As previously stated 0 = off and 1 = on. Figure 65 lists truth tables for the seven basic logic gates.



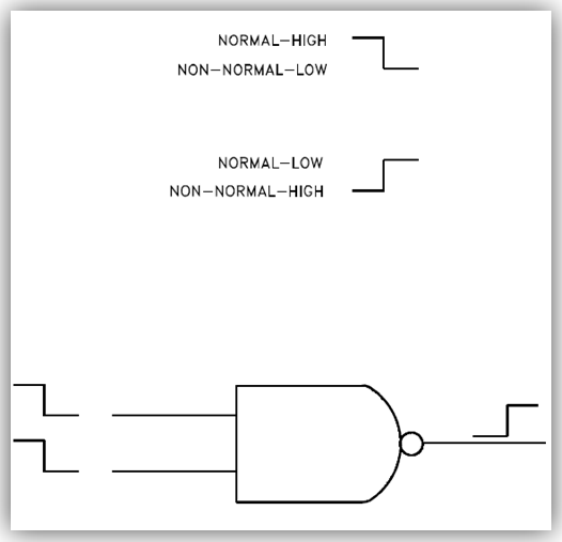
Source: DOE-HDBK-1016/2-93

Figure 64. Truth tables

Video 50. Converting digital circuits to truth tables
http://www.youtube.com/watch?v=oDMmsa8WxJU

Logic Diagrams

When reading logic prints the reader usually must decide the input values to each gate. But, occasionally the print will provide information as to the normal state of each logic gate. This is denoted by a symbol similar to the bistable symbol, as shown in figure 66. The symbol is drawn so that the first part of the square wave indicates the normal state of the gate. The second part of the square wave indicates the off-normal state of the gate. Figure 66 illustrates how this notation is applied.



Source: DOE-HDBK-1016/2-93

Figure 65. Logic gate status notations

Reading a logic diagram that does not provide information on the status of the gates is not any more difficult. It simply requires the reader to choose the initial conditions, determine the response of the circuits, and modify the inputs as needed.

Video 51. How to read an electrical diagram
http://www.youtube.com/watch?v=KbvM5Tkc-UA

c. Given a logic diagram, identify trip settings and trace the resulting actions should a trip occur.

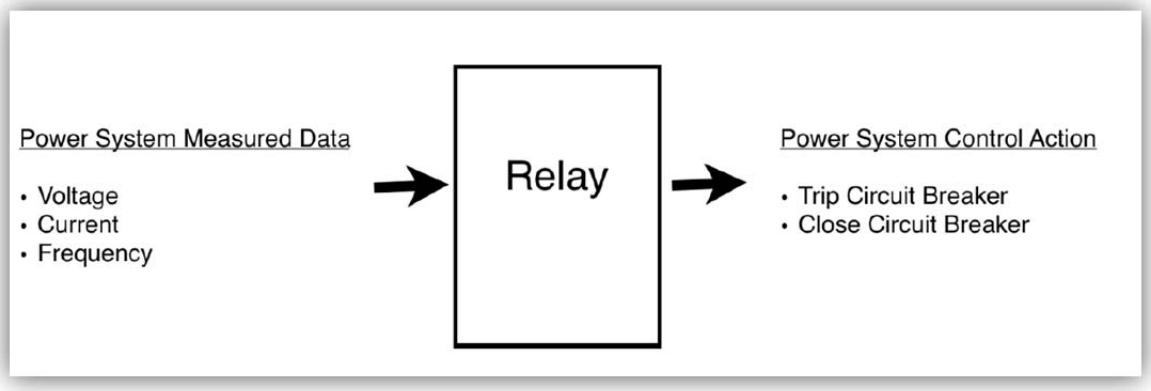
This KSA is performance-based. The Qualifying Official will evaluate its completion.

The following information from National Renewable Energy Laboratory/TP-550-46698, *Understanding Fault Characteristics of Inverter-Based Distributed Energy Resources*, may be helpful.

Protective relays are required on a distribution system to cause the quick removal from service of any electrical equipment associated with the power system when a short-circuit fault occurs or when the power system begins operating in abnormal conditions. Protective relays are essentially the brains that determine when the appropriate circuit breaker tripping action should take place. The mechanical device capable of disconnecting the faulty element

and physically isolating the electrical power system from short circuit disturbances is called a circuit breaker.

Figure 67 describes the protective relay input and output control procedure. The protective relay receives information about the electrical distribution system (voltage, current, and frequency) through current and voltage transformers. These transformers transform the measured voltage and current value to a more appropriate power level to be utilized by the protective relay. This information is processed by the protective relay and reacts to any abnormal conditions detected. Each protective relay needs to be set or programmed for the desired tripping time (i.e., time delay for relay coordination and system reliability purposes). The decision to trip open or to close the circuit breaker is made by the relay logic algorithms and must be programmed by a relay engineer.



Source: National Renewable Energy Laboratory/TP-550-46698

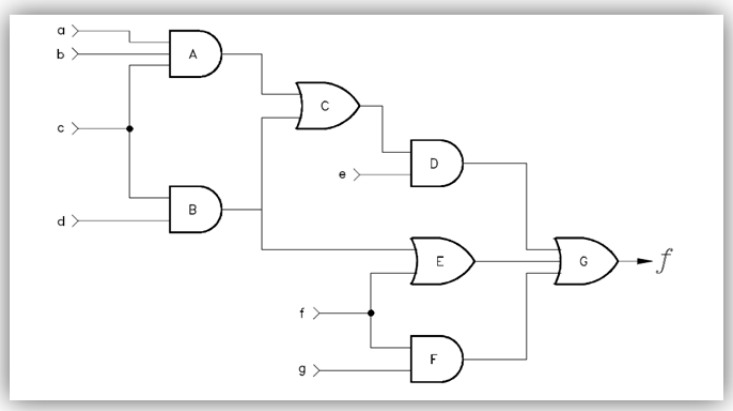
Figure 66. Input and output control of a protective relay

Video 52. How a circuit breaker works
http://www.youtube.com/watch?v=GSh0f94JwaA

d. Discuss the role of control logic diagrams in identifying failure modes and for mapping fault propagation through networks to support the identification of accident vulnerabilities.

The following is taken from DOE-HDBK-1016/1-93.

Logic diagrams and prints can be used to depict several types of information. The most common use is to provide a simplified functional representation of an electrical circuit, as illustrated in figure 68. For example, it is easier and faster to figure out how a valve functions and responds to various input signals by representing a valve circuit using logic symbols, than by using the electrical schematic with its complex relays and contacts. These drawings do not replace schematics, but they are easier to use for certain applications.



Source: DOE-HDBK-1016/1-93

Figure 67. Example of a logic print

12. Nuclear safety specialists must demonstrate a working knowledge of radioactivity and transformation mechanisms.

a. Define "activation."

The following is taken from DOE-HDBK-1108-2002.

Activation is the process of inducing radioactivity by irradiation. The process is commonly referred to as radioactivation or simply activation. Generally, energies above 10 MeV are needed to activate materials for particles other than neutrons.

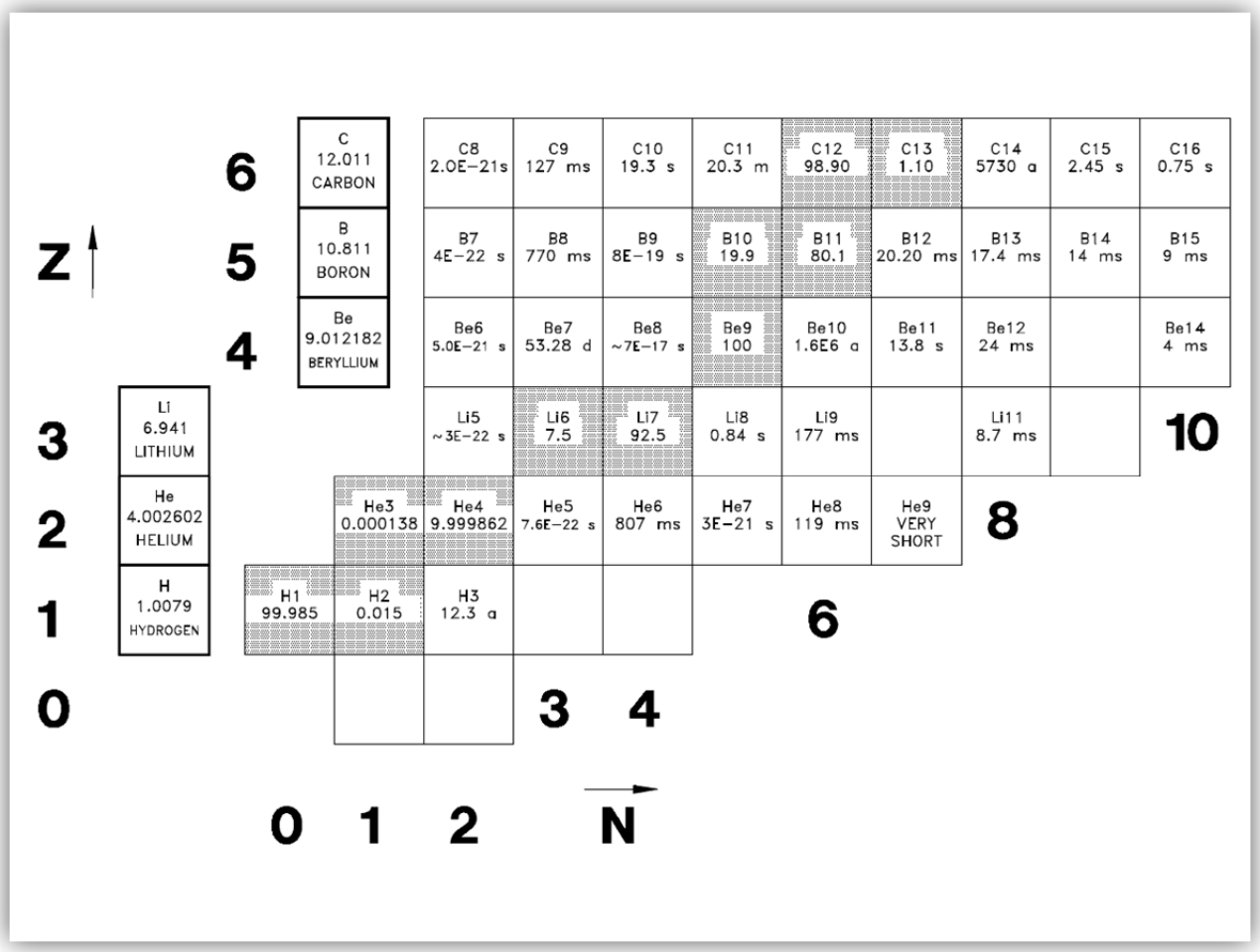
b. Given the "chart of nuclides," trace the decay chain for a specified nuclide.

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

The following information from DOE-HDBK-1019/1-93 may be helpful.

A tabulated chart called the chart of the nuclides lists the stable and unstable nuclides in addition to pertinent information about each one. Figure 69 shows a small portion of a typical chart. This chart plots a box for each individual nuclide, with the number of protons (Z) on the vertical axis and the number of neutrons (N = A–Z) on the horizontal axis.

The completely gray squares indicate stable isotopes. Those in white squares are artificially radioactive, meaning that they are produced by artificial techniques and do not occur naturally. By consulting a complete chart, other types of isotopes can be found, such as naturally occurring radioactive types (but none are found in the region of the chart that is illustrated in figure 69).

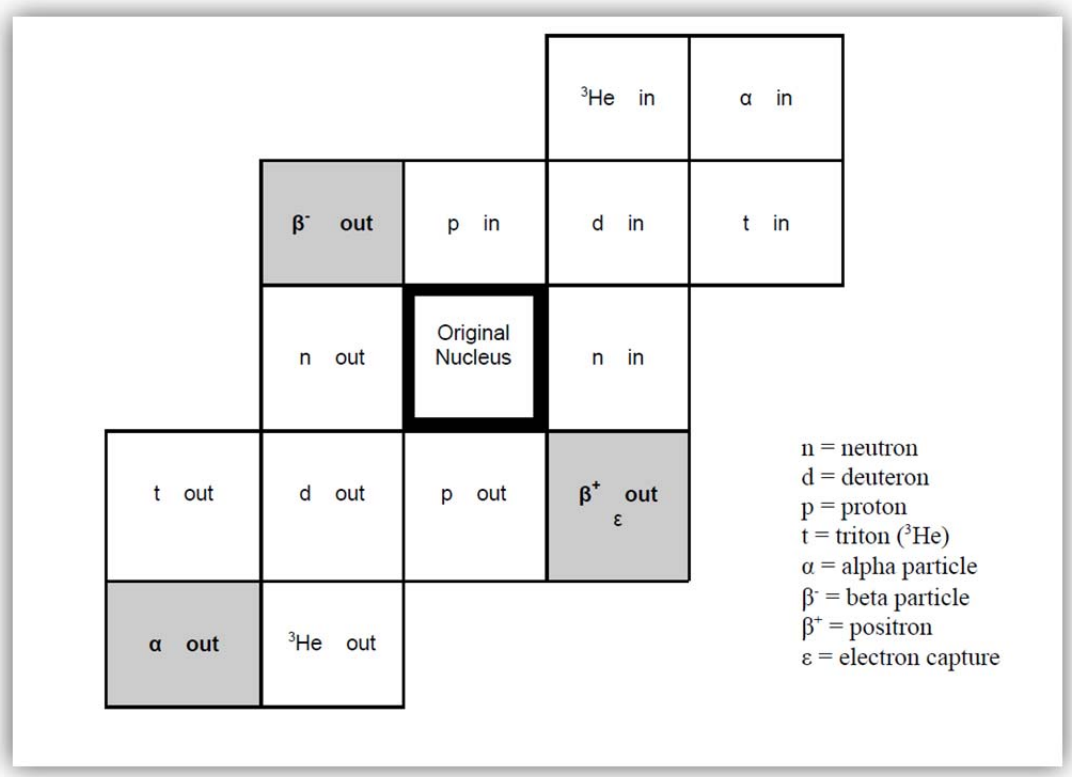


Source: DOE-HDBK-1019/1-93

Figure 68. Chart of the nuclides

Located in the box on the far left of each horizontal row is general information about the element. The box contains the chemical symbol of the element in addition to the average atomic weight of the naturally occurring substance and the average thermal neutron absorption cross section. The known isotopes (elements with the same atomic number Z but different mass number A) of each element are listed to the right.

As a result of decay, radionuclides shift from block to block within the chart of the nuclides. Figure 70 shows the relative locations of the products of various nuclear processes.



Source: DOE-HDBK-1122-99 (Archived)

Figure 69. Relative locations of the products of various nuclear processes

The following is taken from DOE-HDBK-1122-99 (Archived).

As can be seen in figure 70, the relative locations (displacements) of the primary modes of decay are listed:

- alpha (α) down 2, left 2
- beta (β-) up 1, left 1
- positron (β+)/EC down 1, right 1

Displacements can occur as a result of nuclear reactions brought about through bombarding given nuclides with various nuclear particles or gamma photons.

The chart of the nuclides provides considerable information about the behavior of nuclides. There is continuity in composition of the nuclides. For example, a line drawn through the stable nuclides forms a rather smooth curve extending from the lower left to the upper right corner of the chart.

Nuclides below this line are characterized by having an excess of neutrons and will, in general, be beta particle emitters.

Nuclides above this line are characterized by having an excess of protons and will, in general, decay by positron emission or electron capture.

Nuclides lying beyond the line of stability will, in general, demonstrate a tendency to seesaw between alpha decay and beta decay. All nuclides, if followed through their various decay schemes will eventually end in a gray box (stable isotope).

The chart presents in compact style much valuable information concerning the properties of the nuclides. These data include the following:

- Stable nuclides
  - Relative abundance
  - Cross section for activation
- Radioactive nuclides
  - Types of emissions
  - Energies of emissions
  - Half-life

Video 53. Introduction to the chart of nuclides
http://www.youtube.com/watch?v=IoE3AhgjGqc

c. Given either half-life or the radioactive decay constant, solve radioactive decay problems.

The Basic Decay Equation

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

The activity (A) of a sample is the rate of decay of that sample. This rate of decay is usually measured in the number of disintegrations that occur per second. For a sample containing millions of atoms, the activity is the product of the decay constant and the number of atoms present in the sample.

The relationship between the activity, number of atoms, and decay constant is shown in the following equation:

$$A = \lambda N$$

where:
A = activity of the nuclide (disintegrations/second)
Λ = decay constant of the nuclide (second^{-1})
N = number of atoms of the nuclide in the sample

Since λ is a constant, the activity and the number of atoms are always proportional.

Variation of Radioactivity Over Time

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

The rate where a given radionuclide sample decays is stated in the above equation as being equal to the product of the number of atoms and the decay constant. From this basic relationship it is possible to use calculus to derive an expression that can be used to calculate

how the number of atoms present will change over time. The derivation is beyond the scope of this text, but the following equation is the useful result:

$$N = N_o e^{-\lambda t}$$

where:
N = number of atoms present at time (t)
N_o = number of atoms initially present
λ = decay constant (time^{-1})
t = time

Since the activity and the number of atoms are always proportional, they may be used interchangeably to describe any given radionuclide population. Therefore, the following is true:

$$A = A_o e^{-\lambda t}$$

where:
A = activity present at time (t)
A_o = activity initially present
λ = decay constant (time^{-1})
t = time

The radioactive half-life is defined as the amount of time required for the activity to decrease to one-half of its original value. A relationship between the half-life and decay constant can be developed from the above equation.

d. Using the specific activity or decay constant of an isotope, convert between mass quantities and curies.

The following is taken from Cember, Herman and Thomas E. Johnson, *Introduction to Health Physics*, fourth edition.

The concentration of radioactivity, or the relationship between the mass of radioactive material and the activity, is called the specific activity (SA). SA is the number of Bqs (or curies) per unit mass or volume. The specific activity of a carrier-free radioisotope—a radioisotope that is not mixed with any other isotope of the same element—may be calculated as follows:

If λ is the transformation constant in units of reciprocal seconds, then the number of transformations per second and the number of Bqs in an aggregation of N atoms is simply given by λN.

If the radionuclide under consideration weighs one gram, then, the number of atoms is given by:

$$N = \frac{6.02 \times 10^{23} \text{ atoms/mol}}{A \frac{\text{g}}{\text{mol}}} \times W \text{ g},$$

Where A is the atomic weight of the nuclide. The activity per unit weight or the SA; therefore, is:

$$SA = \lambda N = \frac{\lambda \times 6.02 \times 10^{23}}{A} \frac{\text{Bq}}{\text{g}}.$$

Note that these equations are valid only if λ and *T* are given in time units of seconds. A more convenient form for calculating SA may be derived by making use of the fact that there are 3.7 X 10^{10} transformed per second in one gram of ^{226}Ra. The SA, therefore, of ^{226}Ra is 3.7 X 10^{10} Bq/g. The ratio of the SA of any radionuclide, SA_i, to that of ^{226}Ra is:

$$\frac{SA_i}{3.7 \times 10^{10} \frac{\text{Bq}}{\text{g}}} = \left(\frac{4.18 \times 10^{23} / A_i \times T_i}{4.18 \times 10^{23} / A_{\text{Ra}} \times T_{\text{Ra}}} \right)$$

$$SA_i = 3.7 \times 10^{10} \times \frac{A_{\text{Ra}} \times T_{\text{Ra}}}{A_i \times T_i} \frac{\text{Bq}}{\text{g}},$$

Where A_{Ra}, the atomic weight of ^{226}Ra, is 226, A_i is the atomic weight of the radioisotope whose SA is being calculated, and T_{Ra} and T_i are the half-lives of the radium and the radionuclide i. The only restriction is that both half-lives must be in the same units of time. Analogously, the SA in units of $\frac{Ci}{g}$ is given by:

$$SA_i = \frac{A_{\text{Ra}} \times T_{\text{Ra}}}{A_i \times T_i} \frac{\text{Ci}}{\text{g}}.$$

Video 54. Specific activity calculations
http://www.youtube.com/watch?v=aube1q4mCEI

13. Nuclear safety specialists must demonstrate a familiarity level knowledge of the biological effects of radiation.

a. Describe the effects of radiation exposure on the cellular level including:
- **Direct effects**
- **Indirect effects**

The following is taken from Dr. D.K. Maurya, *Types of Damages Induced After Radiation Exposure (Effect of Radiation)*.

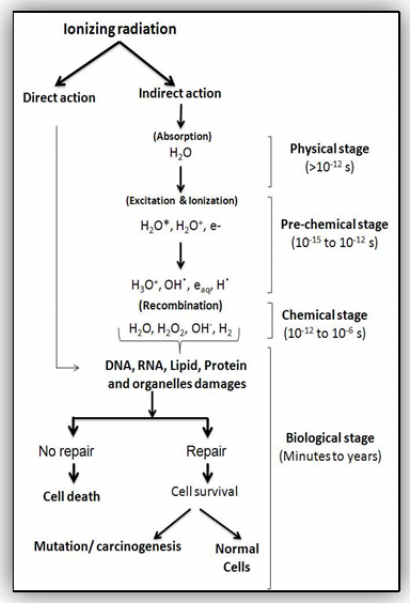
Radiation induced damage of the cell can be lethal (the cell dies), potentially lethal (cells can repair if allowed to remain in the stationary phase for some time after irradiation), and sub-lethal (the cell can repair itself). The effects of radiation include the following:

- Stochastic effects—Those that occur by chance and consist primarily of cancer and genetic effects. Stochastic effects are coincidental and cannot be avoided. They do not have a threshold. These effects can be divided into somatic and genetic. For stochastic effects, there is no threshold dose below which it is relatively certain that an adverse effect cannot occur. Examples: cancer, leukemia, genetic effects, and cataracts.

- Deterministic effects (non-stochastic effects)—Have a threshold of irradiation under which they do not appear and are the necessary consequence of irradiation. The damage they cause depends on the doses.

Exposure of living cells/tissues to ionizing radiation causes damages by transfer of energy to atoms and molecules in the cellular structure. Ionizing radiation causes either excitation or ionization or both to atoms and molecules. These excitations and ionizations can lead to the following events inside the cells/tissues:

- Generation of free radicals
- Breakage of chemical bonds
- Formation of new chemical bonds and cross-linkage between macromolecules
- Damage to biomolecules (e.g., DNA, RNA, lipids, proteins) that regulate vital cell processes



After radiation exposure, radiation may or may not interact with the critical target of the cell. It is considered that the chromosomes (DNA) are the most critical target of the cell since they contain the genetic information and instructions required for the cell to perform its function and to make copies of itself for reproduction purposes. Figure 71 illustrates this process.

Source: Dr. D.K. Maurya, Types of Damages Induced After Radiation Exposure (Effects of Radiation)

Figure 70. Effects of radiation: chain of the cellular events occurring in the cell/tissue after ionizing radiation exposure

The following information is taken from Universidad De Los Andes, Venezuela, *Biological Effects of Radiation*.

Indirect Action

Absorption of radiation energy may produce a chemical reaction called free radical formation. A free radical is a free atom or molecule carrying an unpaired orbital electron in the outer shell. An atom with an unpaired electron in the outer shell usually exhibits a high

degree of chemical reactivity. The two substances in a cell likely to be involved in free radical formation due to ionization are oxygen and water. The reactions are described by the following:

$$H_2O \leftrightarrow H^+ + OH^- \quad O_2 \leftrightarrow O^- + O^+$$

The hydroxyl radical (OH^-) is the major oxidizing agent resulting from ionization of water. Although free radicals are extremely reactive, most of the reactions recombine to form oxygen and water in about 10^{-5} seconds without causing any biological effects. However, biological effects may occur if these free radicals interact with other chemical compounds that diffuse far enough to then damage critical cell compounds. For example, free radicals may act as oxidizing or reducing agents and may form peroxides when they react with water; these may inactivate cellular mechanisms or interact with genetic material in the cell.

Direct Action

When the radiation energy is absorbed in the cell, it is possible that the radiation interacts directly with critical elements in the cell. The atoms of the target may be ionized or excited, initiating a chain of events that lead to biological change.

Direct action is the dominant process with high linear energy transfer (LET) radiation (e.g., particles, protons, and neutrons) primarily because the ionization track is very dense. In addition, direct action is associated with radiation effects for which a zero threshold dose is postulated (e.g., genetic effects). Damage may be transmitted to succeeding generations of cells, making the damage cumulative with the radiation dose.

Possible Cellular Effects

When ionizing radiation strikes the body, it randomly hits or misses millions of cells. For the cells that are not hit, the radiation simply passes through and no harm is done. If a cell is hit directly, the cell may be completely killed or, somewhat less likely, just damaged. Radiation safety is concerned with cellular effects that result in damage to crucial reproductive structures such as the chromosomes and their components (e.g., genes, DNA, etc.). Radiation can produce several different types of damage such as small physical displacement of molecules or the production of ion pairs. If the energy deposited within a cell is high enough, biological damage can occur. Some of the possible results from cellular radiation interactions include the following:

- Repair—The damaged cell can repair itself so no permanent damage is caused. This is the normal outcome for low doses of low LET radiation commonly encountered in the workplace.
- Cell death—The cell can die like millions of normal cells do naturally. The dead cell debris is carried away by the blood and a new cell is usually generated through normal biological processes to replace it.
- Mutate—In a very small number of events, a damaged cell may exhibit a change in the cell's reproductive structure allowing the cell to regenerate as a potentially pre-cancerous cell. Over a period of many years or decades, this may result in a full-blown, malignant cancer.

Video 55. Radiation safety: the basics
http://www.media-partners.com/hazardous_materials_training/radiation_safety_the_basics.htm

b. Describe the regulatory limits established by the Environmental Protection Agency (EPA) Federal guidance reports No. 11 and 13, DOE emergency exposure situations from 10 CFR 835, and EPA protective action guides for nuclear accidents.

Federal Guidance Report No. 11

Radiation protection programs for workers are based, in the U.S., on a hierarchy of limitations stemming from Federal guidance approved by the President. This guidance consists of principles, policies, and numerical primary guides, and is used by Federal agencies as the basis for developing and implementing their own regulatory standards.

The primary guides are usually expressed in terms of limiting doses to workers. The protection of workers against taking radioactive materials into the body however, is accomplished largely through the use of regulations based on derived guides expressed in terms of quantities or concentrations of radionuclides. The values of these derived guides are chosen so as to assure that workers in work environments that conform to them are unlikely to receive radiation doses that exceed the primary guides.

The purpose of Federal Guidance Report No. 11 is to set forth derived guides that are consistent with current Federal radiation protection guidance. They are intended to serve as the basis for regulations setting upper bounds on the inhalation and ingestion of, and submersion in, radioactive materials in the workplace. Federal Guidance Report No. 11 includes tables of exposure-to-date conversion factors, for general use in assessing average individual committed doses in any population that is adequately characterized by ICRP 23, *Report on the Task Group on Reference Man*.

Federal Guidance Report No. 13

Federal Guidance Report No. 13 is intended as the first of a series of documents that provide numerical factors, called "risk coefficients," for estimating risks to health from exposure to radionuclides. These reports apply state-of-the-art methods and models that take into account age and gender dependence of intake, metabolism, dosimetry, radiogenic risk, and competing causes of death in estimating the risks to health from internal or external exposure to radionuclides. Federal Guidance Report No. 13 provides tabulations of cancer risk coefficients for internal or external exposure to any of more than 800 radionuclides through various environmental media.

The risk coefficients developed in Federal Guidance Report No. 13 apply to an average member of the public, in the sense that estimates of risk are averaged over the age and gender distributions of a hypothetical closed stationary population whose survival functions and cancer mortality rates are based on recent data for the U.S. Specifically, the total mortality rates in this population are defined by the 1989–91 U.S. decennial life table and cancer mortality rates defined by U.S. cancer mortality data for the same period.

The EPA, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

The EPA has developed this manual to assist public officials in planning for emergency response to nuclear incidents. The EPA, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents* provides radiological protection criteria intended for

application to all nuclear incidents requiring consideration of protective actions other than nuclear war. It is designed for the use of those in Federal, state, and local government with responsibility for emergency response planning. This manual provides guidance for implementation of the criteria.

10 CFR 835, Occupational Radiation Protection

Except for planned special exposures conducted consistent with 10 CFR 835.204, "Planned Special Exposures," and emergency exposures authorized in accordance with 10 CFR 835.1302, "Emergency Exposure Situations," the occupational dose received by general employees shall be controlled such that the following limits are not exceeded in a year:

- A total effective dose of 5 rems (0.05 Sv)
- The sum of the equivalent dose to the whole body for external exposures and the committed equivalent dose to any organ or tissue other than the skin or the lens of the eye of 50 rems (0.5 Sv)
- An equivalent dose to the lens of the eye of 15 rems (0.15 Sv)
- The sum of the equivalent dose to the skin or to any extremity for external exposures and the committed equivalent dose to the skin or to any extremity of 50 rems (0.5 Sv)

All occupational doses received during the current year, except doses resulting from planned special exposures conducted in compliance with 10 CFR 835.204 and emergency exposures authorized in accordance with 10 CFR 835.1302, shall be included when demonstrating compliance with 10 CFR 835.202, "Occupational Dose Limits for General Employees," paragraph (a) and 10 CFR 835.207, "Occupational Dose Limits for Minors."

Doses from background, therapeutic, and diagnostic medical radiation, and participation as a subject in medical research programs shall not be included in dose records or in the assessment of compliance with the occupational dose limits.

c. Identify and discuss the range of doses above which one may expect acute radiation illness and early fatalities.

The following is taken from U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, Emergency Preparedness and Response, *Acute Radiation Syndrome: A Fact Sheet for Clinicians*.

Acute radiation syndrome (ARS) (sometimes known as radiation toxicity or radiation sickness) is an acute illness caused by irradiation of the entire body (or most of the body) by a high dose of penetrating radiation in a very short period of time (usually a matter of minutes). The major cause of this syndrome is depletion of immature parenchymal stem cells in specific tissues. Examples of people who suffered from ARS are the survivors of the Hiroshima and Nagasaki atomic bombs, the firefighters that first responded after the Chernobyl Nuclear Power Plant event in 1986, and some unintentional exposures to sterilization irradiators.

The required conditions for ARS are as follows:

- The radiation dose must be large (i.e., greater than 0.7 Gy or 70 rad).
  - Mild symptoms may be observed with doses as low as 0.3 Gy or 30 rad.

- The dose usually must be external (i.e., the source of radiation is outside of the patient's body).
  - Radioactive materials deposited inside the body have produced some ARS effects only in extremely rare cases.
- The radiation must be penetrating (i.e., able to reach the internal organs).
  - High energy X-rays, gamma rays, and neutrons are penetrating radiations.
- The entire body (or a significant portion of it) must have received the dose.
  - Most radiation injuries are local, frequently involving the hands, and these local injuries seldom cause classical signs of ARS.
- The dose must have been delivered in a short time (usually a matter of minutes).
  - Fractionated doses are often used in radiation therapy. These are large total doses delivered in small daily amounts over a period of time. Fractionated doses are less effective at inducing ARS than a single dose of the same magnitude.

The three classic ARS syndromes are listed:

1. Bone marrow syndrome (sometimes referred to as hematopoietic syndrome)—The full syndrome will usually occur with a dose between 0.7 and 10 Gy (70–100 rad) though mild symptoms may occur as low as 0.3 Gy or 30 rad.
   - The survival rate of patients with this syndrome decreases with increasing dose. The primary cause of death is the destruction of the bone marrow, resulting in infection and hemorrhage.
2. GI syndrome—The full syndrome will usually occur with a dose greater than approximately 10 Gy (1,000 rad) although some symptoms may occur as low as 6 Gy or 600 rad.
   - Survival is extremely unlikely with this syndrome. Destructive and irreparable changes in the GI tract and bone marrow usually cause infection, dehydration, and electrolyte imbalance. Death usually occurs within two weeks.
3. Cardiovascular/central nervous system syndrome—The full syndrome will usually occur with a dose greater than approximately 50 Gy (5,000 rad) although some symptoms may occur as low as 20 Gy or 2,000 rad.
   - Death occurs within three days. Death likely is due to collapse of the circulatory system as well as increased pressure in the confining cranial vault as the result of increased fluid content caused by edema, vasculitis, and meningitis.

The following is taken from Cember, Herman and Thomas E. Johnson, *Introduction to Health Physics*, fourth edition.

As a group, the effects caused by acute doses are called deterministic. This means that the severity of the effect is determined by the amount of dose received. Deterministic effects usually have some threshold level below which the effect will probably not occur, but above which the effect is expected. When the dose is above the threshold, the severity of the effect will increase with increasing absorbed dose.

Acute whole body doses in the range of greater than 400–450 rad may result in the statistical expectation that 50 percent of the population exposed will die within 60 days without medical treatment. This is the LD50/60.

Other acute effects:

- 200 to 300 rad to the skin—erythema (reddening of the skin)
- 125 to 200 rad to the ovaries—prolonged to permanent suppression of menstruation in about 50 percent of women
- 600 rad to the ovaries or testicles—permanent sterilization
- 50 rad to the thyroid—benign (non-cancerous) tumors

14. Nuclear safety specialists must demonstrate a familiarity level knowledge of the principles and use of radiological instrumentation and radiological monitoring/survey practices.

a. Discuss the purpose, principles of detection, and operation, and field application of the following:

- **Continuous air monitors, including tritium alarms**
- **Area radiation monitors**
- **Criticality detection/alarm systems**
- **Process radiation monitors**

Continuous Air Monitors (CAM), Including Tritium Alarms

The following is taken from DOE-STD-1128-2013.

CAMs are used extensively in plutonium facilities. CAMs and sample extraction lines that go to CAMs and continuous radiation dose monitors should be placed outside the gloveboxes and hoods. In-line processing instrumentation is critical to accurately monitor the work stations and a review should be performed to determine instrumentation locations. CAMs may not have adequate detection capabilities for real-time monitoring at the DAC level. For ^{239}Pu, the ALI is 12 nCi for absorption type M compounds based on the DAC of 5 x 10^{-12} µCi/ml, as given in 10 CFR 835, Appendix A, "Derived Air Concentrations (DAC) for Workers From External Exposure During Immersion in a cloud of Airborne Radioactive Material." DOE G 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835, Occupational Radiation Protection*, chapter 1, "Purpose and Applicability," recommends that real-time air monitors be capable of measuring one DAC when averaged over eight hours under laboratory conditions. Alarm set points for real-time air monitors used for routine monitoring should be set at the lowest practical level so as to accurately indicate loss of containment or the need for corrective action without causing a significant number of false alarms. When monitoring for alpha emitters in areas with high radon concentrations an alarm set point greater than eight DAC-hours may be necessary.

CAMs typically have had poor large-particle response due to particle loss during transport to the filter inside the system. Newer alpha air monitors are able to handle large particles more efficiently. Background levels of radon-thoron decay products may be present in concentrations up to 50 to 100 times greater than the level of plutonium of interest. If calibrated properly, alpha CAMs will subtract background levels of radon-thoron decay products; however, in practice the detection limit for plutonium may be as high as forty DAC-hour in the presence of high radon levels. A new generation of alpha CAMs is able to compensate for radon more effectively and meet the desired eight DAC-hour alarm level.

CAMs should meet the following criteria according to ANSI N317, *American National Standard Performance Criteria for Instrumentation Used for Inplant Plutonium Monitoring*. The primary purpose of any CAM is to detect the presence of airborne radioactivity and activate an alarm to warn personnel in the area so that actions can be taken to minimize personnel exposures. The goal for any CAM should be to perform this function as quickly as possible and at the lowest detectable level of radioactive airborne concentration. The quantity of airborne radioactivity that will result in an alarm within a given time interval is defined in units of DAC-hour for a particular radionuclide and is a function of the nuclide's airborne concentration in DACs, the sampling rate, the lower limit of detection of the instrument, and the time needed for the alarm to occur.

For plutonium facilities, air sampling and monitoring are essential elements of the radiological control program. Real-time air monitoring using alpha-sensitive CAMs should be used to alert workers to rapid degradation of radiological conditions. The air sampling system with a lower limit of detection will be adequate to provide continuing assurance that personnel exposures are within limits and ALARA.

The characteristics of a good plutonium CAM include the following:

- A lower limit of detection equal to, or better than, eight DAC-hour
- High reliability with a minimum of spurious alarms
- A stable and constant flow air mover
- Stable and documented detector efficiency with geometry, filter collection efficiency, self-attenuation, etc., considered
- Methodology for radiation discrimination and natural radioactivity discrimination
- System for activating an alarm
- Shielding for extraneous sources of interference such as radiation, radio frequency, temperature, and vibration
- Mechanical and electrical ruggedness
- Ease of maintenance and calibration

The following is taken from DOE-HDBK-1129-2007 (Archived).

TRITIUM MONITORING SYSTEMS

Several different types of instruments may be used to detect and measure tritium in the operation of a facility. These instruments include the following:

- Portable room air monitors—There are several handheld portable room air monitors on the market and their capabilities and ranges vary as a function of the different manufacturer and the purpose for which they were designed. It is convenient in some activities to have the capability to connect a small hose to the monitor so that it may be used to detect tritium leaks around equipment.
- Fixed station room air monitors—Are designed to be installed in fixed locations and to be used to monitor the room air tritium concentrations. Depending on the manufacturer, they may have several ranges and are equipped with one or two alarm set points and audible as well as visual alarms.
- Glovebox atmosphere monitors—May be open mesh or closed ionization chambers and are designed to monitor the higher levels of tritium inside the glovebox containment systems.

- Hood and exhaust duct, air monitors—Are similar to fixed station monitors in range and characteristics.
- Exhaust stacks, air monitors—Are similar to fixed station air monitors except that they generally have larger ionization chambers to increase the sensitivity of the monitor.
- Personnel friskers and breath analyzers—There has been some interest in instruments that can be used to frisk personnel as they enter and exit tritium-contamination areas. One DOE facility implemented a process of personnel frisking consisting of the use of skin surface wipes counted in a liquid scintillation counter upon entry and exit from tritium-contaminated areas. In another facility, a hand station based on counting the associated gas flow across the hands was used. To date, the development work required to convert measurements made by these techniques to dose or worker exposure has not been completed. It is expected that differences in the body chemistry of personnel and differences in the time delay between tritium exposure and equilibration of tritium in the body will continue to make the results of skin surface contamination measurement and breathe analysis monitoring inconsistent. The impact of false alarms and inconsistent results on worker confidence will probably continue to make these systems unsatisfactory for worker monitoring.

Area Radiation Monitors (ARM)

The following is taken from DOE G 441.1-1C.

10 CFR 835 defines radiation and high radiation areas in terms of the radiation levels at a distance of 30 centimeters from the source or from any surface penetrated by the radiation. Similarly, 10 CFR 835 defines very high radiation areas in terms of the radiation levels at a distance of 100 centimeters. Therefore, area radiation monitoring should be performed at these distances to ensure compliance with the 10 CFR 835.603, "Radiological Areas and Radioactive Material Areas," area posting requirements. However, actual and likely exposure conditions should be considered when performing monitoring for task planning, HA, or dose assessment. If an individual is likely to linger at a distance of several feet from a shield wall, use an obvious travel path between stations, or work within a few inches of a radiation source, measurements should be made at those locations to provide representative information. Such monitoring should be performed as necessary to ensure compliance with 10 CFR 835.401, "General Requirements," paragraph (a). Methods used in performing area radiation monitoring should be adequate to identify localized variations in radiation levels to facilitate dosimeter placement and individual exposure reduction actions.

Important variables that should be considered for inclusion in procedures and training include instrument selection, operation, functional testing, detector orientation, response time, operational limitations, source-to-detector distance considerations, and documentation requirements.

The following is taken from DOE-STD-1098-2008.

Area Radiation Monitors:

- In addition to the requirements and recommendations of DOE-STD-1098-2008, article 551, "General Provisions," ARMs (not to include area monitoring dosimeters)

should be installed in frequently occupied locations with the potential for unexpected increases in dose rates and in remote locations where there is a need for local indication of dose rates prior to personnel entry.

- ARMs should not be substituted for radiation exposure surveys in characterizing a workplace.
- The need for, and placement of, ARMs should be documented and assessed when changes to facilities, systems, or equipment occur.
- In addition to the requirements of DOE-STD-1098-2008, article 562, "Inspection, Calibration, and Performance Tests," ARMs should be tested periodically to verify audible alarm system operability and audibility under ambient working conditions and operability of visual alarms when so equipped.
- If installed instrumentation is removed from service for maintenance or calibration, a radiation monitoring program providing similar detection capability should be provided, consistent with the potential for unexpected increases in radiation dose rates.
- Where an ARM is incorporated in a safety interlock system, the circuitry should be such that a failure of the monitor either prevents entry into the area or prevents operation of the radiation producing device. If the circuitry is required to ensure compliance with the high radiation area access control requirements of 10 CFR 835.502, "High and Very High Radiation Areas," then the circuitry shall be fail-safe.

Criticality Detection/Alarm Systems

The following is taken from DOE-STD-1128-2008 (Archived).

As specified in ANSI/ANS-8.3 the need for criticality alarm systems will be evaluated for all activities where the inventory of fissionable material in individual unrelated work areas exceeds 700 g of ^{235}U, 520 g of ^{233}U, 450 g of ^{239}Pu, or 450 g of any combination of these three isotopes.

- If the fissionable material mass exceeds the ANSI/ANS-8.3 limits and the probability of criticality is greater than 10^{-6} per year, a criticality alarm system will be provided to cover occupied areas where the expected dose exceeds 12 rad in free air. Nuclear accident dosimetry will be provided, as required by 10 CFR 835.1304, "Nuclear Accident Dosimetry." The criticality alarm system should include a criticality detection device and a personnel evacuation alarm.

Note: In what follows, 10^{-6} per year is used as a measure of credibility, and does not mean that a PRA has to be performed. Reasonable grounds for incredibility may be presented on the basis of commonly accepted engineering judgment.

- If the fissionable material mass exceeds the ANSI/ANS-8.3 limits and the probability of criticality is greater than 10^{-6} per year, but there are no occupied areas where the expected dose exceeds 12 rad in free air, then only a criticality detector system is needed.
- If the fissionable material mass exceeds the ANSI/ANS-8.3 limits, but a criticality accident is determined to be impossible or less than 10^{-6} per year, then neither a criticality alarm system nor nuclear accident dosimetry is needed.

The alarm signal will be for immediate evacuation purposes only and of sufficient volume and coverage to be heard in all areas that are to be evacuated. Information on sound levels of the alarm can be found in ANSI/ANS-8.3. The alarm trip point will be set low enough to detect the MAC. The MAC may be assumed to deliver the equivalent of an absorbed dose in free air of 20 rad at a distance of 2 m from the reacting material within 60 seconds. The alarm signal will activate promptly when the dose rate at the detectors equals, or exceeds, a value equivalent to 20 rad/min at 2 m from the reacting material. A visible or audible warning signal will be provided at a normally occupied location to indicate system malfunction or loss of primary power. Each alarm system should be tested at least once every three months. An evacuation drill will be conducted at least annually.

Criticality alarm systems may consist of one to several detectors per unit. In multi-detector units at least two detectors will be at the alarm level before initiating the alarm; in redundant systems, failure of any single channel will be into the trip state.

Process Radiation Monitors

The following is taken from U.S. Nuclear Regulatory Commission, Human Resource Training Development, *Westinghouse Technology Systems Manual*, section 16.0, "Radiation Monitoring System."

The process radiation monitoring (PRM) system monitors the radiation level of various process liquid and gas streams that may serve as discharge routes for radioactive materials. These monitors are provided to indicate the radioactivity of the process stream and to alert operating personnel when operational limits are approached for the normal release of radioactive material to the environment.

On process streams that do not discharge to the environs, such as the component cooling water system, process monitors are provided to indicate process stream malfunctions. This is accomplished by detecting the normal background radiation of the system and by alerting the operator with an annunciator if an accumulation of radioactive material occurs in the system. In addition to providing continuous indication and alarms, the PRM may provide various automatic functions, such as the closing of the vent valves, discharge valves, etc.

If the activity level in the process stream reaches a predetermined set point, the system will perform its automatic function that ensures that the discharge of radioactive material to the environs is limited. The PRM system can monitor a process stream with one of two types of monitors: the in-line and off-line monitors.

IN-LINE MONITOR

An in-line monitor has the detector probe directly immersed in the process stream. The advantage of this type of monitoring is that the detector probe will be provided a representative sample of the process and responds rapidly to activity changes. The disadvantages of this monitor are that if the process has a turbulent flow the detector probe must be protected by placing it in a well that lowers the sensitivity of the probe, and if the probe fails, the system must either be secured or a means of bypassing flow around the probe must be provided.

OFF-LINE MONITOR

An off-line monitor contains piping, valves, detector probes (usually two in parallel), and a motive force device, such as a pump or a fan. This monitor will take suction on the process stream, pass the flow to the detector, and then return the sample to the process stream. The advantages of this monitor are that with lower flow rates, the detector probe can be directly immersed in the sample stream without a protection well; therefore, increasing probe sensitivity, and if a detector fails, it can be isolated, and the process stream is not affected.

The disadvantages of this monitor are that it may not be receiving a representative sample of the process stream and may not be as responsive to rapid changes of activity in the process.

The following is taken from DOE/EH-0173T.

Liquid effluents from DOE-controlled facilities that have the potential for radioactive contamination should be monitored in accordance with the requirements of DOE Orders and standards to show the following:

- Effluent monitoring extraction locations used for providing quantitative effluent release data for each outfall
- Procedures and equipment used to perform the extraction and measurement
- Frequency and analyses required for each extraction location
- Minimum detection level and accuracy
- QA components
- Effluent outfall alarm settings and bases

For those effluent streams requiring continuous monitoring/sampling, all data received from the continuous monitoring system should be used when performing statistical analyses. In the case of discharge points releasing radionuclides emitting alpha or weak beta radiation, with no documentable ratios to beta and/or gamma emitters that could be used as indicator radionuclides, continuous proportional sampling and analysis can be used as an alternative to continuous monitoring. However, the consideration of new technologies to continuously monitor such effluent streams is encouraged.

Reliable quantification of radionuclides in liquid effluent streams requires representative sampling that in turn requires consideration of stream flow rate and variability, sample port and collector design, delivery system reliability, effluent-stream chemical and biological characteristics, and the need for sample preservation. Useful advice on representative liquid sampling is available from the American Public Health Association and the American Society for Testing and Materials.

The following criteria should be considered when operating a liquid effluent sampling system:

- Location of sampling and monitoring systems
- Use of a pump in areas where necessary to provide a uniform continuous flow in the main sample line
- A redundant sample-collection system or one of the following alternatives to permit continued sampling during replacement or servicing of the system:
    - A substitute sample-transport system

    - The capability to shut down the system for fast repair
    - An alternate method for estimating releases when the system is not capable of operating
- Location of sample ports in liquid effluent lines sufficiently far downstream from the last feeder line to allow complete mixing of liquid and design of the sample port to allow intake of a proportional part of the liquid effluent stream
- Capability to determine the effluent stream and sample-line flows within an accuracy of at least ±10 percent
- Design of the system to minimize deformation and sedimentation and to prevent freezing of effluent sample lines

Design considerations for liquid effluent monitoring systems should include the purpose of the monitoring, the types and levels of expected radionuclides, potential background dose rates, expected duration of releases, and environmental effects. One of the primary purposes of using a monitoring system is to utilize its ability to provide a prompt signal if a significant release occurs. Thus, the output signal from monitoring systems should be continuously monitored by responsible personnel. In addition, written responses procedures should be provided describing the action that responsible personnel must take if an abnormal signal is detected. The output signal instrumentation, monitoring system recorders, and alarms should be in a location that is continuously occupied by operations or security personnel.

The following general design criteria should be considered in the design and operation of routine liquid effluent monitoring systems:

- If off-line monitoring is employed
    - use criteria in DOE/EH—0173T, *Environmental Regulatory Guide for Radiological Effluent Monitoring and Environmental Surveillance*, section 2.3, "Sampling System Design Criteria," for sample transport;
    - use criteria in DOE/EH—0173T, section 2.3.8, "Environmental Considerations," for environmental protection, maintenance, and modification;
    - use characterization study data for radionuclide measurements, including ratios of radionuclides not directly measurable, if present;
    - use adequate shielding for detector operation and to maintain personnel exposure ALARA;
    - use a predefined alarm level that is just above normal variations in release levels;
    - locate alarm annunciators in normally occupied locations; and
    - use stable electric power sources to provide uniform voltage to the monitor and alarm systems.
- If in-line monitoring is employed
    - use the criteria for off-line monitoring; and
    - use interpretive curves that allow quick conversion of dose rates or count rates to radionuclide release rates, such that concentrations of and curies released by the various radionuclides can be estimated.

To signal the need for corrective actions that may be necessary to prevent public or environmental exposures from exceeding the limits or recommendations given in DOE Orders, when continuous monitoring systems are required, they should have alarms set to

provide timely warnings. To prevent the cumulative impacts of small releases from producing a significant impact, routine grab, continuous, or proportional samples should be collected often enough to detect radionuclides of interest including those with relatively short half-lives.

REGULATORY

15. Nuclear safety specialists must demonstrate a working level knowledge of 10 CFR 830.204, "Documented Safety Analysis," and DOE Guide (G) 421.1-2, *Implementation Guide for Use in Developing Safety Analyses to Meet Subpart B of 10 CFR 830*, with respect to their impact on the Department's nuclear safety.

a. Discuss the basic purposes and objectives of a DSA.

The following is taken from 10 CFR 830.3.

A DSA is a documented analysis of the extent to which a nuclear facility can be operated safely with respect to workers, the public, and the environment, including a description of the conditions, safe boundaries, and hazard controls that provide the basis for ensuring safety.

The following is taken from DOE G 421.1-2A.

The DSA for a DOE hazard category 1, 2, or 3 nuclear facility in accordance with 10 CFR 830.204 must, as appropriate for the complexities and hazards associated with the facility or activity

- describe the facility, activities, and operations (including the design of safety SSCs 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 that 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 them 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 criticality safety 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.

b. Describe the responsibilities of contractors for the development and maintenance of a DSA.

The following was taken from DOE G 421.1-2A.

In accordance with 10 CFR 830.202, "Safety Basis," contractors must ensure that information in a DSA is current and applicable. The safety basis rule applies to all facilities that satisfy the criteria for category 3 or higher hazard nuclear facilities except those specifically excluded in 10 CFR 830.2, "Exclusions." Therefore, when a facility changes status, say from a production or mission-oriented status to inactive, transition surveillance and maintenance, deactivation activities, or decommissioning, the DSA and TSR associated with the facility or activity needs to be updated to describe the activities, consider the hazards associated with the new status, and the controls associated with these hazards. Any facility or activity DSA that does not reflect its current status is out of compliance with the safety basis rule. The annual update required by the rule applies to all DSAs. DOE contractors and DOE remain accountable for safety during the period those DSAs are being upgraded.

The USQ requirements (10 CFR 830.203, "Unreviewed Safety Question Process") have a primary role in preserving the DOE safety basis for each nuclear facility. The concept of the USQ allows contractors to make physical and procedural changes and to conduct tests and experiments without prior DOE approval, as long as these changes do not affect the safety basis of the facility.

When a facility does not change status, but does have changes that affect the safety basis, the DSA and TSR should be updated to reflect those changes. Usually the changes will be the subject of a USQ determination. If there are no changes, notifying DOE of that fact is sufficient for the update. The rule is silent on a cutoff date for changes to the facility to be included in a DSA update. This can be determined on an ad hoc basis but should be compatible with the annual report on USQ determinations (see 10 CFR 830.203). The USQ determinations and associated safety analyses as well as supporting safety analyses for any DOE-approved changes to a facility are considered part of the safety basis until incorporated in an annual update.

The contractor responsible for a facility can provide annual DSA updates by

- stating in a letter to DOE that the existing DSA remains fully applicable;
- providing supplements or amendments to make the DSA current, to DOE approval; or
- submitting, for DOE approval, a DSA that is proposed to supersede the current DSA.

Generally, depending on the complexity of the facility, it may be impractical to incorporate the most recent USQ determinations and facility changes in the DSA annual update. However, at least those implemented six months or more before the submittal of the annual update should be included. In addition, 10 CFR 830.203(f) requires that the contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must annually submit to DOE a summary of the USQ determinations performed since the prior submission.

Consistent with the integrated safety management (ISM) requirements for feedback specified in the DOE Acquisition Regulation clause 48 CFR 970.5223-1, "Integration of Environment Safety and Health into Work Planning and Execution," DOE expects that updates of DSA for facilities in operation for one year or more will address the results of the experience feedback program for that facility.

c. Define the following terms and discuss the purpose of each:

- **Design basis**
- **Safety analysis**
- **Safety basis**
- **Basis for interim operation**
- **Transportation safety document**
- **SAR for packaging**
- **Health and safety plan**
- **Hazards analysis report**

Design Basis

The following is taken from DOE-STD-3009-94, Chg 3.

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.

Safety Analysis

The following is taken from DOE-STD-3009-94, Chg 3.

The safety analysis is 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 following is taken from DOE-STD-3009-94, Chg 3.

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.

BIO

The following is taken from DOE-STD-3011-2002.

The BIO is an acceptable form of DSA in accordance with 10 CFR 830, subpart B, "Safety Basis Requirements," appendix A, "General Statement of Safety Basis Policy."

The DSA (BIO) must include the following:

- Facility categorization according to DOE-STD-1027-92, Chg 1
- A description of the facility (including the work to be performed)
- A systematic identification of hazards associated with the facility

- Evaluation or normal, abnormal, and accident conditions (including potential natural phenomenon hazards that might be associated with long-term status) that might be associated with the generation or release of radioactive or other hazardous materials, including consideration of the need for analysis of beyond DBA
- Derivation and classification of hazard controls necessary to protect workers, the public, and the environment
- Definition of the characteristics of safety management programs necessary to ensure safe operation, including criticality safety, when criticality hazards exist

Transportation Safety Document

The following is taken from DOE-HDBK-1129-99 (Archived).

The transportation safety document describes the methodology and compliance process to meet equivalent safety for any deviation from the hazardous materials regulations.

SAR for Packaging

The following is taken from DOE O 461.1B.

The SAR for packaging (SARP) is the document that conforms to NRC Regulatory Guide 7.9, revision 2, *Standard Format and Contents of Part 71 Applications for Approval of Packages for Radioactive Materials*, and provides a comprehensive technical evaluation of a package. The SARP consists of sections containing general information; structural, thermal, containment, shielding and criticality evaluations; operating procedures; acceptance tests, and maintenance and QAPs. The purpose of the SARP is to demonstrate conformity with the applicable sections of 10 CFR 71, "Packaging and Transportation of Radioactive Material," and 49 CFR 171, "General Information, Regulations, and Definitions," through 49 CFR 180, "Continuing Qualification and Maintenance of Packagings."

Health and Safety Plan

The following is taken from DOE-EM-5503-94 (Archived).

A high-quality health and safety plan (HASP) must provide the following:

- A clear chain of command for safety and health activities
- Accountability for safety and health performance
- Well-defined headquarters expectations regarding safety and health
- Well-defined task and operational hazards/risks
- Comprehensive hazard prevention and control methods
- Recordkeeping requirements to track program progress

Hazards Analysis Report (HAR)

The following is taken from DOE O 461.1B.

The HAR is a document submitted to NNSA to support an applicant's request for an authorization for offsite transport of noncompliant packages and/or shipping configurations that contain less than a type B quantity of radioactive material or no radioactive materials, but which contain regulated hazardous materials as defined in 49 CFR 171-180. The HAR identifies the type and quantity of hazardous material, proposed packaging/handling gear, mode of transportation, shipment destinations, tie-down procedures, tests that will be

performed on the unit, the post-test status of hazardous components and procedures to verify that the shipment can be conducted safely (if applicable).

d. Discuss the six items a DSA for a hazard category 1, 2, or 3 DOE nuclear facility must address:
- **Description of facility and work to be performed**
- **Systematic identification of natural and man-made hazards**
- **Evaluation of normal, abnormal, and accident conditions**
- **Derivation of hazard controls, demonstration of the adequacy of the controls, and description of how the controls are maintained**
- **Definition of the characteristics of safety management programs**
- **Definition of criticality safety program, when required**

The following is taken from DOE-STD-3009-94, Chg 3.

Description of Facility and Work to be Performed

Chapter 2 of a DSA provides descriptions of the facility and processes to support assumptions used in the hazard and accident analyses. These descriptions focus on all major facility features necessary to understand the HA and accident analysis, not just safety SSCs. Expected products of this chapter, as applicable based on the graded approach, include the following:
- Overview of the facility, its inputs and its outputs, including mission and history
- Description of the facility structure and design basis
- Description of the facility process systems and constituent components, instrumentation, controls, operating parameters, and relationships of SSCs
- Description of confinement systems
- Description of the facility safety support systems
- Description of the facility utilities
- Description of facility auxiliary systems and support systems

Existing supporting documentation is to be referenced. Include brief abstracts of referenced documentation with enough of the salient facts to provide an understanding of the referenced documentation and its relation to this chapter.

The development of this chapter for hazard category 2 and 3 facilities is an iterative process dependent on the development of the hazard and accident analyses. The facility description should provide a model of the facility that would allow an independent reader to develop an understanding of facility operations and an appreciation of facility structure and operations without extensive consultation of controlled references. The level of detail required in the facility description is based on the significance of preventive and mitigative features identified and the degree of facility context necessary to understand the analyses. For a hazard category 3 facility, provide a brief description of the facility, processes, and major SSCs. Grading will be based predominantly on complexity.

Systematic Identification of Natural and Man-Made Hazards

The purpose of chapter 3 of the DSA is to provide information that will satisfy the requirements of 10 CFR 830, “Nuclear Safety Management,” to evaluate normal, abnormal, and accident conditions, including consideration of natural and man-made external events,

identification of energy sources or process that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents that may be beyond the design basis of the facility.

This chapter describes the process used to systematically identify and assess hazards to evaluate the potential internal, man-made external, and natural events that can cause the identified hazards to develop into accidents. This chapter presents the results of this hazard identification and assessment process. HA considers the complete spectrum of accidents that may occur due to facility operations; analyzes potential accident consequences to the public and workers; estimates likelihood of occurrence; identifies and assesses associated preventive and mitigative features; identifies SS SSCs; and identifies a selected subset of accidents, designated DBAs, to be formally defined in accident analysis. Subsequent accident analysis evaluates these DBAs for comparison with the EG. This chapter covers the topics of hazard identification, facility hazard categorization, hazard evaluation, and accident analysis.

Expected products of this chapter, as applicable based on the graded approach, include the following:

- Description of the methodology for, and approach to, hazard and accident analyses.
- Identification of hazardous materials and energy sources present by type, quantity, form, and location.
- Facility hazard categorization, including segmentation in accordance with DOE-STD-1027-92, Chg 1.
- Identification in the HA of the spectrum of potential accidents at the facility in terms of largely qualitative consequences and frequency estimates. The summary of this activity will include the following:
  - Identification of planned design and operational safety improvements
  - Summary of DID, including identification of SS SSCs, SACs, and other items needing TSR coverage in accordance with 10 CFR 830
  - Summary of the significant worker safety features, including identification of SS SSCs and any relevant programs to be covered under TSR and ACs, including those controls designated as SACs
  - Summary of design and operational features that reduces the potential for large material releases to the environment
  - Identification of the limited set of unique and representative accidents to be assessed further in accident analysis
- Accident analysis of DBAs identified in the HA. The summary of this activity will include for each accident analyzed, the following:
  - Estimation of source term and consequence
  - Documentation of the rationale for binning frequency of occurrence in a broad range in HA (detailed probability calculations not required)
  - Documentation of accident assumptions and identification of SC SSCs based on the EG

Evaluation of Normal, Abnormal, and Accident Conditions

The principal purpose of the accident analysis is to identify any SC SSCs, SACs, and TSRs needed for protection of the public. Each accident sequence needs to be analyzed through the use of a documented, deterministic, DBA. Whenever possible, DBAs are analyzed using the

simplest applicable deterministic, phenomenological calculations (e.g., pressure estimates from a simple ideal gas law calculation, hand-calculated Gaussian plume dispersions). The nondeterministic aspects of DBA analysis are simplified by estimating overall sequence frequencies in broad frequency ranges in HA. This process is considered sufficient for DSA purposes and accident analysis need only document the basis for the binning performed in HA. Detailed probabilistic calculations are neither expected nor required.

Natural evens and man-made external events are special cases. Natural event DBAs are those events with a phenomenon initiating frequency. External events are not typically design bases for facilities. However, they will be referred to as DBAs and analyzed as such if frequency of occurrence is estimated to exceed 10^{-6}/year conservatively calculated, or 10^{-7}/year realistically calculated.

Accident analysis typically starts with formal descriptions of accident scenarios. Basic ETs may support such descriptions. All major assumptions in scenarios must be identified. The next step is determination of accident source terms. Source terms for accidents are obtained through phenomenological and system response calculations. Once a source term has been determined, consequences due to atmospheric dispersion or other relevant pathways of concern are determined. As with every phase of the analysis, the effort expended is a function of the estimated consequence. If the source term is small, a simple, dispersion hand calculation for consequences would be sufficient. If source terms are large, computer modeling to determine consequences may be required. The consequences finally determined are compared to the EG. From this activity, it is determined if SC SSC designation is needed. The need for accident specific TSRs to meet the EG will be determined.

Derivation of Hazard Controls, Demonstration of the Adequacy of the Controls, and Description of How the Controls are Maintained

The purpose of chapter 4 of the DSA is to provide information necessary to support the safety basis requirements of 10 CFR 830 for derivation of hazard controls.

This chapter provides details on those facility SSCs that are necessary for the facility to protect the public, provide DID, or contribute to worker safety. Similarly, this chapter provides details on SACs that are significant to specific accident risk reduction. Descriptions are provided of the attributes (i.e., functional requirements and performance criteria) required to support the safety functions identified in the hazard and accident analyses and to support subsequent derivation of TSRs. Expected products of this chapter, as applicable based on the graded approach, include the following:

- Descriptions of safety SSCs and SACs including safety functions
- Identification of support systems safety SSCs depended on to carry out safety functions
- Identification of the functional requirements necessary for the safety SSCs and SACs to perform their safety functions, and the general conditions caused by postulated accidents under which the safety SSCs or SACs must operate
- Identification of the performance criteria necessary to provide reasonable assurance that the functional requirements will be met
- Identification of assumptions needing TSR coverage

Existing supporting documentation is to be referenced. Maximum advantage should be taken of pertinent existing safety analyses and design information (i.e., requirements and their bases) that are immediately available or can be retrieved through reasonable efforts. Include a brief summary for each such reference that explains its relevance to this chapter and provides an introductory understanding of the reference.

Definition of the Characteristics of Safety Management Programs

Typical safety management programs include criticality protection, radiation protection, hazardous material protection, institutional safety provisions, procedures and training, operational safety, and emergency preparedness.

Definition of Criticality Safety Program, When Required

The purpose of chapter 6 is to provide information that will support the development of a safety basis in compliance with the provisions of 10 CFR 830.204(b)(6) regarding the definition of a criticality safety program. If this information is available in a site-wide criticality safety program description, and it complies with the rule requirements, then it can be included by reference and summarized in this chapter.

Expected products of this chapter include the following:

- Definition of a criticality safety program that 1) ensures that operations with fissionable material remain subcritical under all normal and credible abnormal conditions; 2) identifies applicable NCS standards; and 3) describes how the program meets applicable nuclear criticality standards
- Description of the basis and analytical approach the facility uses for deriving operational criticality limits
- Summary of design and ACs used by the criticality safety program

Existing supporting documentation is to be referenced. Include brief abstracts of referenced documentation with enough of the salient facts to provide an understanding of the reference documentation and its relation to this chapter.

e. Discuss the approval requirements for the DSA for new facilities and subsequent changes to the DSA.

The following is taken from 10 CFR 830, subpart B, appendix A to subpart B.

The safety basis requirements of 10 CFR 830 require the contractor responsible for a DOE nuclear facility to analyze the facility, the work to be performed, and associated hazards and to identify the conditions, safe boundaries, and hazard controls necessary to protect workers, the public, and the environment from adverse consequences. The safety basis requirements are intended to further the objective of making safety an integral part of how work is performed throughout the DOE complex.

A contractor must establish and maintain a safety basis for a hazard category 1, 2, or 3 DOE nuclear facility because these facilities have the potential for significant radiological consequences. DOE-STD-1027-92, chg 1, sets forth the methodology for categorizing a DOE nuclear facility. The hazard categorization must be based on an inventory of all radioactive materials within a nuclear facility. The safety basis requirements only apply to hazard

category 1, 2, and 3 nuclear facilities and do not apply to nuclear facilities below hazard category 3.

DOE expects that if a contractor has adequately implemented ISM, few additional requirements will stem from 10 CFR 830, subpart B and, in such cases, the existing safety basis prepared in accordance with ISM requirements of 10 CFR 830, subpart B. As part of the approval process, DOE will review the content and quality of the safety basis documentation.

A DSA must address all hazards (radiological and non-radiological hazards) and the controls necessary to provide adequate protection to the public, workers, and the environment from these hazards. A DSA must demonstrate the extent to which a nuclear facility can be operated safely with respect to workers, the public, and the environment.

Contractors are expected to use a graded approach to develop a DSA and describe how the graded approach was applied. The level of detail, analysis, and documentation will reflect the complexity and hazard associated with a particular facility. Thus, the DSA for a simple, low-hazard facility may be relatively short and qualitative in nature, while the DSA for a complex, high-hazard facility may be quite elaborate and more quantitative. DOE will work with its contractors to ensure a DSA is appropriate for the facility for which it is being developed.

DOE will review each DSA to determine whether the rigor and detail of the DSA are appropriate for the complexity and hazards expected at the nuclear facility. In particular, DOE will evaluate the DSA by considering the extent to which the DSA 1) satisfies the provisions of the methodology used to prepare the DSA; and 2) adequately addresses the criteria set forth in 10 CFR 830.204. DOE will prepare a safety evaluation report (SER) to document the results of its review of the DSA. A DSA must contain any conditions or changes required by DOE. In most cases, the contract will provide the framework for specifying the methodology and schedule for developing a DSA.

For construction after December 11, 2000, the contractor responsible for the design and construction of a new DOE nuclear facility or a major modification to an existing DOE nuclear facility must prepare a PDSA. A PDSA can ensure that substantial costs and time are not wasted in constructing a nuclear facility that will not be acceptable to DOE. If a contractor is required to prepare a PDSA, the contractor must obtain DOE approval of the PDSA prior to procuring materials or components or beginning construction.

DOE will examine and approve the TSRs as part of preparing the SER and reviewing updates to the safety basis. As with all hazard controls, TSRs must be kept current and reflect changes in the facility, the work, and the hazards as they are analyzed in the DSA.

The USQ process is an important tool to evaluate whether changes affect the safety basis. A contractor must use the USQ process to ensure that the safety basis for a DOE nuclear facility is not undermined by changes in the facility, the work performed, the associated hazards, or other factors that support the adequacy of the safety basis. The USQ process permits a

contractor to make physical and procedural changes to a nuclear facility and to conduct tests and experiments without prior approval, provided these changes do not cause a USQ.

The USQ process provides a contractor with the flexibility needed to conduct day-to-day operations by requiring only those changes and tests with a potential to impact the safety basis (and the safety of the nuclear facility) be approved by DOE. This allows DOE to focus its review on those changes significant to safety. The USQ process helps keep the safety basis current by ensuring appropriate review of, and response to, situations that might adversely affect the safety basis.

The DOE management official for a DOE nuclear facility (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. The DOE management official is responsible for ensuring the timely and proper review of all safety basis documents submitted to DOE and preparation of a safety evaluation report concerning the safety basis for a facility.

The following is taken from DOE G 421.1-2A.

DOE employs DSAs, TSRs, and SERs as the principal safety documentation in its decision to authorize operation of nuclear facilities. DOE-STD-1104-2009, *Review and Approval of Nuclear Facility Safety Basis and Safety Design Activities*, provides guidance on the preparation of SERs. 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.

The following is taken from DOE-STD-1189-2008.

DOE-STD-1189-2008, *Integration of Safety Into The Design Process*, applies to the design and construction of the following:

- New DOE hazard category 1, 2, and 3 nuclear facilities
- Major modifications to DOE hazard category 1, 2, and 3 nuclear facilities (as defined by 10 CFR 830)
- Other modifications to DOE hazard category 1, 2, and 3 nuclear facilities managed under the requirements of DOE O 413.3B, *Program and Project Management for the Acquisition of Capital Gains*

The activities and processes in DOE-STD-1189-2008 may be applied to new facilities and to modifications to those facilities not listed in DOE-STD-1189-2008.

f. Define who approves facility operations prior to achieving DSA upgrade approval.

The following is taken from 10 CFR 830.207.

Pending issuance of an SER in which DOE approves a safety basis for a hazard category 1, 2, or 3 existing DOE nuclear facility, the contractor responsible for the facility must continue to perform work in accordance with the safety basis for the facility in effect on October 10, 2000, or as approved by DOE at a later date, and maintain the existing safety basis consistent with the requirements of 10 CFR 830.207, "DOE Approval of Safety Basis."

The following is taken from DOE-STD-1104-2009.

10 CFR 830, appendix A, states "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 "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 safety evaluation report concerning the safety basis for a facility."

DOE O 413.3B assigns the authority to designate a safety basis approval authority (SBAA) 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 10 CFR 830.

The SBAA is responsible for providing a defensible review and approval of the DSA. Achieving defensible review and approval is facilitated by an independent review process. Since the preparation of the DSA and its review and approval typically fall within the purview of the SBAA, the SBAA assigns a review team leader the responsibility for performing the independent review. In making this assignment, the SBAA ensures that the review team leader maintains sufficient independence of the line organization responsible for DSA preparation (i.e., no responsibility for preparation of the DSA under review) and possesses the technical competence relevant to the DSA of concern. The details of independently reviewing the DSA, up to and including recommending approval to the SBAA, are managed by the review team leader.

The SBAA has responsibility as the single point of contact between DOE and the facility contractor for all matters regarding review of the DSA. This responsibility is typically assigned to the review team leader, but the SBAA remains the final authority on any points requiring arbitration. The single point of contact is the focal point through which DOE and the facility contractor interface and from which directions to the facility contractor originate. Requests for any material on the DSA, determination of the significance of identified issues on such material, and direction to the facility contractor for resolution of issues are approved by the single point of contact. As appropriate, transmittal of official communications and

directions involving significant work effort by the facility contractor are coordinated with the contracting officer. Line management personnel and representatives of organizations, responsible for monitoring and auditing implementation of 10 CFR 830, coordinate their activities through the single point of contact as well.

The SBAA has the specific responsibility of ensuring that the review and approval process represents all DOE entities with vested interest in the facility under review and considers commitments made to agencies outside DOE. Agencies external to DOE, however, have no standing under the orders/rules structure for approval. Identifying safety issues and their resolution may involve negotiations between concerned organizations. Issues raised by any vested interest should be given proper consideration to enhance safety assurance.

On behalf of the SBAA, the review team leader coordinates the day-to-day aspects of managing the review and approval process for the DSA. General responsibilities in this capacity include the following:

- Serving as the focal point for interface between DOE and the facility contractor for review matters
- Developing a DSA review plan, including review milestones developed in consultation with the facility contractor
- Establishing and managing the review team
- Supervising the overall review process, including planning and scheduling changes
- Coordinating, scheduling, and arbitrating issue resolution
- Preparing an SER

The SBAA has the responsibility for ensuring adequate performance of the review team leader in fulfilling assigned responsibilities.

g. Discuss the provisions for deviations and temporary and permanent exemptions from the 10 CFR 830.204 and safe harbor methodologies.

The following is taken from DOE-STD-1104-2009.

If a contractor uses a method other than a safe harbor method from 10 CFR 830, appendix A, table 2, per 10 CFR 830.204, the contractor must obtain DOE approval of the method before developing the DSA. Likewise, if a contractor uses a safe harbor method to develop the DSA, but does not follow the method completely, per 10 CFR 830.204, the contractor must request DOE approval of the method with the specific deviations noted. Requirements and responsibilities for the use of alternative methods or specific deviations from the safe harbor methods are contained in DOE O 410.1, *Central Technical Authorities Responsibilities Regarding Nuclear Safety Requirements*.

h. Discuss the application of the graded approach relative to the DSA development.

The following is taken from DOE-STD-3009-94, Chg 3.

10 CFR 830 prescribes the use of a graded approach for the effort expended in safety analysis and the level of detail presented in associated documentation. The graded approach applied to DSA preparation and updates is intended to produce cost efficient safety analysis and DSA

content that provide adequate assurance to the DOE that a facility has acceptable safety provisions without providing unnecessary information. As described in 10 CFR 830, the graded approach adjusts the magnitude of the preparation effort to the characteristics of the subject facility based on the following seven factors:

1. The relative importance to safety, safeguards, and security
2. The magnitude of any hazard involved
3. The life-cycle stage of a facility
4. The programmatic mission of a facility
5. The particular characteristics of a facility
6. The relative importance of radiological and nonradiological hazards
7. Any other relevant factor

10 CFR 830 provides for developing the DSA based on judgment of the facility in relation to these seven factors. For example, simple hazard category 3 facilities or facilities that have a short operational life may only require a limited but adequate analysis documented to a level less than that required for a hazard category 2 facility. In addition, facilities with short operational lives (or other compelling circumstances) should consider the appropriateness of using DOE-STD-3011-2002, *Guidance for Preparation of Basis for Interim Operation (BIO) Documents*, to meet the requirements of 10 CFR 830. On the opposite end of the spectrum, a complex hazard category 1 facility that is just going into operation requires extensive analysis and highly detailed documentation.

The application of the graded approach may allow for much simpler analysis and documentation for some of these facilities. For facilities of little hazard, or hazards at the hazard category 3 level, for which only a modest reduction of risk is required, the DSA may be simple and short. In such cases, all of the topics for the DSA listed in DOE-STD-3009-94, Chg 3, may not be necessary and with proper technical bases some topics may be omitted or reduced in the detail that would otherwise be required of hazard category 1 or 2 facilities.

16. Nuclear safety specialists must demonstrate a working level knowledge of 10 CFR 830.207, "DOE Approval of Safety Basis," and DOE-STD-1104-96, *Review and Approval of Nuclear Facility Safety Basis Documents (Documented Safety Analyses and Technical Safety Requirements)*, with respect to their impact on the Department's nuclear safety.
[Note: DOE-STD-1104-96 has been updated by DOE-STD-1104-2009.]

a. Describe the basic purpose and contents of a safety evaluation report.

The following is taken from DOE-STD-1104-2009.

The SER is primarily a management document that provides the SBAA the basis for the extent and detail of the review of the PDSA, DSA, and TSRs and the bases for any conditions of approval. DOE-STD-1104-2009 endorses the concept that the contents of an SER are concise summary statements and that little benefit is gained from the wholesale repetition of elements already contained in a PDSA, DSA, or TSR or from reproducing original analysis that, if deemed critical, is performed as part of the review process.

SERs document the bases for approving revisions of DSAs/TSRs, including annual updates. Those revisions determined to not involve a USQ in accordance with 10 CFR 830.203, may be reviewed and approved for accuracy and completeness by DOE subsequent to implementation of the changes by the facility contractor.

b. Describe the bases for approval contained in an SER.

Base Information

The following is taken from DOE-STD-1104-2009.

Base information is the first of the approval bases that should be reviewed and encompasses elements of DSA preparation, completeness, and general content. Base information is reviewed for sufficiency to allow assessment of the other approval bases that rely on this information. The review for sufficiency can range from a simple screening effort to more detailed discussions, depending on the complexity of the DSA.

Insufficient or incomplete base information in a DSA may prevent further review of the DSA. Reviewers should require resolution of major discrepancies in base information (e.g., incomplete site characteristics) before evaluation of the more specific aspects (e.g., hazard and accident analyses) of the safety basis proceeds. It is for this reason that the SER need only provide a brief statement as to the adequacy of base information.

For example, for DSAs adhering to the DOE-STD-3009-94, Chg 3 format, the review of base information primarily determines the sufficiency of the information provided in the executive summary, chapter 1, "Site Characteristics"; chapter 2, "Facility Description"; and to some extent, material generic to all DSA chapters (e.g., statutes, rules, Orders, and principal health and safety criteria). Determining the adequacy of base information generally entails being able to conclude that the DSA contains sufficient documentation and basis to arrive at the following conclusions:

- The facility contractor development and approval processes (e.g., personnel involvement in developing the DSA, management cognizance and acceptance, internal reviews) demonstrate sufficient commitment to establish the facility safety basis.
- The facility mission(s) and scope of operations (i.e., the scope of work to be performed) for which safety basis approval is being sought are clearly stated and reflected in the type and scope of operations analyzed in the DSA. For example, a DSA documenting the safety basis of a spent fuel storage facility whose mission includes size reduction of spent fuel elements would be unacceptable if the DSA omitted safety analysis of size-reduction operations.
- A description of the facility's life-cycle stage, mission(s), scope of operations, and the design of safety SSCs is presented, including explanation of the impact on the facility safety basis.
- Clear bases for, and provision of, exemptions, consent agreements, and open issues are presented.
- Description of the site, facility, and operational processes provide a knowledgeable reviewer sufficient background material to understand the major elements of the safety analysis.

- Correlation is established between actual facility arrangements and operations with those stated in the DSA. This may be accomplished successfully through reference to facility walkthroughs during DSA preparation. Walkthroughs may be warranted during DSA review to provide some level of assurance that the actual physical arrangement of a facility corresponds to that documented in the DSA. For example, a walkthrough may be considered for a facility and/or operation that were modified in the time frame between when DSA development was started and completed. This is not intended to imply the review team must perform detailed verifications of facility configuration. The objective is to allow the review team to conclude that the basic descriptions provided are fundamentally up-to-date and correct.

Hazard and Accident Analyses

The following is taken from DOE-STD-1104-2009.

Another of the DSA approval bases is hazard and accident analyses that form the foundation for the remaining approval bases (i.e., safety SSCs, SACs, derivation of TSRs, and safety management program characteristics). Determining the adequacy of hazard and accident analyses generally entails being able to conclude that the DSA contains sufficient documentation and basis to arrive at the following conclusions:

- The HA includes hazard identification that specifies or estimates the hazards relevant for DSA consideration (i.e., natural and man-made hazards associated with the work and the facility) in terms of type, quantity, and form and includes properly performed facility hazard categorization.
- The final hazard category for the facility is determined consistent with DOE-STD-1027-92, Chg 1, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. Any differences between the final hazard category and the initial hazard category are explained.
- The HA includes a hazard evaluation that covers the activities for which approval is sought, is consistent in approach with safe harbor methodologies, identifies preventive and mitigative features for the spectrum of events examined, and identifies dominant accident scenarios through ranking.
- The HA evaluates 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 that may be beyond the design basis of the facility.
- The HA results are clearly characterized in terms of public safety, DID, worker safety, and environmental protection as part of the safety basis of the facility. The logic behind assessing the results in terms of SS SSCs, SACs, and designation of TSRs is understandable and internally consistent.
- Subsequent accident analysis clearly substantiates the findings and delineations of HA for the subset of events examined and confirms their potential consequences. SC SSCs and SS SSCs, SACs, and associated TSRs have been identified for preventing and/or mitigating events potentially exceeding EGs.

The goal of the review is to ensure that the safety basis is comprehensive relative to hazards presented and is based on a consistent, substantiated logic. As a minimum, reviewers should utilize the safe harbor methodology used in developing the DSA, DOE-STD-1189-2008, and DOE's DSA guide DOE G 421.1-2 as a reference to support their review. In addition, the review of the HA should include the entire analysis and not just the summary of the analysis that might be included in the DSA.

Safety Structures, Systems, and Components

The following is taken from DOE-STD-1104-2009.

The next DSA approval basis is safety SSCs. Identification of safety SSCs is a product of the hazard and accident analyses. Determining the adequacy of safety SSCs generally entails being able to conclude that the DSA contains sufficient documentation and basis to arrive at the following conclusions:

- The safety SSCs identified and described are consistent with the logic presented in the hazard and accident analyses.
- Safety functions for safety SSCs are defined with clarity and are consistent with the bases derived in the hazard and accident analyses.
- The boundaries of safety SSCs are clearly defined, including the support systems.
- Functional requirements and system evaluations are derived from the safety functions and provide evidence that the safety functions can be performed when called upon.
- System evaluation is performed to assure functional requirements are met.
- Control of safety SSCs relevant to TSR development is clearly defined.

Derivation of Technical Safety Requirements

The following is taken from DOE-STD-1104-2009.

Derivation of TSRs is the next of the DSA approval bases. Hazard controls are derived to eliminate, limit, or mitigate hazards. The controls generally are safety SSCs, SACs, or commitments to safety management programs that are ultimately included in TSRs. Identification of TSRs results from the most significant preventive and mitigative features identified in the hazard and accident analyses and from the designation of safety SSCs and SACs. Determining the adequacy of the derivation of TSRs generally entails being able to conclude that the DSA contains sufficient documentation and bases to arrive at the following conclusions:

- TSRs are identified to ensure adequate protection of workers, the public, and the environment.
- The bases for deriving TSRs are identified and described in the hazard and accident analyses, safety SSC, and SAC chapters and are consistent with the logic and assumptions presented in the analyses.
- The bases for deriving SLs, LCSs, LCOs, surveillance requirements (SRs), and ACs are provided as appropriate.
- The process for maintaining the TSRs current at all times and for controlling their use is defined.

Safety Management Program Characteristics

The following is taken from DOE-STD-1104-2009.

Safety management program characteristics are the last of the DSA approval bases and encompass the elements of institutional programs and facility management that are necessary to ensure safe operations based on assumptions made in the hazard and accident analyses. While these elements must be addressed in the DSA, generic descriptions of these institutional programs should not be duplicated in the DSA if they can be referenced in ISMS documents or site-wide manuals. These institutional programs include (where applicable) QA, procedures, maintenance, personnel training, conduct of operations, emergency preparedness, fire protection, waste management, radiation protection, and criticality safety. Identification of safety management program characteristics and credited attributes is a product of hazard and accident analyses, designation of safety SSCs and SACs, and derivation of TSRs. Determining the adequacy of safety management program characteristics generally entails being able to conclude that the DSA contains sufficient documentation and basis to arrive at the following conclusions:

- The major programs needed to provide programmatic safety management are identified.
- Basic provisions of identified programs are noted, and references to facility or site program documentation are provided.

The review of safety management programs is normally performed at the site-wide level and a reference to the site-wide review is sufficient to support the DSA review. The acceptance of safety management program characteristics does not constitute acceptance of the adequacy of program compliance with DOE directives. Acceptance can be accomplished only by a detailed compliance review or each of the programs that is beyond the scope of a DSA. Detailed reviews of the adequacy of safety management programs only need to be performed for safety management programs that are specific to the facility.

The approval bases for the TSR document are the TSR provisions. These TSR provisions may be design features, SLs, operating limits (OLs), SRs, or ACs (primarily commitments to implement safety management programs according to the facility-specific characteristics described in the DSA). The approval bases for a TSR document include a disciplines analysis and tracing of commitments to hazard controls through appropriate provisions that implement these controls in a TSR document. In some cases the specific treatment of safety controls in the TSR is committed to in the DSA; in other cases, it is a judgment call as to the appropriate TSR treatment. Determining the adequacy of the TSR provisions generally entails being able to conclude that

- hazard controls discussed in the DSA are faithfully translated to TSR provisions
- TSR provisions are appropriate and consistent with the DSA

The sources of information in a DSA regarding these provisions are the HA (including description of hazard controls); the description of safety SSCs; the classification of these SSCs as SC, SS, or other important DID SSCs; the description of the functional requirements for the safety SSCs; the description and functional requirements for SACs; the derivation of TSRs; and the descriptions of the safety management programs.

An HA will include a disciplined analysis of all hazards within the scope of the DSA, including a listing of applicable preventive and mitigative hazard controls. These controls may include safety SSCs, design features, SACs, and provisions of various safety management programs. These controls should be regarded as DSA commitments. They should be traced through DSA documentation to specific TSR provisions.

Safety SSCs must be described in sufficient detail in a DSA so that their functional requirements are defined and the bases for TSR requirements are derived. These safety SSCs will be either active or passive. If passive, they should be considered for designation as "design features" in the TSR. These are features of facility design that may not be changed without DOE review and approval. A cross-check between DSA-identified important design features and the design features section of the TSR should be conducted to ensure consistency. If active, safety SSCs will usually have a safety limit and a limiting control setting associated with them, as well as a surveillance requirement. An active SS SSC may have an LCO and SRs and/or specific provisions of a maintenance management program associated with it. In any case, SC SSCs must be addressed specifically in TSR provisions. Technical bases for LCSs and surveillance requirements in the bases appendix of the TSR should be reviewed for adequacy. All of these provisions are directed at ensuring that the safety function of the SSC will be protected.

When SACs are used, they must be controlled through the TSR. DOE-STD-1186-2004, *Specific Administrative Controls*, specifies the methodologies that are acceptable to use for SACs. The first involves using the conventions LCOs and associated SRs. The second method available to incorporate SACs in TSR documents is to identify the specific requirement/action in a special section in the AC section of the TSR. This format may be appropriate when it is essential that the SAC be performed every time and without any delay when called on or when definitive program requirements for specific activities can be established.

DOE line management can identify conditions of approval in the SER for the DSA. The conditions of approval may include the identification of additional important DID items and ACs. Furthermore, although not preferred, DOE may require additional important DID items and ACs, independent of safety analysis. If important DID items and ACs are identified, the review team should ensure that TSRs have been developed to provide assurance of the identified safety function as appropriate.

HAs may invoke particular aspects of safety management programs, such as emergency preparedness, criticality safety, procedures, and training. Any particular provisions of these programs unique to the facility should have been described in the DSA. The AC section of the TSR should include commitments to implement those programs identified in the DSA as important to the facility safety basis.

<u>Mandatory Performance Activities</u>:

a. Demonstrate by direct preparation or assistance in the development of at least two SERs consistent with DOE-STD-1104-2009, *Review and Approval of Nuclear Facility Safety Basis and Safety Design Basis Documents*, for a new safety basis, significant update, or safety basis amendment.

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

17. Nuclear safety specialists must demonstrate a working level knowledge of 10 CFR 830.206, "Preliminary Documented Safety Analysis," with respect to its impact on the Department's nuclear safety.

a. Describe the application of the requirements of DOE O 420.1B, *Facility Safety*, and its guidance to the development process for the PDSA.
[Note: DOE O 420.1B has been updated to DOE O 420.1C and no longer contains information on the PDSA process.]

The following is taken from DOE-STD-1189-2008.

The PDSA is the 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 following:

- 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
- An initial listing of the safety management programs that must be developed to address operational safety considerations

The following is taken from DOE-STD-3009-94, Chg 3.

Guidance provided by DOE-STD-3009-94, Chg 3, describes a DSA preparation method that is acceptable to the DOE as delineated for those specific facilities listed in 10 CFR 830, appendix A to subpart B, table 2. It was developed to assist hazard category 2 and 3 facilities in preparing SARs that will satisfy the requirements of 10 CFR 830. Hazard category 1 facilities are typically expected to be category A reactors for which extensive precedents for SARs already exist.

Guidance provided by DOE-STD-3009-94, Chg 3, is generally applicable to any facility required to document its safety basis in accordance with 10 CFR 830. For new facilities where conceptual design or construction activities are in process, PDSA elements of this guidance may be more appropriately handled as an integral part of the overall design requirements process. The methodology provided by DOE-STD-3009-94, Chg 3, focuses more on characterizing facility safety with or without well-documented information than on the determination of facility design. Accordingly, contractors for facilities that are documenting conceptual designs for PDSAs should apply the process and format of DOE-STD-3009-94, Chg 3, to the extent it is judged to be of benefit.

b. Describe the sequencing of the PDSA relative to design, procurement, construction, and operation of new facilities.

The following is taken from DOE G 421.1-2A.

In accordance with DOE O 420.1C and DOE O 413.3B, DOE-STD-1189-2008 governs the development of a safety design basis from conceptual design through final design, including the development of a PDSA during final design. 10 CFR 830.206 allows limited procurement and construction activities before a PDSA is approved if DOE determines that the activities are not detrimental to public health and safety or the environment and are in the best interests of DOE.

The PDSA for a new facility prepared under the guidance of DOE-STD-1189-2008 is of the same format as DOE-STD-3009-94, Chg 3, for existing facilities. However, the process of establishing an operational safety basis for a new facility is different from that for an existing facility because the safety in design process of DOE-STD-1189-2008 results in a documented safety design basis.

DOE-STD-1189-2008 provides several tables illustrating the sequencing of the design process to the development of safety basis documents.

c. Describe the circumstances when a PDSA must be prepared.

The following is taken from DOE-STD-1189-2008.

In accordance with DOE O 413.3B, contractors responsible for the design of DOE hazard category 1, 3, and 3 nuclear facilities must implement DOE-STD-1189-2008, appendix A, "Safety System Design Criteria," safety criteria and submit the following safety basis documents for DOE approval (unless otherwise agreed to):

- Conceptual design stage—safety design strategy
- Conceptual design stage—conceptual safety design report (CSDR)
- Preliminary design stage—preliminary safety design report (PSDR)
- Final design stage—PDSA
- Prior to operations—DSA and TSRs

The Federal project director is responsible for ensuring the implementation of the requirements of DOE O 413.3B and does this through contract requirements and support of DOE entities. The Federal project director approves or concurs on the approval of all safety submittals of the project. The DOE SBAA approves of the CSDR, PDSR, and PDSA based on a review by a safety basis review team. DOE-STD-1189-2008, table 2-1, "Roles and Responsibilities," shows the entities responsible for preparation, review, and approval of these safety documents.

The following is taken from 10 CFR 830.206.

The contractor responsible for a hazard category 1, 2, or 3 new DOE nuclear facility or a major modification to a hazard category 1, 2, or 3 DOE nuclear facility must

- prepare a PDSA for the facility; and

- obtain DOE approval of
  - the nuclear safety design criteria to be used in preparing the PDSA unless the contractor uses the design criteria in DOE O 420.1C; and
  - the PDSA before the contractor can procure materials or components or begin construction, provided that DOE may authorize the contractor to perform limited procurement and construction activities without approval of a PDSA if DOE determines that the activities are not detrimental to public health and safety and are in the best interests of DOE.

d. Describe the relationship between the PDSA and the design process.

The following is taken from DOE G 421.1-2 (Archived).

For the design and construction of a new facility or activity, it is imperative that safety be addressed early so that it can be "designed-in" instead of "added-on." To achieve this integration of safety into design, there needs to be continuous interaction between safety analysis and the designers throughout the design process. All of the hazards (nuclear, explosive, natural phenomena, fire, criticality, etc.) should be addressed as early as possible in the design of new nuclear facilities and major modifications so that passive and active design concepts can be economically incorporated into the design.

18. Nuclear safety specialists must demonstrate a working level knowledge of 10 CFR 830.202, "Safety Basis," and 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 nuclear safety.

a. Describe when a contractor must establish a safety basis for a facility.

The following is taken from DOE, Oak Ridge Operations Office, *Nuclear Facility Safety Basis Fundamentals Self-Study Guide*.

10 CFR 830, Subpart B, requires the contractor responsible for a DOE nuclear facility to analyze the facility, the work to be performed, and the associated hazards and then identify the conditions, safe boundaries, and hazard controls necessary to protect the workers, the public, and the environment from adverse consequences. These analyses and hazard controls constitute the safety basis on which the contractor and DOE rely to conclude the facility operations or activities can be conducted safely.

The contactor submits the DSA describing the facility risks and the proposed controls that will be employed to assure that operations can be conducted safely to the authorizing official within DOE for review and approval. This document becomes the safety basis that is used as the primary input for DOE to determine whether operations can be authorized. The process of approving the safety basis constitutes DOE's acceptance of the risks. The concept of authorizing operations based on risk acceptance is essentially a license under which the contractor may operate a facility.

b. Describe the requirements for the safety basis.

The following is taken from 10 CFR 830.202.

In establishing the safety basis for a hazard category 1, 2, or 3 DOE nuclear facility, the contractor responsible for the facility must

- define the scope of the work to be performed;
- identify and analyze the hazards associated with the work;
- categorize the facility consistent with DOE-STD-1027-92, Chg 1;
- prepare a DSA for the facility; and
- establish the hazard controls on which the contractor will rely to ensure adequate protection of workers, the public, and the environment.

c. Describe the requirements the contractor must perform to maintain the safety basis.

The following is taken from 10 CFR 830.202.

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;
- 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.

d. Discuss the purpose of hazard categorization and determine the hazard categorization of an operating nuclear facility.

The following is taken from 29 CFR 1910.1200.

The purpose of hazard categorization is to ensure that the hazards of all chemicals produced or imported are classified, and that information concerning the classified hazards is transmitted to employers and employees. The transmittal of information is to be accomplished by means of comprehensive hazard communication programs that are to include container labeling and other forms of warning, safety data sheets and employee training.

The following is taken from DOE-STD-1027-92, Chg 1.

DOE-STD-1027-92, Chg 1, focuses on the following:

- The definition of the standard identifying nuclear facilities required to have SARs
- The SAC implementation plan and schedule
- The hazard categorization methodology to be applied to all facilities
- The accident analysis techniques appropriate for the graded approach

Facilities that do not meet or exceed category 3 threshold criteria, but still possess some amount of radioactive material, may be considered radiological facilities. Radiological facilities are except but they are not exempt from other safety requirements (they must meet 10 CFR 835 requirements). The hazard categories include the following:

- Category 1 hazard—The HA shows the potential for significant offsite consequences.
- Category 2 hazard—The HA shows the potential for significant onsite consequences.
- Category 3 hazard—The HA shows the potential for significant but localized consequences.

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 as directed. 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).

Once an HA has been performed, the hazard categorization can be finalized. The final categorization is based on an "unmitigated release" of available hazardous material. For the purposes of hazard categorization, "unmitigated" is meant to consider material quantity, form, location, dispersability, and interaction with available energy sources, but not to consider safety features that will prevent or mitigate a release.

e. Describe the exclusion types for radionuclides associated with hazard categorization determination.

The following is taken from DOE-STD-1027-92, Chg 1.

Sealed radioactive sources that are engineered to pass the special form testing specified by the Department of Transportation (DOT) in 49 CFR 173.469, "Tests for Special Form Class 7 (Radioactive) Materials," or testing specified in ANSI N43.6, *Sealed Radioactive Sources, Categorization*, may be excluded from summation of a facility's radioactive inventory. The facility must have documentation that the source or prototypes of the source have been tested and passed the tests specified by DOT or ANSI. Facilities must have in place a source control policy. Should a sealed radioactive source fail, as indicated by an increase in the removable activity, the source shall be removed from service and handled in accordance with the source control policy established for the facility.

Hazardous materials used in exempted, commercially available products, should not be considered part of a facility's inventory. These materials are described in 10 CFR 30.11, "Specific Exemptions" through 10 CFR 30.19, "Self-Luminous Products Containing Tritium, Krypton-85, or Promethium-147," and include timepieces, illumination devices, thermostats, electron tubes, microwave receiver tubes, etc.

Additionally, material contained in DOT type B shipping containers (with or without overpack) may be excluded from summation of a facility's radioactive inventory if the certificates of compliance are kept current and the materials stored are authorized by the

certificate. However, type B containers without overpack should have heat protection provided by the facility's fire suppression system. These exclusions do not apply to fissile material in the determination of hazard category 2 status relative to criticality.

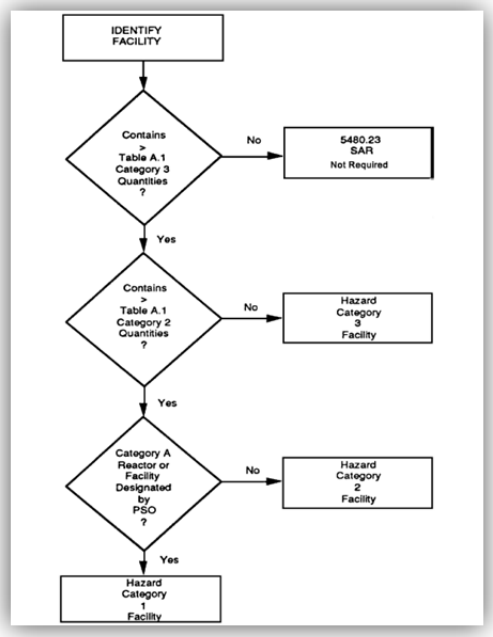
f. Describe the differences between initial and final hazard categorizations and where these designations occur in the DSA development process.

Initial Radiological Hazards Screening

The following is taken from DOE Energy Facility Contractors Group, *EM Hazard Categorization Guidance*.

The initial hazard categorization of the facility is based strictly on comparison of the total inventory of radionuclides to the DOE-STD-1027-92, table A.1, "Threshold Quantities (TQ)," as well as consideration of criticality mass limits for fissile materials (i.e., per the asterisk to table A.1). Facilities with radionuclide inventory below the DOE-Std-1027-92 TQ values for hazard category 3 can be categorized as radiological, unless the facility contains fissile materials in quantities greater than the theoretical mass limits for criticality emergencies as specified in ANSI/ANS-8.1-1998. The facility would then be considered hazard category 2. For facilities that have an initial or final categorization above hazard category 3, the basis and assumptions should be described in the DSA. Facilities that have an initial hazard categorization below hazard category 3 are not required to obtain DOE approval on the categorization; however, the basis and assumptions that support the initial hazard categorization should still be documented and provided to DOE for information purposes.

An overview of the radiological hazards screening is provided in figure 72.



Source: DOE-STD-1027-92, Chg 1

Figure 71. Radiological hazards screening

Final Hazard Categorization

The following is taken from DOE-STD-1027-92, Chg 1.

Once an HA has been performed, the hazard categorization can be finalized. The final categorization is based on an unmitigated release of available hazardous material. For the purposes of hazard categorization, unmitigated is meant to consider material quantity, form, location, dispersability, and interaction with available energy sources, but not to consider safety features (e.g., ventilation system, fire suppression, etc.) that will prevent or mitigate a release.

The HA (or other existing safety analyses) provides an understanding of the material that can physically be released from the facility. This inventory should be compared against the TQ identified in DOE-STD-1027-92, Chg 1, attachment 1, "Hazard Categorization of DOE Facilities." The airborne release fractions (ARFs) used in generating the TQ values for category 2 are provided in DOE-STD-1027-92, Chg 1, attachment 1, page A-9. As discussed in the attachment, these are intended to be generally conservative for a broad range of possible situations. Therefore, the inventory values may be used directly for determination as to whether a facility exceeds category 2. Alternatively, for final categorization, for facilities initially classified as hazard category 2, if the credible release fractions can be shown to be significantly different than these values based on physical and chemical form and available dispersive energy sources, the threshold inventory values for category 2 may be divided by the ratio of the maximum potential release fraction to that found on page A-9. All assumptions that are used to reduce the inventory at risk should be supported in the HA. This applies to ground rules identified in DOE-STD-1027-92, Chg 1, attachment 1, to demonstrate that the ground rule conditions exist.

The following is taken from DOE Energy Facility Contractors Group, *EM Hazard Categorization Guidance*.

Final hazard categorizations that result in a determination of below hazard category 3, based on an HA, require DOE approval. Since final hazard categorization documents are the primary safety basis document for the facilities determined to be below hazard category 3 by analysis, the documents should provide sufficient background information for DOE to understand the nature of the facility and its operations.

If a change is made, or new information discovered that affects a condition, parameter, or assumption that helps form the basis for the hazard category downgrade, approved safety basis documents must be revised to reflect the change. The revised final hazard categorization documents must then be reviewed by DOE prior to making the change to ensure that the basis for the approval of the hazard category has not changed.

Mandatory Performance Activities:

a. Participate in at least two hazard categorization designation reviews.

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

19. Nuclear safety specialists must demonstrate a familiarity level knowledge of DOE O 420.1B, *Facility Safety*; DOE G 420.1-1, *Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria Guide for Use with DOE O 420.1, Facility Safety*; and DOE-STD-1020-2002, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of energy Facilities*; with respect to their impact on the Department's nuclear safety.

[Note: DOE O 420.1B has been updated to DOE O 420.1C; DOE G 420.1-1 has been updated to DOE G 420.1-1A; and DOE-STD-1020-2002 is now DOE-STD-1020-2012, *Natural Phenomena Hazards Analysis and Design Criteria for Department of Energy Facilities*.]

a. Discuss the purpose and policy associated with DOE O 420.1B.

The following is taken from DOE O 420.1C.

Objectives

The objective of DOE O 420.1C is to establish facility and programmatic safety requirements for the DOE, including the NNSA, for

- nuclear safety design criteria
- fire protection
- criticality safety
- NPH mitigation
- cognizant system engineer program

Facility safety requirements for explosive, chemical, and industrial hazards are contained in other DOE rules and directives.

Policy

DOE O 420.1C applies to all DOE elements with responsibility for design, construction, management, operation, decontamination, decommissioning, or demolition of government-owned or government-leased facilities and onsite contractor-leased facilities used for DOE mission purposes. Except for the equivalencies and exemptions listed in DOE O 420.1C, the contractor requirements document (CRD) sets forth requirements of DOE O 420.1C that will apply to contracts that include the CRD. The CRD, or its requirements, must be inserted in all contracts that require design, construction, management, operation, decontamination, decommissioning, or demolition of government-owned and government-leased facilities.

b. Discuss the role of the Department's nuclear safety specialist with respect to the implementation of the requirements of DOE O 420.1B.

The following is taken from DOE O 420.1C.

The role of the Department's NSS in respect to the implementation of the requirements of DOE O 420.1C is to enforce DOE's nuclear safety policy (DOE P 420.1, *Department of Energy Nuclear Safety Policy*). DOE's nuclear safety policy is to design, construct, operate, and decommission its nuclear facilities in a manner that ensures adequate protection of workers, the public, and the environment.

c. Discuss the Department policy and objectives with respect to safety-class and safety-significant criteria.

The following is taken from DOE O 420.1C.

Objectives

The objective of the SC and SS criteria is to establish requirements for the design and construction of SSCs, SC and SS, by identifying an applicable set of industry codes and standards, as well as DOE design criteria, standards, and directives. Compliance with these requirements will ensure reliable performance of the safety function of SSCs under those conditions and events for which they are intended.

Policy

These requirements apply to the design and construction of

- new hazard category 1, 2, and 3 nuclear facilities as defined in 10 CFR 830; and
- major modifications to hazard category 1, 2, and 3 nuclear facilities, as defined in 10 CFR 830 that substantially change the facility safety basis.

It does not impose requirements on existing facilities, except for major modifications to those facilities and it does not apply to nuclear activation or decontamination and decommissioning activities at the end of facility life, if the safety analysis demonstrates that adequate protection is provided consistent with the requirements of 10 CFR 830 through alternate means and it is not cost-beneficial to apply the provisions for the limited remaining life of the activity.

d. Discuss the facility and activity applicability of DOE O 420.1B, with respect to implementation associated with the design of nuclear facilities.

[Note: DOE O 420.1B has been updated to DOE O 420.1C.]

The following is taken from DOE O 420.1C.

Nuclear facility design:

- The nuclear facility design must include multiple layers of protection (as part of the design DID) to prevent or mitigate the unintended release of radioactive materials into the environment. DID must include all of the following:
  - Choosing an appropriate site
  - Minimizing the quantity of material-at-risk
  - Applying conservative design margins
  - Applying QA
  - Using successive/multiple physical barriers for protection against radioactive releases
  - Using multiple means to ensure safety functions are met by
    - controlling processes;
    - maintaining processes in safe status;
    - providing preventive and/or mitigative controls for accidents with the potential for radiological releases; and

        - providing means for monitoring facility conditions to support recovery from upset or accident conditions.
    - Using equipment in combination with ACs that
        - restrict deviation from normal operations;
        - monitor facility conditions during and after an event; and
        - provide for responses to accidents to achieve a safe condition.
    - Providing means to monitor accident releases as required for emergency response
    - Establishing emergency plans for minimizing the effects of an accident
- Hazard category 1, 2, and 3 nuclear facilities with uncontained radioactive materials must have the means to confine the uncontained radioactive materials to minimize their potential release in facility effluents during normal operations and during and following accidents, up to and including DBAs. Confinement design must include the following:
    - For a specific nuclear facility, the number, arrangement, and characteristics of confinement barriers as determined on a case-by-case basis
    - The type, quantity, form, and conditions for dispersing the radioactive material in the confinement system design
    - An active confinement ventilation system as the preferred design approach for nuclear facilities with potential for radiological release
    - Documentation of the adequacy of confinement systems consistent with the safety in design process as described in DOE-STD-1189-2008
- Hazard category 1, 2, and 3 nuclear facilities must be designed to
    - facilitate safe deactivation, decommissioning, decontamination, and demolition at the end of facility life, including incorporation of design considerations during the operational period that facilitate future decontamination and decommissioning;
    - facilitate inspections, testing, maintenance, repair, and replacement of safety SSCs as part of a reliability, maintainability, and availability program with the objective of maintaining the facility in a safe state;
    - keep occupational radiation exposures within regulatory limits, and ALARA;
    - provide controls consistent with the hierarchy described in DOE-STD-1189-2008; and
    - protect against chemical hazards and toxicological hazards consistent with DOE-STD-1189-2008 and direction from the responsible program office.
- Facility process systems must be designed to minimize waste production and mixing of radioactive and non-radioactive wastes.
- Safety SSCs and safety software must be designed to perform their safety functions when called on.
- Active SC systems must be designed to meet single point failure criterion.
- DOE G 420.1-1A provides an acceptable method to meet the requirements stated in DOE O 420.1C, chapter I, “Nuclear Safety Design Criteria.”
- New DOE nuclear reactors must comply with the requirements of DOE O 420.1C, attachment 2, as well as the design requirements of DOE Order 5480.30, Chg 1, *Nuclear Reactor Safety Design Criteria.*

- Critical experiments facilities must be designed and operated in accordance with ANSI and ANS standards, ANSI/ANS-1-2000, *Conduct of Critical Experiments*, or ANSI/ANS-14.1-2004, *Operation of Fast Pulse Reactors*.
- Facility design must be integrated with other design requirements, as applicable, including explosive safety, industrial safety, and nuclear explosive safety (if applicable).

e. Identify and discuss the use of DOE standards for seismic safety.

The following is taken from DOE-STD-1020-2012.

To comply with Public Law (PL) 101-614, "An Act to Authorize Appropriations for the Earthquake Hazards Reduction Act of 1977, and for Other Purposes," and Executive Order (EO) 12941, "Seismic Safety of Existing Federally Owned or Leased Buildings," the guidelines provided in the Interagency Committee on Seismic Safety in Constructions (ICSSC) RP-6, *Standards of Seismic Safety for Existing Federally Owned and Leased Buildings*, shall be used to

- determine when a seismic evaluation and retrofitting of an existing non-nuclear facility will be necessary; and
- establish the evaluation and mitigation requirements. Additional guidelines are provided in ICSSC RP-5, *ICSSC Guidance on Implementing Executive Order 12941, Seismic Safety of Existing Federally Owned or Leased Buildings*.

For facilities with significant chemical or toxicological hazards, the design should consider the guidance and criteria in DOE-STD-1020-2012, section 2.3.9, "Additional Guidance for Hazard Category 1, 2, and 3 Nuclear Facilities with Chemical Hazards."

The following is taken from ICSSC RP-6.

In response to PL 101-614, the ICSSC RP-4, *Standards of Seismic Safety for Existing Federally Owned or Leased Buildings and Commentary*, was issued by ICSSC in 1994. Pursuant to EO 12941, the standards are periodically updated to incorporate advanced knowledge in earthquake engineering gained from research and from observed performance of structures in recent earthquakes. ICSSC RP-6 is the revision of ICSSC RP-4.

The intent of ICSSC RP-6 is to provide Federal agencies with common minimum and higher standards for the evaluation and mitigation of seismic risks in their owned or leased buildings, and privately owned buildings on Federal land to ensure that all agencies have a balanced, agency-conceived and controlled seismic safety program.

The following is taken from EO 12699, "Seismic Safety of Federal and Federally Assisted or Regulated New Building Construction."

Section 1, "Requirements for Earthquake Safety of New Federal Buildings"—Each Federal agency responsible for the design and construction of each new Federal building shall ensure that the building is designed and constructed in accordance with appropriate seismic design and construction standards. This requirement pertains to all building projects where

development of detailed plans and specifications is initiated subsequent to the issuance of the order.

Section 2, "Federally Leased, Assisted, or Regulated Buildings"—Each Federal agency responsible for the construction and lease of a new building for Federal use shall ensure that the building is designed and constructed in accordance with appropriate seismic design and construction standards; each Federal agency assisting in the financing, through Federal grants or loans, or guaranteeing the financing, through loan or mortgage insurance programs, of newly constructed buildings shall plan, and initiate no later than three years subsequent to the issuance of EO 12699, *Seismic Safety of Federal and Federally Assisted or Regulated New Building Construction*, measures consistent with section 3(a) or EO 12699, to assure appropriate consideration of seismic safety; and each Federal agency with generic responsibility for regulating the structural safety of buildings shall plan to require use of appropriate seismic design and construction standards for new buildings within the agency's purview, implemented no later than three years subsequent to the issuance of EO 12699.

f. Define PC-1, PC-2, PC-3, and PC-4 and its relationship to nuclear facility design and the DSA.

The following is taken from DOE-STD-1021-93, Chg 1.

Performance category (PC)-4—Failure of an SSC during an NPH event could result in offsite release consequences greater than, or equal to, the unmitigated release from a large (>200 megawatt thermal) category A reactor severe accident.

PC-3—If an SSC is not covered in PC-4, and if its failure results in adverse release consequences greater than SC SSC EGs but much less than those associated with PC-4 SSCs. An SSC can be placed in PC-3 for improved performance if justified from cost-benefit considerations.

PC-2—An SSC may be placed in PC-2 if any of the following conditions apply:

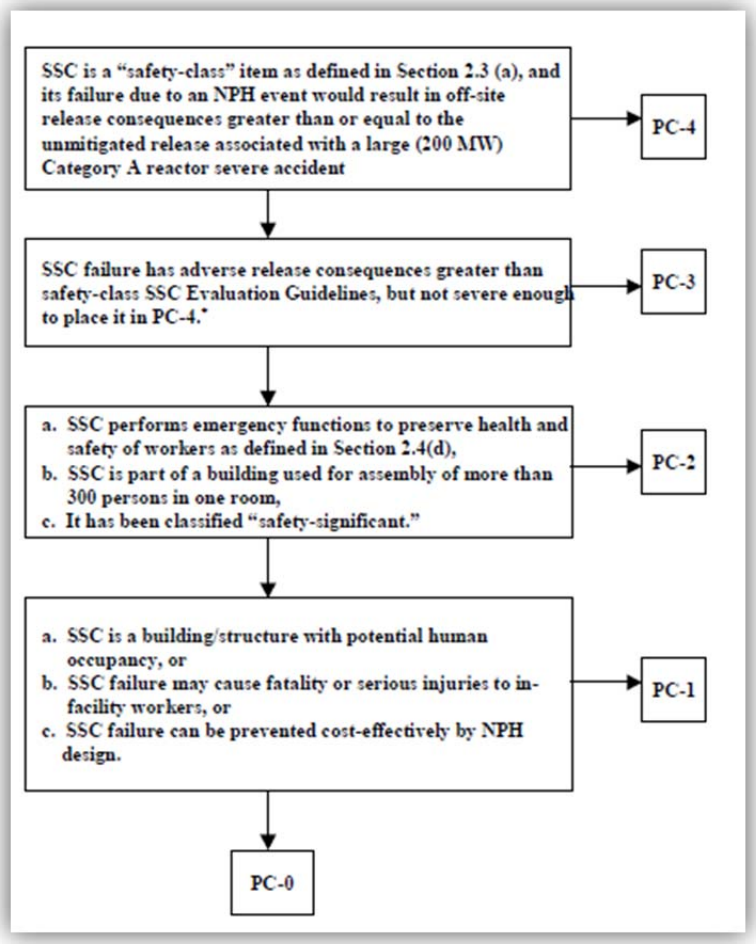
- The SSC's failure by itself or in combination with one or more SSCs may result in loss of function of any emergency handling, hazard recovery, fire suppression, emergency preparedness, communication, or power system that may be needed to preserve the health and safety of workers and visitors; this includes NPH-caused release of radioactive and toxic materials that would result in these consequences.
- The SSC is part of a building that is primarily used for assembly of more than 300 persons (in one room), and the SSC failure may adversely affect the life safety of the occupants.
- The SSC has been classified SS.

PC-1—An SSC that is not covered in the above PC categories shall be placed in preliminary PC-1 if any of the following conditions apply:

- It is a building/structure with potential human occupancy.
- The SSC's failure may cause a fatality or serious injuries to in-facility workers.
- The SSCs failure may cause damage that can be prevented or reduced cost-effectively by designing it to withstand NPH effects.

PC-0—An SSC that is not covered in the above PC categories may be placed in PC-0 if it is not important because of safety, mission, or cost considerations, and if it is more cost-effective to replace or repair it than to design it to withstand NPH effects; however, an SSC whose failure may have any adverse effect on the performance of a PC-1, PC-2, PC-3, or PC-4 shall not be placed in PC-0.

See figure 73 for the basic guidelines for preliminary NPH performance categorization of SSCs.



Notes:
SSC = Structure, system, or component
Use this figure only in the context of DOE-STD-1021-93, section 2; see subsection 2.5 for system interaction effects
*Refer to DOE G 420.1-2 for additional applications

Source: DOE-STD-1021-93

Figure 72. Basic guidelines for preliminary NPH performance categorization of SSCs

The following is taken from DOE-STD-1023-95.

For sites containing facilities with SSCs in only PC-1 or -2, it is sufficient to utilize seismic hazard maps from the current version of model building codes or national consensus standards if no site-specific probabilistic seismic hazard assessment (PSHA) has been

conducted for the sites. For sites that have site-specific PSHAs, the SSCs in PC-1 or -2 shall be evaluated or designed for the greater of the site-specific values or the model code values unless lower site-specific values are approved by DOE. Limitations on use of site-specific probabilistic natural phenomena assessments contained within model building codes shall be complied with.

For sites containing facilities with SSCs in PC-3 or -4, a site-specific PSHA shall be conducted to determine the design/evaluation basis earthquake (DBE).

Results of the flood screening analysis determine whether floods could impact DOE operations. For sites that could be exposed to flooding and do not meet the design basis, a comprehensive flood HAs is required. The need to perform a site comprehensive hazard assessment depends on the potential design basis flood level impact on the facilities for the flood
hazard exceedance probabilities. Guidelines to evaluate these impacts are provided in DOE-STD-1020-2012. These guidelines recommend the design basis for DOE facilities based on the following factors:

- Types of potential flood hazards
- PC
- Reliability of flood protection devices
- Acceptable level of risk

g. Discuss aspects of fire protection and fire hazards analysis and their relationship to nuclear facility design and the DSA.

The following is taken from DOE O 420.1C.

Fire Protection

A process must be established to ensure that fire protection program requirements are documented and incorporated in plans and specifications for design of new facilities and modifications to existing facilities.

Fire protection thresholds include the following:

- New facilities (non-relocatable) exceeding 5,000 sq. ft. of floor area must be of type I or II construction, as defined in the applicable building codes.
- Automatic fire suppression systems must be provided throughout new facilities exceeding 5,000 sq. ft. of floor area of where a maximum possible fire loss (MPFL) exceeds $5 million, unless the NFPA code(s) allow for specific relief within the facility.
- Automatic fire suppression systems must be provided throughout facilities where any of the following conditions exist:
  - Where required by safety basis document
  - Significant life safety hazards
  - Where fire may cause unacceptable mission or program interruption if automatic fire suppression systems are not provided
  - Where a modification to a facility would cause the MPFL to exceed $5 million
  - Where a modification causes a facility to exceed 5,000 sq. ft. of floor area

- For property protection, multiple fire protection approaches, such as a fire suppression system and a fire detection and alarm system, must be provided in areas where the MPFL exceeds $150 million.
- For property protection, fire areas must be established such that the MPFL for each fire area does not exceed $350 million. Fire area walls or other separation approaches may be used to meet this requirement.

Fire Hazard Analysis (FHA)

FHAs, using a graded approach, must be conducted for the following cases:

- All hazard category 1, 2, and 3 nuclear facilities and major modifications thereto
- Facilities that represent unique fire safety risks
- New facilities or modifications to existing facilities with value greater than $150 million
- When directed by the responsible DOE authority

The FHAs must be

- performed under the direction of a fire protection engineer (FPE);
- reviewed every three years by an FPE and revised as appropriate;
- revised when
  - changes to the facility structure or layout, processes, occupancy, safety basis documentation or baseline needs assessment impacts the analysis in the FHA;
  - a modification to an associated facility or process adds a significant new fire safety risk; or
  - the periodic (three-year) review identifies the need for changes; and
  - integrated into safety basis documentation.

Video 56. New design for nuclear reactors

http://live.wsj.com/video/new-designs-for-nuclear-reactors/7E36DDF0-6CF2-4340-9BD7-C333B051093E.html#!7E36DDF0-6CF2-4340-9BD7-C333B051093E

h. Identify and discus the strengths and weaknesses of methods utilized to analyze the initiation and propagation of fires and of their potential release of hazardous materials.

Consolidated Fire Growth and Smoke Transport Model (CFAST)

The following is taken from National Institute of Standards and Technology (NIST) Special Publication 1026.

CFAST is a model that predicts the environment within compartmented structures resulting from a fire prescribed by the user. CFAST was developed and is maintained primarily by the Fire Research Division of NIST.

CFAST has been developed for use in solving practical fire problems in fire protection engineering. It is intended for use in system modeling of building and building components. A prediction of flame spread or fire growth on objects is not modeled; rather, consequences of a specified fire are estimated. It is not intended for detailed study of flow within a compartment.

The most extensive use of the model is in fire and smoke spread in complex buildings. The efficiency and computational speed are inherent in the few computation cells needed for a zone model implementation. The use is for design and reconstruction of timelines for fire and smoke spread in residential, commercial, and industrial fire applications. Some applications of the model have been for design of smoke control systems.

- Compartments—CFAST is generally limited to situations where the compartment volumes are strongly stratified. Algorithms are provided for corridor flow, smoke detector activation, and detailed heat conduction through solid boundaries. This model does provide for non-rectangular compartments, although the application is intended to be limited to relatively simple spaces. There is no intent to include complex geometries where a complex flow field is a driving force. For these applications, computational fluid dynamics models are appropriate.
- Gas layers—There are limitations inherent in the assumption of stratification of the gas layers. The zone model implies a sharp boundary between the upper and lower layers, whereas in reality, the transition is typically over about 10 percent of the height of the compartment and can be larger in weakly stratified flow. While it is possible to make predictions within 5 percent of the actual temperatures of the gas layers, this is not the optimum use of the model. It is more properly used to make estimates of fire spread, smoke detection and contamination, and life safety calculations.
- Heat release rate—CFAST does not predict fire growth on burning objects. The model does include the ability to limit the specified burning based on available oxygen. There are limitations inherent in the assumptions used in the application of the empirical models. As a general guideline, the heat release should not exceed about the maximum volumetric heat transfer coefficient 1 MW/m^3. This is a limitation on the numerical routines attributable to the coupling between gas flow and heat transfer through boundaries. The inherent two-layer assumption is likely to break down well before this limit is reached.
- Radiation—Because the model includes a sophisticated radiation model and ventilation algorithms, it has further use for studying building contamination through the ventilation system, as well as the stack effect and the effect of wind on air circulation in buildings. Radiation from fires is modeled with a simple point source approximation. This limits the accuracy of the model near fire sources. Calculation of radiative exchange between compartments is not modeled.
- Ventilation and leakage—In a single compartment, the ratio of the area of vents connecting one compartment to another to the volume of the compartment should not exceed roughly 2 m. This is a limitation on the plug flow assumption for vents.
- Thermal properties—The accuracy of the model predictions is limited by how well the user can specify the thermophysical properties. There is a higher level of uncertainty of the predictions if the properties of real materials and real fuels are unknown or difficult to obtain, or the physical processes of combustion, radiation, and heat transfer are more complicated than their mathematical representations in CFAST.

User feedback indicates that using CFAST to predict the transport of heat and combustion products from a prescribed fire is straightforward, easily and quickly accomplished, and the results are within expectations.

Toolbox and Toolbox-Equivalent Software Applications

The following is taken from DOE G 414.1-4.

Toolbox codes represent a small number of standard computer models or codes supporting DOE safety analysis. These codes have widespread use and are of appropriate qualification for use within DOE. The toolbox codes are acknowledged as part of DOE's safety software central registry. These codes are verified and validated and constitute a safe harbor methodology. The following six widely applied safety analysis computer codes have been designated as toolbox codes:

1. Areal Locations of Hazardous Atmospheres (ALOHA) (chemical dispersion analysis)
2. CFAST (fire analysis)
3. Emergency Prediction Information Code (EPIcode) (chemical dispersion analysis)
4. Generalized Environmental Radiation Dosimetry Software System—Hanford Dosimetry System (GENII) (radiological dispersion analysis)
5. Methods for Estimation of Leakages and Consequences of Releases (MELCOR) (leak path factor [LPF] analysis)
6. MELCOR Accident Consequence Code System 2 (MACC2) (radiological dispersion analysis)

The current designated toolbox codes and any software recognized in the future as meeting the toolbox equivalency criteria are no different from other custom developed safety software as defined in DOE G 414.1-4, *Safety Software Guide for Use with 10 CFR 830 Subpart A, Quality Assurance Requirements, and DOE O 414.1D, Admin Chg 1, Quality Assurance*, section 2.1.

i. Identify and discuss the methods used to determine the seismic hazard level to be used in design.

The following is taken from DOE-STD-1020-2012.

The guidelines and criteria for design categorization and limit state of SSCs subjected to seismic hazards shall be the same as those in ANSI/ANS-2.26-2004, *Categorization of Nuclear Facility Structures, Systems, and Components for Seismic Design*, except that consequence evaluation criteria shall be as defined in DOE-STD-1020-2002 (Archived), table 2-1. This table is replicated as table 7.

Table 7. Seismic performance categories and seismic hazard exceedance levels

Performance Category	Mean Seismic Hazard Exceedance Levels, P_H	Remarks
0	No Requirements	
1	Follow IBC 2000 in its entirety*	Use IBC 2000 seismic use group I criteria 2/3 MCE ground motion
2	Follow IBC 2000 in its entirety*	Use IBC 2000 seismic use group III criteria 2/3 MCE ground motion with importance factor of 1.5
3	$4x10^{-4}$ ($1x10^{-3}$)[1]	Establish DBE per DOE-STD-1023 analysis per DOE-STD-1020
4	$1x10^{-4}$ ($2x10^{-4}$)[1]	Establish DBE per DOE-STD-1023 analysis per DOE-STD-1020

*Based on maximum considered earthquake (MCE) ground motion—generally 2 percent exceedance probability in 50 years from the seismic hazard maps, modified to account for site effects. P_H=$4x10^{-4}$

[1]For sites such as Los Alamos National Laboratory, Sandia National Laboratory-Livermore, Stanford Linear Accelerator Center, Lawrence Berkeley National Laboratory, and Energy Technology Engineering Center that are near tectonic plate boundaries.

Source: DOE-STD-1020-2002 (Archived)

PC-2 and lower SSCs may be designed or evaluated using the approaches specified in International Code Consortium *International Building Code 2000*, seismic provisions. However, for PC-3 or higher, the seismic evaluation must be performed by a dynamic analysis approach.

j. Identify and discuss the methods used to assess the structural response of structures and determine whether safety systems may be expected to remain functional following an earthquake of postulated intensity.

The following is taken from Wikipedia, *Seismic Analysis*.

Seismic analysis is a subset of structural analysis and is the calculation of the response of a building, or non-building structure to earthquakes. It is part of the process of structural design, earthquake engineering, or structural assessment and retrofit in regions where earthquakes are prevalent.

Structural analysis methods can be divided into the following five categories:

1. Equivalent static analysis
2. Response spectrum analysis
3. Linear dynamic analysis
4. Nonlinear static analysis
5. Nonlinear dynamic analysis

Equivalent Static Analysis

This approach defines a series of forces acting on a building to represent the effect of earthquake ground motion, typically defined by a seismic design response spectrum. It assumes that the building responds in its fundamental mode. For this to be true, the building must be low-rise and must not twist significantly when the ground moves. The response is read from a design response spectrum, given the natural frequency of the building. The applicability of this method is extended in many building codes by applying factors to account for higher buildings with some higher modes, and for low levels of twisting. To account for the effects due to yielding of the structure, many codes apply modification factors that reduce the design forces.

Response Spectrum Analysis

This approach permits the multiple modes of response of a building to be taken into account. This is required in many building codes for all except for very simple or very complex structures. The response of a structure can be defined as a combination of many special modes that in a vibrating string correspond to the harmonics. Computer analysis can be used to determine these modes for a structure. For each mode, a response is read from the design spectrum, based on the model frequency and the modal mass, and they are then combined to provide an estimate of the total response of the structure.

The result of a response spectrum analysis using the response spectrum from a ground motion is typically different from what would be calculated directly from a linear dynamic analysis using that ground motion directly, since phase information is lost in the process of generating the response spectrum.

In cases where structures are either too irregular, too tall, or of significance to a community in disaster response, the response spectrum approach is no longer appropriate, and more complex analysis is often required, such as nonlinear static or dynamic analysis.

Video 57. Response spectrum analysis
http://www.youtube.com/watch?v=UYsisjqOaiI

Linear Dynamic Analysis

Static procedures are appropriate when higher mode effects are not significant. This is generally true for short, regular buildings. Therefore, for tall buildings, buildings with torsional irregularities, or non-orthogonal systems, a dynamic procedure is required. In the linear dynamic procedure, the building is modeled as a multi degree of freedom system with a linear elastic stiffness matrix and an equivalent viscous damping matrix.

The seismic input is modeled using either modal spectral analysis or time history analysis; however, in these cases, the corresponding internal forces and displacements are determined using linear elastic analysis. The advantage of these linear dynamic procedures with respect to linear static procedures is that higher modes can be considered. However, they are based on linear elastic response and hence the applicability decreases with increasing nonlinear behavior that is approximated by global force reduction factors.

In linear dynamic analysis, the response of the structure to ground motion is calculated in the time domain, and all phase information is therefore maintained. Only linear properties are assumed. The analytical method can use modal decomposition as a means of reducing the degrees of freedom in the analysis.

Nonlinear Static Analysis

In general, linear procedures are applicable when the structure is expected to remain nearly elastic for the level of ground motion or when the design results in nearly uniform distribution of nonlinear response throughout the structure. As the performance objective of the structure implies greater inelastic demands, the uncertainty with linear procedures increases to a point that requires a high level of conservatism in demand assumptions and acceptability criteria to avoid unintended performance. Therefore, procedures incorporating inelastic analysis can reduce the uncertainty and conservatism.

This approach is also known as "pushover" analysis. A pattern of forces is applied to a structural model that includes nonlinear properties (such as steel yield), and the total force is plotted against a reference displacement to define a capacity curve. This can then be combined with a demand curve. This essentially reduces the problem to a single degree of freedom (SDOF) system.

Nonlinear static procedures are equivalent SDOF structural models and represent seismic ground motion with response spectra. Story drifts and component actions are related subsequently to the global demand parameter by the pushover or capacity curves that are the basis of the nonlinear static procedures.

Nonlinear Dynamic Analysis

Nonlinear dynamic analysis uses the combination of ground motion records with a detailed structural model; therefore, is capable of producing results with relatively low uncertainty. In nonlinear dynamic analyses, the detailed structural model subjected to a ground motion record produces estimates of component deformations for each degree of freedom in the model and the modal responses are combined using schemes such as the square-root-sum-of-squares.

In nonlinear dynamic analysis, the nonlinear properties of the structure are considered as part of a time domain analysis. This approach is the most rigorous, and is required by some building codes for buildings of unusual configuration or of special importance. However, the calculated response can be very sensitive to the characteristics of the individual ground motion used as seismic input; therefore, several analyses are required using different ground motion records to achieve a reliable estimation of the probabilistic distribution of structural response. Since the properties of the seismic response depend on the intensity, or severity, of the seismic shaking, a comprehensive assessment calls for numerous nonlinear dynamic analyses at various levels of intensity to represent different possible earthquake scenarios. This has led to the emergence of methods like the incremental dynamic analysis.

Video 58. Impact of ground motion modification process in dynamic analysis
http://www.youtube.com/watch?v=i-p_eMBVvyU

k. Identify the methods for evaluating the tolerance of structures and systems for natural phenomenon.

The following is taken from DOE-STD-1020-2012.

Seismic evaluation for existing facilities would be similar to evaluations performed for new designs except that a single as-is configuration is evaluated instead of several configurations in an iterative manner. The evaluation of existing facilities for seismic hazards can result in a number of options based on the evaluation results. If the existing facility can be shown to meet the design and evaluation criteria presented in DOE-STD-1020-2012, section 3, and good seismic design practices had been employed, the facility would be judged to be adequate for potential seismic hazards to which it might be subjected. If the facility does not meet the seismic evaluation criteria, an analysis should be conducted to determine appropriate action. Several alternatives may be considered including the following:

- If an existing SSC is close to meeting the criteria, a slight increase in the annual risk to NPHs can be allowed within the tolerance of meeting the target performance goals. Note that reduced criteria for seismic evaluation of existing SSCs is supported in ICSSC RP-8, *Standards of Seismic Safety for Existing Federally Owned and Leased Buildings* documents. As a result, some relief in the criteria may be allowed by performing the evaluation using higher hazard exceedance probability as permitted in DOE-STD-1020-2012, section 9.3.
- The SSC may be strengthened such that its seismic resistance capacity is sufficiently increased to meet these seismic criteria. When upgrading is required, it should be designed for the current design criteria.
- The usage of the facility may be changed such that it falls within a less hazardous PC and, consequently, less stringent seismic requirements.
- It may be possible to conduct the aspects of the seismic evaluation in a more rigorous manner that removes conservatism such that the SSC may be shown to be adequate. Alternatively, a probabilistic assessment might be undertaken to demonstrate that the performance goals can be met.

Priorities should be established on the basis of PC, cost of strengthening, and margin between as-is SSC capacity and the capacity required by the criteria in DOE-STD-1020-2012. For SSCs that are within ten percent of meeting the criteria, the risk from noncompliance is likely to be small and it may not be cost-effective to strengthen the SSC in order to obtain a small reduction in risk. As a result, as specified, some relief from the criteria is permitted for evaluation of existing SSCs. It is permissible to perform such evaluations using NPH exceedance probability of twice the value specified for new design, provided that the resulting reduction in the hazard level is less than, or equal to, twenty percent. This amount of relief is within the tolerance of meeting the performance goals and is only a minor adjustment of the corresponding NPH design and evaluation criteria. In addition, it is consistent with the intent of the EO 12941 developed by the ICSSC. When upgrading becomes necessary, the design should be based on the design criteria for new SSCs defined in DOE-STD-1020-2012.

The following is taken from DOE-STD-1020-2002 (Archived).

The design and evaluation criteria for SSCs in PCs 0, 1, and 2 are similar to those given in model building codes. PC-0 recognizes that for certain lightweight equipment items, furniture, etc., and for other special circumstances where there is little or no potential impact on safety, mission, or cost, design or evaluation for NPHs may not be needed. Assignment of an SSC to PC-0 is intended to be consistent with, and not take exception to, model building code NPH provisions.

PC-1 criteria include no extra conservatism against NPHs beyond that in model building codes that include earthquake, wind, and flood considerations. PC-2 criteria are intended to maintain the capacity to function and to keep the SSC operational in the event of NPHs. Model building codes would treat hospitals, fire, and police stations, and other emergency-handling facilities in a similar manner to DOE-STD-1020-2002 PC-2 NPH design and evaluation criteria.

PC-3 and -4 SSCs handle significant amounts of hazardous materials or have significant programmatic impact. Damage to these SSCs could potentially endanger worker and public safety and the environment or interrupt a significant mission. As a result, it is very important for these SSCs to continue to function in the event of NPHs, such that the hazardous materials may be controlled and confined. For these categories, there must be a very small likelihood of damage due to NPHs. DOE-STD-1020-2002 NPH criteria for PC-3 and higher SSCs are more conservative than requirements found in model building codes and are similar to Department of Defense (DoD) criteria for high-risk buildings and NRC criteria for various applications as illustrated in table 8. Table 8 illustrates how DOE-STD-1020-2002 criteria for the PCs defined in DOE O 420.1C and the associated guides compare with NPH criteria from other sources.

Video 59. A new chapter of earthquake resistance
http://www.youtube.com/watch?v=RCk8YuO37jI

Table 8. Comparison of performance categories from various sources

Source	SSC Categorization			
DOE-STD-1020—*DOE Natural Phenomena Hazard Criteria*	1	2	3	4
Uniform Building Code	General facilities	Essential facilities	-	-
DoD *Tri-Service Manual for Seismic Design of Essential Buildings*	-	-	High risk	-
Nuclear Regulatory Commission	-		NRC fuel facilities	Evaluation of existing reactors

Source: DOE-STD-1020-2002 (Archived)

I. Discuss the importance of integrating safety analysis development with facility design.

The following is taken from DOE O 420.1C.

Safety must be integrated into the design early in, and throughout the design process through use of DOE-STD-1189-2008.

Safety analyses must be used to

- identify SC SSCs and SS SSCs needed to fulfill the safety functions to prevent and/or mitigate DBAs, including natural and man-made hazards and events;
- identify the safety functional requirements of the SC SSCs and SS SSCs; and
- identify SACs needed to fulfill safety functions.

The nuclear facility design must include multiple layers of protection to prevent or mitigate the unintended release of radioactive materials in the environment.

DID must include all of the following:

- Choosing an appropriate site
- Minimizing the quantity of material-at-risk
- Applying conservative design margins
- Applying QA
- Using successive/multiple physical barriers for protection against radioactive releases
- Using multiple means to ensure safety functions are met by
  - controlling processes
  - maintaining processes in safe status
  - providing preventive and/or mitigative controls for accidents with the potential for radiological releases
  - providing the means for monitoring facility conditions to support recovery from upset or accident conditions

- Using equipment in combination with ACs that
  - restrict deviation from normal operations
  - monitor facility conditions during and after an event
  - provide for response to accidents to achieve a safe condition
- Providing means to monitor accident releases as required for emergency response
- Establishing emergency plans for minimizing the effects of an accident

Hazard category 1, 2, and 3 nuclear facilities with uncontained radioactive materials must have the means to confine the uncontained radioactive materials to minimize their potential release in facility effluents during normal operations and during and following accidents, up to and including DBAs. Confinement design must include the following:

- For a specific nuclear facility, the number, arrangement, and characteristics of confinement barriers as determined on a case-by-case basis.
- The type, quantity, form, and conditions for dispersing the radioactive material in the confinement system design.
- An active confinement ventilation system as the preferred design approach for nuclear facilities with potential for radiological release. Alternate confinement approaches may be acceptable if a technical evaluation demonstrates that the alternate confinement approach results in very high assurance of the confinement of radioactive materials.

20. Nuclear safety specialists must demonstrate a working level knowledge of the TSRs as described in 10 CFR 830.205, "Technical Safety Requirements," and DOE G 423.1-1, *Implementation Guide for Use in Developing Technical Safety Requirements*, with respect to their impact on the Department's nuclear safety.

[Note: DOE G 423.1-1 has been superseded by DOE G 423.1-1A.]

a. Discuss the purpose of TSRs.

The following is taken from DOE G 423.1-1A.

TSRs define the performance requirements of SSCs and identify the safety management programs used by personnel to ensure safety. TSRs 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 HA of the activities to be performed and identification of the potential sources of safety issues. Safety analyses to identify and analyze a set of bounding accidents that take into account all potential causes of releases of radioactivity contribute to development of TSRs.

Through analysis of the encompassing bounding accidents, the necessary safety systems and accident mitigating systems are identified and their characteristics are defined. Flowing from the analyses is information that provides the bases for controls, limits, and conditions for operation, known as TSRs.

b. Describe the responsibilities of contractors authorized to operate nuclear facilities for TSRs.

The following is taken from DOE G 423.1-1A.

10 CFR 830.205 requires DOE contractors responsible for hazard category 1, 2, and 3 DOE nuclear facilities to develop TSRs. These TSRs 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 this rule, these may be a need to establish TSRs for safe operation for radiological facilities. The TSR rule requires contractors to prepare and submit TSRs for DOE approval.

Contractors, in the preparation of DSA, identify how the safety requirements of 10 CFR 830 apply to a specific facility, and describe how the contractor undertakes to design, build, and operate the facility to be in conformance with the applicable statutes, DOE rules, and directives to ensure facility safety. The analysis of operations and accidents defines the limits of safe operations, identifies the required performance of SC SSCs and SS SSCs, and describes any ACs or procedures that are necessary to meet the safety criteria for the facility.

c. Define the following terms and discuss the purpose of each:

- **Safety limits**
- **Operating limits**
- **Limiting control settings**
- **Limiting conditions for operations**
- **Surveillance requirements**
- **Administrative controls**
- **Specific administrative controls**

The following definitions are taken from DOE G 423.1-1A unless otherwise stated.

Safety Limits (SLs)
SLs are 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 EGs.

Operating Limits (OLs)
OLs, to include LCSs and LCOs, protect against exceeding SLs and can ensure availability of SS SSCs important to worker safety.

Limiting Control Settings (LCSs)
LCSs define the settings on safety systems that control process variables to prevent exceeding an SL.

Limiting Conditions for Operations (LCOs)
LCOs define the limits that represent the lowest functional capability or performance level of safety SSCs required to perform an activity safely.

Surveillance Requirements (SRs)
SRs are used to ensure operability or availability of the safety SSCs identified in the OLs. SRs are most often used with LCOs to periodically validate the operability of active systems or components that are subject to a limiting condition.

ACs
ACs are the provisions relating to organization and management, procedures, recordkeeping, reviews, and audits necessary to ensure safe operation of the facility.

SAC
The following is taken from DOE-STD-1186-2004.

SACs are ACs that provide a specific preventive or mitigative function for accident scenarios identified in the DSA where the safety function has importance similar to, or the same as, the safety function of a safety SSC.

d. Describe the general content of each of the following sections of the TSR:
- **Use and application**
- **Basis**
- **Design features**

The following is taken from DOE G 423.1-1A.

Use and Application
The use and application 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. Include a note on the first page of the list stating that defined terms appear in uppercase type throughout the TSR.
- Operational modes (reactors)—Define the operational modes for reactor facilities and state that in the interest of uniformity, the operational conditions or modes listed are preferred, and an attempt should be made to fit each reactor facility in this scheme.
- Operational modes (nonreactor nuclear facilities)—Define the operational modes for nonreactor facilities and state that in the interest of uniformity, the operational conditions and modes listed are preferred, and an attempt should be made to fit each nonreactor nuclear facility in this scheme. If, however, a nonreactor nuclear facility cannot be made to fit, modes may be defined as needed, provided the definitions are clear, and there are definite lines of demarcation between modes.
- Frequency notation—The frequency notations, as used in the surveillances and elsewhere, should be defined when included in the TSR.

Basis
The bases appendix provides summary statements of the reasons for the SLs, LCSs, LCOs, and associated SRs. The bases show how the numeric values, the conditions, the surveillances, and the action statements fulfill the purpose derived from the safety documentation. The primary purposes for describing the bases of each requirement are to ensure future changes to the requirements will not affect its original intent or purpose by

invalidating the safety analysis and to aid in understanding why the requirement exists. The bases appendix should reference the more specific detailed safety analyses related to the TSR and the derivation of TSR section of the DSA for other related analyses discussed in the DSA.

Design Features

A design features section should be included with the TSR. 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:

1. Vital passive safety SSCs such as piping, vessels, supports, structures, and containers.
2. 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 definition and implementation principles for the term "operability" as used in a TSR.

The following is taken from DOE G 423.1-1A.

Operability embodies the principle that a system, subsystem, train, component, or device can perform its safety function(s) only if all necessary support systems are capable of performing their related support functions. This definition extends the requirements of an LCO for those systems that directly perform a specified safety function to those that perform a required support function.

A system or component can be degraded but still operable if it remains capable of performing its required safety function at the level assumed in the accident analysis. If systems, components, or equipment are observed to be functioning but under stress, judgment must be used concerning a declaration of inoperability.

The following are general principles of operability:

- General principle 1—A system is considered operable as long as there exists assurance that it is capable of performing its specified safety function(s).
- General principle 2—A system can perform its specified safety function(s) only when all of its necessary support systems are capable of performing their related support functions.
- General principle 3—When all systems designed to perform a certain safety function are not capable of performing that safety function, a loss of function condition exists.
- General principle 4—When a system is determined to be incapable of performing its intended safety function(s), the declaration of inoperability should be immediate.

f. Discuss the relationship of functional requirements and performance criteria to the TSR.

The following is taken from DOE G 423.1-1A.

TSRs define the performance requirements of SSCs and identify the safety management programs used by personnel to ensure safety. TSRs 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 HA of the activities to be performed and identification of the potential sources of safety issues. Safety analyses to identify and analyze a set of bounding accidents that take into account all potential causes of releases of radioactivity also contribute to development of TSRs.

Through analyses of the encompassing bounding accidents, the necessary safety systems and accident mitigating systems are identified and their characteristics are defined. Flowing from the analyses is information that provides the bases for controls, limits, and conditions for operation, known as TSRs. TSRs explicitly show this relationship. The content of the DSA must remain valid so that the safety basis of the facility, as implemented in operations through the TSR, remains valid. Therefore, there is commitment to the process of USQs regarding any proposed change to the facility or its operations as described in the DSA. Likewise, all changes to the TSR bases presented in the DSA should be incorporated in the TSRs to ensure the information contained reflects the current safety basis of the facility.

Any proposed revision to a TSR should be examined to ensure the basis for the change is supported in the DSA. The TSR rule requires that such revisions be submitted to DOE for review with the basis for the proposed change. The change to the TSR must be approved by DOE before it is implemented.

g. Discuss the conditions that constitute a violation of the TSR and state the reporting requirements should a violation occur.

The following is taken from DOE G 423.1-1A.

Although the TSR elements have an importance hierarchy, a TSR violation can occur for each type of TSR. Violations of a TSR occur as a result of the following circumstances:

- Exceeding an SL
- Failure to complete an action statement within the required time limit following exceeding an LCS or failing to comply with an LCO
- Failure to perform surveillance within the required time limit
- Failure to comply with an AC statement

Failure to comply with an AC statement is a TSR violation when either the AC is directly violated, as would be the case with not meeting minimum staffing requirements, for example, or the intent of a referenced program is not fulfilled. To quality as a TSR violation, the failure to meet the intent of the referenced program would need to be significant enough to render the DSA summary invalid. TSR violations involving SLs require the facility to begin immediately to go to the most stable, safe condition attainable, including total shutdown.

A grace period is sometimes provided to perform a missed surveillance to provide time for the performance of the missed surveillance; thereby avoiding the need for a facility to take immediate, possible unnecessary corrective action. Entering the grace period remains a TSR violation even though an immediate corrective action may not be required.

Reporting of all TSR violations should be made in accordance with the provisions of DOE O 231.1B, admin Chg 1, *Environment, Safety and Health Reporting*. 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.

h. Discuss the requirements for ACs of the TSR.

The following is taken from DOE G 423.1-1A.

ACs are the provisions relating to organization and management, procedures, recordkeeping, reviews, and audits necessary to ensure safe operation of the facility. ACs may include reporting deviations from TSRs (i.e., exceeding LCOs, LCSs, or SRs, or violation of a TSR), staffing requirements for facility positions important to safe operation of the facility, ACs of the criticality safety program and commitments to safety management programs important to worker safety.

In general, the ACs should document all those administrative functions that are required to meet facility safety criteria as identified in the DSA, including commitments to safety management programs. It is expected that the ACs will be tailored to the facility activities and the hazards identified in the DSA. This tailoring should be a direct result of the DSA, but it may result from institutional requirements that address many facilities. As a general practice, safety controls for individual accident scenarios based on engineered SSCs are preferred to ACs because they are usually more reliable and more predictable.

The tendency to use ACs as an expedient alternative to an LCO or LCS should be avoided when possible. Efforts should be made to use engineered SSCs whenever possible for controlling the likelihood and consequences of accidents. ACs should be considered for DID rather than the primary or redundant controls. While ACs may be acceptable for ensuring safe operation, their generally lower reliability, compared with engineered controls, should be evaluated carefully when choosing safety measures for long-term hazardous activities.

Human actions, taken either in response to an event or taken proactively to establish desired conditions, are subject to errors of omission or commission. Sets of ACs are prone to CCF. The following attributes that can be tailored as appropriate, can increase reliability:

- Use of reader/worker/checker systems
- Independent verification
- Positive feedback systems
- Human factor analysis
- Operator training and certification
- Continuing training and requalification
- Abnormal event response drills
- Ergonomic considerations in procedures

When invoking ACs for control of accident scenarios, the preceding attributes, appropriate to the consequences of the accidents they are intended to prevent, should be considered and invoked.

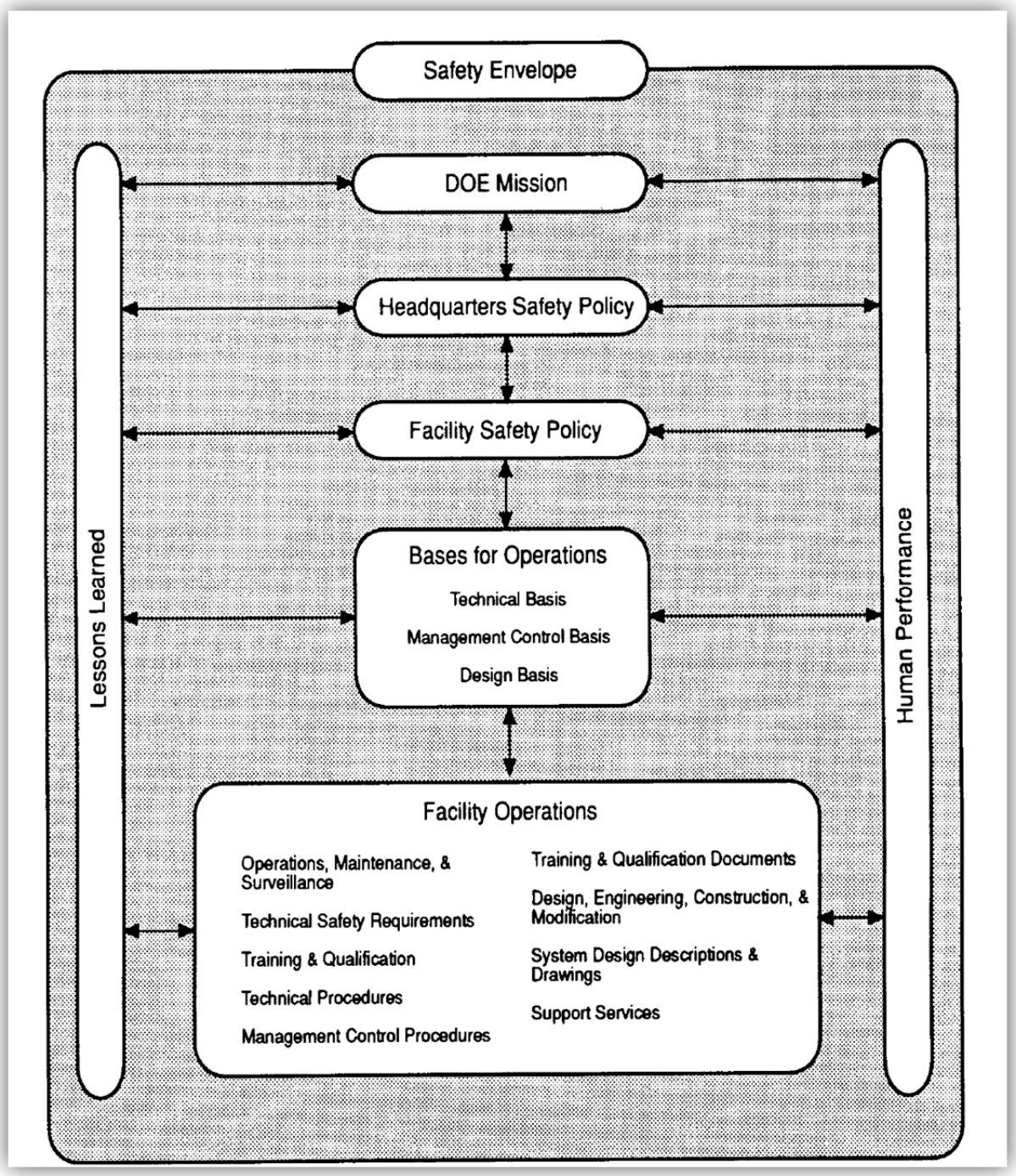
i. Discuss the role of DSA in selecting a TSR and the respective flowdown.

The following is taken from DOE G 423.1-1A.

The DSA required by 10 CFR 830.204 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.

The following is taken from DOE-STD-1029-92, Chg 1.

The overall safe operation of the facility depends on the structured interrelationship among DOE requirements and guidance, the bases documentation, and the facility's procedures—generally referred to as its safety envelope. A facility's procedures define how requirements, management philosophies and strategies, and technical knowledge will be integrated and applied to performing work in the facility. Figure 74 illustrates a facility's safety envelope, and shows how requirements, guidance, and technical and managerial constraints flow down through a facility's basis documentation and are incorporated in the facility's operations.



Source: DOE-STD-1029-92, Chg 1

Figure 73. Safety envelope diagram

j. Discuss the requirements for emergency actions that depart from the approved TSR.

The following is taken from DOE G 423.1-1A.

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. 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. If emergency action is taken, a verbal notification should be made to the responsible head of the field element and a written report made to the cognizant secretarial officer within twenty-four hours.

k. Discuss the provisions a contractor may follow to develop alternatives to TSR for environmental restoration activities.

The following is taken from DOE G 423.1-1A.

A contractor for an environmental restoration activity may follow the provisions of 29 CFR 1910.120, "Hazardous Waste Operations and Emergency Response," or 29 CFR 1926.65, "Hazardous Waste Operations and Emergency Response," (HAZWOPER) for construction activities (see 10 CFR 830, appendix A to subpart B, table 2) to develop the appropriate hazard controls (rather than this TSR guidance) provided the activity involves either 1) work not done within a permanent structure or 2) the decommissioning of a facility with only low-level residual radioactivity. Implicit in this guidance is an understanding that reasonable efforts to remove radioactive systems, components, and stored materials have been completed and that the work does not prudently require the use of active safety systems or components designed to prevent or mitigate the accidental release of hazardous radioactive materials. DOE-STD-1120-2005, *Integration of Environment, Safety, and Health Into Facility Disposition Activities*, provides guidance that must be considered in the development of TSRs. DOE G 423.1-1A, appendix D, "Performance of Implementation Verification Reviews (IVRs) of Safety Basis Controls," provides guidance on performance of IVRs of safety basis controls.

l. Discuss the requirements for the contractor to change the TSR.

The following is taken from DOE Order 5480.22, Chg 2 (Archived).

The TSR shall be kept current at all times so that it reflects the facility as it exists and as it is analyzed in SARs. Contractors will determine whether revisions to the TSRs are required upon originating or proposing a revision to a SAR, and, if so, will prepare revisions and submit them with their basis for PSO approval concurrent with the revisions to the SAR. The TSR must be approved prior to facility or facility practice change. To assure that the TSR is current, it will be reviewed at least annually along with the facility SAR.

m. Discuss the application of the graded approach relative to the TSR.

The following is taken from DOE G 423.1-1A.

The graded approach is not directly applicable to the TSRs required by 10 CFR 830.205. However, the graded approach is specified for DSAs required by 10 CFR 830.204. Thus, the level of detail in the DSA and the number of safety parameters identified in the DSA section deriving the TSRs will have a direct effect on the number and type of resulting TSRs.

Mandatory Performance Activities:

a. Participate on a combination of at least five safety basis document reviews and/or review of safety basis documents and/or performance of a field walkdown of a safety related SSC (including the associated surveillance requirements and LCO/limiting condition statements) to determine proper derivation of SL/LCS/LCO/SAC/AC, including associated surveillance requirements.

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

21. Nuclear safety specialists must demonstrate a working level knowledge of 10 CFR 830.203, "Unreviewed Safety Question Process," and DOE G 424.1-1A, *Implementation Guide for Use in Addressing Unreviewed Safety Question Requirements*, with respect to their impact on the Department's nuclear safety.

[Note: DOE G 424.1-1A has been superseded by DOE G 424.1-1B, Admin Chg 2.]

a. Discuss the purpose of the USQ process.

The following is taken from DOE G 424.1-1B, Admin Chg 2.

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.

b. Discuss the reasons for performing a USQ determination.

The following is taken from DOE G 424.1-1B, Admin Chg 2.

10 CFR 830.203 requires an unreviewed safety question determination (USQD) for the following:

- Temporary or permanent changes in nuclear facilities as described in the existing safety analysis text, drawing, or other information that is part of the facility safety basis
- Temporary or permanent changes to procedures that are identified in the facility DSA
- Tests or experiments not described in the existing safety analyses

- When a contractor identifies or is informed of a situation that indicates the safety analyses that support the DOE-approved safety basis may not be bounding or may be otherwise inadequate

c. Define the following terms:

- **Discrepant as-found condition**
- **Potential inadequacy in the safety analysis**
- **Proposed change**

The following are taken from DOE G 424.1-1B, Admin Chg 2.

Discrepant As-Found Condition
A discrepant as-found condition is a condition where the analysis does not match the current physical configuration, is inappropriate, or contains errors. Analytical errors might involve using incorrect input values, invalid assumptions, improper models, or calculation errors.

Potential Inadequacy in the Safety Analysis
A potential for inadequate safety analysis (PISA) may result from situations that indicate the safety basis may not be bounding or may be otherwise inadequate; for example, discrepant as-found conditions, operational events, or the discovery of new information. The main consideration is that the safety analysis does not match the current physical configuration, or the safety analysis is inappropriate or contains errors.

Proposed Change
For the purposes of DOE G 424.1-1B, change means any change to procedures or equipment (including prior undocumented changes), any new tests or experiments, or any new information that has the potential to invalidate the safety basis. The contractor must evaluate any proposed change to ensure that it will not affect the safety basis of the facility either explicitly or implicitly.

d. Define the conditions for a USQ.

The following is taken from DOE G 424.1-1A (Archived).

If the answer to any of the following questions is yes, the change is considered a USQ:

1. Could the proposed change increase the probability of an accident previously evaluated in the facility's existing safety analysis?
2. Could the proposed change increase the consequences (to workers or the public) of an accident previously evaluated in the facility's existing safety analyses?
3. Could the proposed change increase the probability of a malfunction of equipment important to safety previously described in the facility's existing safety analyses?
4. Could the proposed change increase the consequences of a malfunction of equipment important to safety described in the facility's safety analyses?
5. Could the proposed change create the possibility of an accident of a different type than any previously evaluated in the facility's existing safety analyses?

6. Could the proposed change create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the facility's existing safety and analyses?
7. Could the proposed change reduce a margin of safety?

e. Describe the responsibilities of contractors authorized to operate 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; and
- 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 safety analyses or a possible reduction in the margin of safety as defined in the TSR.

Potential Inadequacy of Safety Analyses (PISA)

The following is taken from 10 CFR 830.203.

If a contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility discovers or is made aware of a potential inadequacy of the DSA, it must

- take action, as appropriate, to place or maintain the facility in a safe condition until an evaluation of the safety of the situation (ESS) is completed;
- notify DOE of the situation;
- perform a USQD and notify DOE promptly of the results; and
- submit the ESS to DOE prior to removing any operational restrictions initiated to meet 10 CFR 830.203(g)(1).

g. Discuss the qualification and training requirements for personnel who implement the USQ process.

The following is taken from DOE G 424.1-1B, Admin Chg 2.

Implementing procedures should establish the training and qualifications for personnel performing the USQ process such as education background, years and/or types of work experience and knowledge of the facility, understanding of DOE facility safety basis requirements, and familiarity with the facility-specific safety basis.

All personnel responsible for preparing, reviewing, or approving USQ documents should receive training on the application of 10 CFR 830.203, including any facility-specific procedures. The recommended interval for retraining is every two years.

The contractor should maintain a list of those personnel who are currently qualified to perform the USQ process.

h. Discuss the actions to be taken if it is determined that a potential inadequacy in the safety analysis is involved.

The following is taken from DOE G 424.1-1B, Admin Chg 2.

When a degraded or nonconforming SSC is identified as a PISA, the contractor must first take action, as appropriate, to place or maintain the facility in a safe condition. A safe condition may include continued facility operation if, although a degraded or nonconforming SSC is not fully qualified, the impact on safe facility operations is judged to be acceptable, possibly aided by operational restrictions and the TSRs are still being met in terms of required operable equipment for the given mode of operations and associated actions. If a SSC identified in a TSR is determined to be inoperable, then in accordance with the facility TSR, the action statement(s) in the TSR must be implemented. The TSR action statement may direct the facility operator to go to a mode where the piece of equipment is not required or the facility operator may choose to take this action even though the TSRs do not explicitly direct it.

There may be situations where an SSC has been degraded such that there is a loss of quality or functional capability or a nonconforming condition may exist with the SSC or its documentation, but the SSC has not been determined to be inoperable. These situations may constitute a PISA. When a degraded or nonconforming SSC is identified as a PISA, the contractor must first take action, as appropriate, to place or maintain the facility in a safe condition. A safe condition may include continued facility operation if, although a degraded or nonconforming SSC is not fully qualified, the impact on safe facility operations is judged to be acceptable, possibly aided by operational restrictions and the TSRs are still being met in terms of required operable equipment for the given mode of operations.

An operability determination is a forward-looking evaluation by the operating contractor of whether there is a reasonable expectation that continued operation of the facility is safe even when a degraded or nonconforming condition exists. An immediate operability determination should be made based on the best available information and operational restrictions imposed, if necessary, on confirmation of the condition. Subsequently, a final determination should be made and documented following a thorough engineering evaluation. The elements of the final operability determination should include the following:

- Description of the degraded or nonconforming condition of the SSC
- Description of the relationships on safe operations of the SSC functions
- Evaluation of the operability of the SSC given its condition
  - using analysis, tests, operating experience, and/or engineering judgment
  - considering availability of other equipment, conservatisms and margins, and cumulative effects of other outstanding degraded or nonconforming conditions

- Specification of additional operating restrictions if necessary (e.g., compensatory measures, additional engineering analysis by a certain date)
- Specification of restoration actions (may be added later)

Restoration actions for the degraded or nonconforming condition are to be developed by the contractor and scheduled at the first available opportunity commensurate with the safety significance and extent of restoration actions in an integrated manner with other facility commitments and resources. The final operability determination may be included as part of the ESS required to be submitted to DOE before removal of any operational restrictions.

Written USQDs are needed when a contractor identifies or is informed of a situation that indicates that the safety analyses that support the DOE-approved safety basis may not be bounding or may be otherwise inadequate.

In general, PISA arise from the following entry conditions:

- A discrepant as-found condition
- An operational event or incident
- New information, including discovery of an error, sometimes from an external source

The main consideration is that the analysis does not match the current physical configuration, or the analysis is inappropriate or contains errors. The analysis might not match the facility configuration because of a discrepant as-found condition. Analytical errors might involve using incorrect input values, invalid assumptions, improper models, or calculation errors. The USQ process starts when facility management has information that gives reason to believe that there is potential that the facility DSA might be inadequate.

Because a safety analysis inadequacy has potential to call into question information on which authorization of operations is based, per 10 CFR 830.203 the contractor is to

- take action, as appropriate, to place or maintain the facility in a safe condition until an ESS is completed;
- notify DOE of the situation;
- perform a USQD and notify DOE promptly of the results; and
- submit the ESS to DOE prior to removing any operational restrictions that were initiated.

DOE G 424.1-1B, Admin Chg 2, attachment C, "Guidance on Processing Potential Inadequacies in the Existing Safety Analyses," provides additional guidance on the timing of processing multiple PISAs found during audits, and the development of an ESS and justification for continued operation (JCO).

The USQ process is not applicable when new requirements are being implemented or different analysis methods that are used result in changed accident consequences or probabilities.

A JCO is a mechanism by which a contractor may request that DOE review and approve a temporary change to the facility safety basis that would allow the facility to continue operating in view of a specific and unexpected situation, considering the safety significance of the situation and any compensatory measures being applied during this period. A JCO is

associated only with situations where the PISA USQD is positive. It is appropriate to update the ESS in lieu of developing a JCO.

If the PISA arises from the situation where analytical errors in the DSA are identified or the analysis is otherwise inappropriate, a proposed DSA change should be prepared and submitted to DOE. However, if the DSA change cannot be submitted in a timely manner and a strong programmatic need exists to continue operations, a JCO that defines specific operational restrictions or other compensatory measures that will be maintained should be submitted to DOE for approval. A PISA could arise from a discrepant as-found condition. In this case, the facility should be restored to meet the design conditions. However, there may evolve situations where it may not be possible to align the facility configuration with the safety analysis in a timely manner and there may be a need to continue operations. In this situation, a JCO that defines specific operational restrictions or other compensatory measures that will be maintained should be submitted to DOE for approval.

The JCO should analyze the hazards and identify controls, appropriate for the hazards associated with the PISA and the length of time the conditions that resulted in the PISA are expected to exist. This analysis should be consistent with the approach in 10 CFR 830, appendix A to subpart B, for developing a DSA. Given that a JCO is intended to address emergent conditions in a timely manner, the associated analysis and controls/compensatory measures can be more simplified and conservative/bounding than in a final DSA. By taking a more bounding approach, control effectiveness can be assured even though the analysis may not be as complete as would be in a final DSA.

When DOE approves such a JCO, the JCO and any DOE-imposed conditions of approval become temporary additions to the safety basis that would permit operations to continue under the conditions specified, including a defined termination point. DOE review of the JCO should follow a similar approach to approval review of the DSA and should be documented in an SER (e.g., using a graded approach consistent with DOE-STD-1104). The approval authority for the DOE should be at the same level as the SBAA level for the facility.

A JCO is expected to define an appropriate set of temporary hazard controls to be in effect during the life of the JCO. In some cases, these hazard controls might involve temporary changes to the facility TSRs. If the JCO results in additional or modified controls, a review of the planned changes should be done prior to implementation to ensure the changes properly address the JCO. A JCO should have a predefined, limited life as necessary to perform the safety analysis of the unexpected situation, to identify and implement corrective actions, and to update the safety basis documents on a permanent basis. The JCO should define the termination point of the life of the JCO. In most cases, this would take the form of a functional point, such as the completion of turnover of a physical modification for routine operations that would occur after implementing the modification, post-modification testing, updating critical documentation, and training of the operations staff. The contractor should take actions to resolve the conditions that require the JCO or modify the safety basis during the next annual update to make the JCO no longer necessary. JCOs should not continue past a required annual DSA update unless the JCO was submitted within three months of the submittal date of the annual update. In some rare cases, it may not be practical to achieve this goal of being within three months of the submittal date of the annual update. It is

recommended that those changes be handled as soon as practicable. If this cannot be accomplished, the contractor should formally notify DOE of the reasons.

A JCO is not an appropriate means to request a change of the safety basis for a planned operation, a new experiment, a major modification, or new construction. In these cases, a request for a change to the facility safety basis should be prepared by the contractor and submitted to DOE for approval. Because the JCO is established in response to an unexpected condition, event, or new information, it is inappropriate to use it in planning new activities. A JCO should not be used in place of an exemption to 10 CFR 830 requirements. The JCO should be maintained as provided for in 10 CFR 830.202(c) for safety basis maintenance until the conditions that resulted in the PISA have been corrected and the JCO has been terminated.

A special case regarding PISAs exists when dealing with external audit team multiple concerns, each of which may (or may not) indicate the possibility of a potential inadequacy in the safety analyses. It may be impractical for facility staff to assess the situation quickly and disposition multiple concerns in the time frame normally expected for deciding whether a concern indicates a PISA and performing a USQD for each finding that qualifies as a PISA. In the face of multiple issues, it might be concluded that the short-term response to assuring a safe condition is to shut down operations. This might not be appropriate in all cases.

As an option in these cases, except where it is apparent that an imminent hazard exists, DOE should be consulted without delay, and a mutually agreed on approach to handling the concerns, including an expeditious schedule, should be developed. This approach should have a high priority for addressing the concerns, should prioritize the safety related concerns, and should disposition each as described for individual PISAs. Where it is apparent that an imminent hazard exists, the four steps for a potential inadequacy should be undertaken without delay.

A similar situation exists for design basis reconstitution projects where documentation on the original design bases may be lost or outdated. In this case, it can be expected that a team of engineers may identify many questions or issues that may not have current documentation and that may or may not constitute PISAs. For the purposes of the USQ process, design reconstitution projects can be regarded as DSA upgrades. For DSA upgrades, USQs should not result from the use of new analytical tools or in response to new requirements. A reconstitution project should have a process for prompt sorting and prioritizing of questions and issues between those that should be addressed as a normal part of the reconstitution project and those that should be handled promptly as PISAs. This process should be sufficiently timely to ensure that the expectations for PISAs can be met.

The USQ process is not applicable when new requirements are being implemented or different analysis methods that are used result in changed accident consequences or probabilities. 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. A reconstitution project should have a process for prompt sorting and prioritizing of the questions and issues between those that can be addressed as a normal part of the

reconstitution project and those that are to be handled promptly as PISAs. This process should be sufficiently timely to ensure that the expectations for PISAs can be met.

The USQ process does not apply to DSA upgrades in response to new requirements or to the use of new or different analytical tools during the upgrade process. However, the USQ process does apply when there is reason to believe that the current safety basis may not be bounding or may be otherwise inadequate.

i. Discuss the following terms as they apply to USQs:

- **Categorical exclusions**
- **Prior USQ determinations**
- **Inconsequential changes**
- **Margin of safety**
- **Design/evaluation basis accidents**
- **Important to safety**
- **Safety basis**
- **Restoration modification**
- **Evaluation of safety**
- **USQ**
- **Justification for continued operation**

Categorical Exclusions

The following is taken from DOE G 424.1-1B, Admin Chg 2.

A categorical exclusion is an exclusion from the requirements that USQDs be performed on proposed changes to a category of SSCs or procedures as a result of a determination that the category cannot credibly have the capability of creating a USQ if changed. Categorical exclusions are regarded as part of the contractor's USQ procedure and require DOE approval.

Prior USQ Determinations

The following is taken from DOE G 424.1-1B, Admin Chg 2.

Because application of the USQ process depends on facility-specific information, results of a USQD in one facility generally cannot be extrapolated to other facilities.

Inconsequential Changes

The following is taken from DOE G 424.1-1B, Admin Chg 2.

10 CFR 830.203 allows contractors to make physical and procedural changes and to conduct tests and experiments without prior DOE approval if the proposed change can be accommodated within the existing safety basis. The contractor must evaluate any proposed change to ensure that it will not affect the safety basis of the facility either explicitly or implicitly. The USQ process is applicable to the DSA and includes conditions of approval in SERs and facility-specific commitments made in compliance with DOE rules, Orders, or policies.

Margin of Safety

The following is taken from DOE G 424.1-1B, Admin Chg 2.

For purposes of performing the USQD, 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 would be expected to be related to the condition where some accident prevention or mitigation action is taken in response to the upset or accident, not the actual predicted failure point of some component.

Hazard control documents set forth the minimum acceptable limits for operation under normal and specified failure conditions; they ensure that the available safety equipment and operating conditions meet the assumptions in the existing safety analyses. They distill those aspects of the safety analyses that are required to ensure the performance of safety SSCs and personnel as relied on and defined in the safety analyses.

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 a 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 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 is evaluated to determine whether the change would reduce a margin of safety. (See video 67 for an explanation of margin of safety.)

10 CFR 830.205 requires that changes to the TSRs be submitted to DOE for review and approval. Changes to the TSRs could involve the need either to modify an existing TSR or to add a new TSR. If it is known that a proposed change only involves a TSR change, it is not required to go through the USQD to determine if DOE approval is required by 10 CFR 830.203. Similarly, if a change involves a TSR change, calling the change a positive USQD just because it requires DOE approval would be inappropriate.

Video 60. Margin of safety eroded at Davis-Besse
http://www.youtube.com/watch?v=oBHXWRuHsAY

Design/Evaluation Basis Accidents

The following is taken from DOE-STD-3009-94, Chg 3.

The major categories of DBAs are internally initiated operational accidents (e.g., fires, explosions, spills, criticality); natural events for the site (e.g., earthquakes, tornadoes) that could affect the facility; and man-made externally initiated events such as airplane crashes, transportation accidents, adjacent facility events, etc., that can either cause releases at the facility under examination or have a major impact on facility operations.

Where DBAs do not exist, or do not adequately cover the range of scenarios or bounding conditions, surrogate evaluation bases are needed. These derivative DBAs are used to estimate the response of SSCs to the range of accident scenarios and stresses that bound the envelope of accident conditions to which the facility could be subjected in order to evaluate accident consequences.

Video 61. More accurate headline would be "Fukushima Containment Worked"
http://atomicinsights.com/more-accurate-headline-would-be-fukushima-containment-worked/

Important to Safety

The following is taken from DOE G 424.1-1B, Admin Chg 2.

Equipment important to safety should be understood to include any equipment whose function, malfunction, or failure can affect safety either directly or indirectly. This includes SC SSCs and SS SSCs, and other systems that perform an important DID 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 safety systems, and should be included.

Safety Basis

The following is taken from DOE-STD-3009-94, Chg 3.

A 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.

Restoration Modification

The following is taken from DOE G 424.1-1B, Admin Chg 2.

Restoration actions for the degraded or nonconforming condition are to be developed by the contractor and scheduled at the first available opportunity commensurate with the safety significance and extent of restoration actions in an integrated manner with other facility commitments and resources. The final operability determination may be included as part of the ESS required to be submitted to DOE before removal of any operational restrictions.

Evaluation of Safety

The following is taken from 10 CFR 830.203.

If a contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility discovers or is made aware of a potential inadequacy of the DSA, it must

- take action, as appropriate, to place or maintain the facility in a safe condition until an ESS is complete;
- notify DOE of the situation;
- perform a USQD and notify DOE promptly of the results; and
- submit the ESS to DOE prior to removing any operational restrictions initiated to meet 10 CFR 830.203.

USQ

The following is taken from DOE G 424.1-1B, Admin Chg 2.

A USQ exists when

- 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
- the probability of an accident or malfunction of a different type than any evaluated previously in the DSA could be created
- a margin of safety could be reduced

Justification for Continued Operations (JCO)

The following is taken from DOE G 424.1-1B, Admin Chg 2.

A JCO is a mechanism by which a contractor may request that DOE review and approve a temporary change to the facility safety basis that would allow the facility to continue operating in view of a specific and unexpected situation, considering the safety significance of the situation and any compensatory measures being applied during this period. A JCO is associated only with situations where the PISA USQD is positive.

A JCO is expected to define an appropriate set of temporary hazard controls (compensatory measures) to be in effect during the life of the JCO. In some cases, these hazard controls might involve temporary changes to the facility TSRs.

A JCO should have a predefined, limited life as necessary to perform the safety analysis of the unexpected situation, to identify and implement corrective actions, and to update the safety basis documents on a permanent basis. The JCO should define the termination point of the life of the JCO. In most cases, this would take the form of a functional point, such as the completion of turnover of a physical modification for routine operations that would occur after implementing the modification, post-modification testing, updating critical documentation, and training of the operations staff.

j. Discuss the responsibilities of the contractor associated with USQ summaries and the USQ procedure.

The following is taken from 10 CFR 830.203.

The contractor responsible for a hazard category 1, 2, or 3 DOE existing nuclear facility must submit for DOE approval a procedure for its USQ process by April 1, 2001. Pending DOE approval of the USQ procedure, the contractor must continue to use its existing USQ procedure.

The contractor responsible for a hazard category 1, 2, or 3 DOE new nuclear facility must submit for DOE approval a procedure for its USQ process on a schedule that allows DOE approval in an SER issued pursuant to 10 CFR 830.207.

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

The contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility must annually submit to DOE a summary of the USQDs performed since the prior submission.

k. Describe DOE's responsibilities when not agreeing with a negative determination.

The following is taken from DOE G 424.1-1B, Admin Chg 2.

DOE can declare that a USQ exists as part of its oversight responsibility of the USQ process. Such a declaration might result from a disagreement with a contractor's negative USQD or might result from a condition for which the contractor has not done a USQD. When DOE declares a USQ, it is because DOE believes it should be involved in the review and approval of the situation to fulfill its responsibilities.

l. Discuss why the application of the graded approach does not apply to the USQ process.

The following is taken from DOE G 424.1-1B, Admin Chg 2.

In some situations, attempts to apply the graded approach to the USQ process have resulted in inadequate USQDs. For example, some contractors have applied the graded approach to the change control processes in a manner that does not require the application of formal change controls unless the change involves equipment, procedures, or operations important to safety. Caution needs to be applied in such a situation to ensure that this approach does not prevent changes from being considered by the USQ process that is a subset of change control. The only application of the graded approach to the USQ process is indirect. The graded approach may give a rough indication of how much justification or basis information

should be provided when explaining the answers to each of the seven USQD criteria. More elaborate and thorough basis information would be expected for changes to safety equipment than for non-safety equipment. In any case, the justification for the answers to the USQD criteria needs to be defensible.

Mandatory Performance Activities:

a. Demonstrate by participation on at least one assessment of the contractor USQ process or one regulatory review of either a new or revised contractor USQ procedure or review of one JCO.

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

22. Nuclear safety specialists must demonstrate a familiarity level knowledge of the relationships between the problems being addressed by safety analysis and computer codes, the design requirements for the codes, and the components of the codes.

a. Identify how functional requirements and applicability of safety analysis and design computer codes are defined, documented, and controlled relative to modeling and data assumptions, design constraints, sizing and timing conditions, and input/output parameters.

The following is taken from DOE O 414.1D, Admin Chg 1.

Safety software must be acquired, developed, and implemented using the ASME National Quality Assurance (NQA)-1-2012, *Quality Assurance Requirements for Nuclear Facilities Applications*, with the NQA-1a-2009 addenda, *Quality Assurance Requirements for Nuclear Facility Applications*, or other national or international consensus standards that provide an equivalent level of QA requirements as ASME/NQA-1-2012. DOE-approved QAPs applicable to safety software based on requirements from DOE O 414.1D, Admin Chg 1, *Quality Assurance*, are acceptable. The standards used must be specified by the user and approved by the designated DOE approval authority. Management of safety software must include the following elements:

- Involve the facility design authority, as applicable, in the identification of, requirements specification, acquisition, design, development, V&V (including inspection and testing), configuration management, maintenance, and retirement.
- Identify, document, control, and maintain safety software inventory. Inventory entries must include at a minimum the following: software description, software name, version identifier, safety software designation, grade level designation, specific nuclear facility application used, and the responsible individual.
- Establish and document grading levels for safety software using the graded approach. Grading levels must be submitted to, and approved by, the responsible DOE approval authority.
- Using the consensus standard selected and the grading levels established and approved, select and implement applicable safety software QA (SSQA) work activities from the following list:
  - Software project management and quality planning

- Software risk management
- Software configuration management
- Procurement and supplier management
- Software requirements identification and management
- Software design and implementation
- Software safety analysis and safety design methods
- Software V&V
- Problem reporting and corrective action
- Training of personnel in the design, development, use, and evaluation of safety software

b. Explain how a safety analysis problem being addressed by software is translated into functional requirements, how these requirements are established and controlled, and how the code is reconciled with the original safety analysis problem.

The following is taken from DOE G 200.1-1, chapter 4 (Archived).

Functional requirements define what the software product must do to support the system owner's business functions and objectives. The functional requirements should answer the following questions:

- How are inputs transformed into outputs?
- Who initiates and receives specific information?
- What information must be available for each function to be performed?

Identify requirements for all functions whether they are to be automated or manual. Describe the automated and manual inputs, processing, outputs, and conditions for all functions. Include a description of the standard data tables and data or records that will be shared with other applications. Identify the forms, reports, source documents, and inputs/outputs that the software product will process or produce to help define the functional requirements.

A functional model should be developed to depict each process that needs to be included. The goal of the functional model is to represent a complete top-down picture of the software product.

Flow diagrams should be used to provide a hierarchical and sequential view of the system owner's business functions and the flow of information through the processes.

c. Discuss the DOE toolbox codes (reference http://www.hss.energy.gov/CSA/CSP/sqa/central_registry.htm), their strengths, weaknesses, and other factors governing their appropriate use and the applicable DOE standards and guides for modeling their phenomena.

[Note: http://www.hss.energy.gov/CSA/CSP/sqa/central_registry.htm is no longer valid. The toolbox codes listed below can be found at http://orise.orau.gov/emi/scapa/software-quality-assurance/toolboxcodes.htm.]

The following is taken from DOE G 414.1-4.

Toolbox codes represent a small number of standard computer models or codes supporting DOE safety analysis. These codes have widespread use and are of appropriate qualification for use within DOE. The toolbox codes are acknowledged as part of DOE's Safety Software Central Registry. These codes are verified and validated and constitute a safe harbor methodology. That is to say, the analysts using these codes do not need to present additional defense as to their qualification provided that the analysts are sufficiently qualified to use the codes and the input parameters are valid. These codes may include commercial or proprietary design codes where DOE considers additional SQA controls are appropriate for repetitive use in safety applications and there is a benefit to maintain centralized control of the codes. The following seven widely applied safety analysis computer codes have been designated as toolbox codes:

1. ALOHA (chemical dispersion analysis)
2. CFAST (fire analysis)
3. EPIcode (chemical dispersion analysis)
4. GENII (radiological dispersion analysis)
5. IMBA (radiological dispersion analysis)
6. MACCS2 (radiological dispersion analysis)
7. MELCOR (LPF analysis)

The current designated toolbox codes and any software recognized in the future as meeting the toolbox equivalency criteria are no different from other custom-developed safety software as defined in DOE G 414.1-4. Consequently, software of this category should be developed or acquired, maintained, and controlled applying sound software practices as described in DOE O 414.1-4.

In the future, new versions of software may be added to the central registry while the older versions are removed. Over time, some of the software may be retired and recommended not to be used in DOE safety analysis. Still other software may be added through the formal toolbox-equivalent process, having been recognized as meeting the equivalency criteria. Thus, the central registry collection of safety software applications will be expected to evolve as software life-cycle phases, usage, and application requirements change. DOE G 414.1-4, appendix B, "Procedure for Adding or Revising Software to or Deleting Software from the DOE Safety Software Central Registry," addresses the process for adding new software applications and versions to, and removal of retired software from, the central registry.

Areal Locations of Hazardous Atmospheres (ALOHA)

ALOHA is an atmospheric dispersion model maintained by the Hazardous Materials Division of the NOAA. ALOHA is one of three separate, integrated software applications in the computer-aided management of emergency operations suite.

ALOHA is used primarily for the evaluations of the consequences of atmospheric releases of chemical species. In addition to safety analysis applications in DOE, ALOHA is applied for emergency response situations and for training. ALOHA allows the user to estimate the downwind dispersion of a chemical cloud based on the toxicological/physical characteristics of the released chemical, atmospheric conditions, and specific circumstances of the release. Graphic outputs include a "cloud footprint" that can be plotted on maps with mapping applications for response, planning, and local operational tasks to display the location of other facilities storing hazardous materials and vulnerable locations.

In May 2004, DOE performed an evaluation of ALOHA against DOE's SSQA criteria. A gap analysis report was generated that identifies the strengths and weaknesses based on ASME NQA-1 criteria. On completion of this evaluation, ALOHA version 5.2.3 was included in DOE's safety software central registry as a safety analysis toolbox code. Inclusion in DOE's safety software central registry provides to DOE sites knowledge of the pedigree of the SSQA level for ALOHA.

Consolidated Model of Fire and Smoke Transport (CFAST)

CFAST is a computer program that fire investigators, safety officials, engineers, architects, and builders can use to simulate the impact of past or potential fires and smoke in a specific building environment. CFAST is a two-zone fire model used to predict the environment in a multi-compartment structure subjected to a fire. It calculates the time-evolving distribution of smoke and fire gases and the temperature throughout a building during a user-specified fire. For more information on CFAST visit http://cfast.nist.gov/.

Emergency Prediction Information Code (EPIcode)

EPIcode program was developed to provide emergency response personnel and emergency planners with a software tool to help evaluate the atmospheric release of toxic substances. EPIcode allows fast estimation and assessment of chemical release scenarios associated with accidents from industry and transportation. The software can be used for safety analysis planning purposes on facilities handling toxic materials. EPIcode performs calculations for chemical source terms and resulting downwind concentrations. Source term calculations determine the rate at which chemical material is released to the atmosphere, release height, release durations, and the form and properties of the chemical on release. The release chemical may be a gas, a vapor, or an aerosol. For more information and view of EPIcode materials visit http://www.epicode.com.

In May 2004, DOE performed an evaluation of ALOHA against DOE's SSQA criteria. A gap analysis report was generated that identifies the strengths and weaknesses based on ASME NQA-1 criteria. On completion of this evaluation, EPIcode version 7.0 was included in DOE's safety software central registry as a safety analysis toolbox code. Inclusion in DOE's safety software central registry provides DOE sites knowledge of the pedigree of the SSQA level for EPIcode version 7.0.

GENII

GENII is a second generation of environmental dosimetry computer code compiled in the Hanford Environmental Dosimetry System (Generation II). GENII provides a state-of-the-art, technical peer-reviewed, documented set of programs for calculating radiation dose and risk from radionuclides released in the environment. The GENII system includes capabilities for calculating radiation doses following postulated chronic and acute releases.

GENII version 2.10 is currently being evaluated for inclusion in the Central Registry. For more information on GENII visit http://radiologicalsciences.pnl.gov/resources/hardware.asp.

Integrated Modules for Bioassay (IMBA)

IMBA ExpertTM(IX) software suite comprises a series of independent modules that implement ICRP Publication 66, *Human Respiratory Tract Model (HRTM)*, and the ICRP Publications 30 (series), *Limits for Intakes of Radionuclides by Workers*, through 39, *Principles for Limiting Exposure of the Public to Natural Sources of Radiation*; ICRP 67, ICRP 68, ICRP 69, and ICRP 71, biokinetic models.

In July 2006, DOE completed an evaluation of IMBA ExpertTM and IMBA Professional Plus. A gap analysis report was generated that identifies the strengths and weaknesses based on DOE O 414.1C (Archived) and DOE G 414.1-4 safety software requirements and criteria. The results of that evaluation are available at DOE-EH-0711, *Gap Analysis for IMBA and DOE Safety Software Central Registry Recommendation*.

The MELCOR Accident Consequence Code Systems (MACCS)

The MACCS code was originally developed for the NRC. MACCS2 evaluates doses and health risks from the accidental atmospheric releases of radionuclides. The principal phenomena considered in MACCS2 are atmospheric transport and deposition under time-variant meteorology, short-term and long-term mitigative actions and exposure pathways, deterministic and stochastic health effects, and economic costs.

In May 2004, DOE Office of Quality Assurance Programs performed an evaluation of MACCS2 against DOE's SSQA criteria. A gap analysis report was generated that identifies the strengths and weaknesses based on ASME NQA-1 criteria. On completion of this evaluation, MACCS2, version 1.13.1 was included in DOE's safety software central registry as a safety analysis toolbox code.

Methods for Estimation of Leakages and Consequences of Releases (MELCOR)

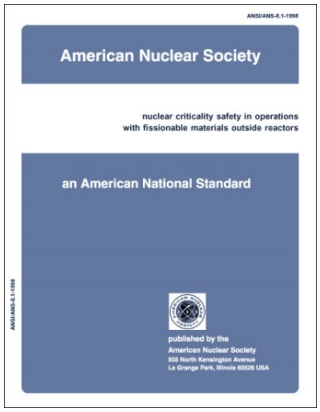
MELCOR is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of accidents in light water reactor nuclear power plants. Major uses of MELCOR for nonreactor facilities include estimation of confinement behavior due to radiological source terms under postulated accident conditions (and their sensitivities and uncertainties in a variety of applications), evaluation of LPFs, and survivability of fans, filters, and other engineering safety features. For more information on MELCOR visit: http://melcor.sandia.gov/.

23. Nuclear safety specialists must demonstrate a familiarity level knowledge of the following criticality safety-related American National Standards Institute/American Nuclear Society (ANSI/ANS) standards:

- **ANSI/ANS-8.1, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors***
- **ANS-8.3 (ANSI N-16.2), *Criticality Accident Alarm System***
- **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-8.15, *Nuclear Criticality Control of Special Actinide Elements***
- **ANS-8.19, *Administrative Practices for Nuclear Criticality Safety***

a. Describe the contents, requirements, and relationship among the above ANSI/ANS standards.

ANSI/ANS-8.1, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors



ANSI/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. Generalized basic criteria are presented, and limits are specified for some single fissionable units of simple shape containing ^{233}U, ^{235}U, or ^{239}Pu, but not for multiunit arrays. Requirements are stated for establishing the validity and areas of applicability of any calculational method used in assessing NCS. ANSI/ANS-8.1 does not include the details of ACs, the design of processes or equipment, the description of instrumentation for process controls, or detailed criteria to be met in transporting fissionable materials.

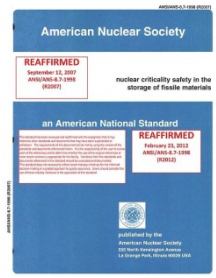
ANS-8.3 (ANSI N-16.2), Criticality Accident Alarm System

ANS-8.3 is applicable to all operations involving fissionable materials where inadvertent criticality can occur and cause personnel to receive unacceptable exposure to radiation. ANS-8.3 is not applicable to detection of criticality events where no excessive exposure to personnel is credible, nor to nuclear reactors or criticality experiments. ANS-8.3 does not include details of administrative actions or of emergency response actions that occur after alarm activation.

ANS-8.5 (ANSI N-16.4), Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material

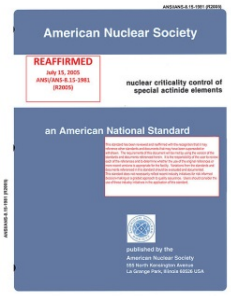
ANS-8.5 provides guidance for the use of borosilicate-glass Raschig rings as a neutron absorber for criticality control in ring-packed vessels, containing solutions of ^{235}U, ^{239}Pu, or ^{233}U. The chemical and physical environment, properties of the rings and packed vessels, maintenance inspection procedures, and operating guidelines are specified.

ANSI/ANS-8.7, Guide for Nuclear Criticality Safety in the Storage of Fissile Materials

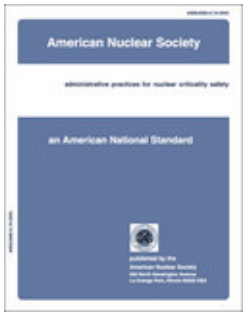


ANSI/ANS-8.7 is applicable to the storage of fissile materials. Mass and spacing limits are tabulated for uranium containing greater than 30 wt-% ^{235}U, for ^{233}U, and for plutonium, as metals and oxides. Criteria for the range of application of these limits are provided.

ANS-8.15, Nuclear Criticality Control of Special Actinide Elements



ANSI/ANS-8.15 provides guidance for the prevention of criticality accidents in the handling, storing, processing, and transporting of special actinide elements. Subcritical mass limits are provided for fourteen nuclides beginning with ^{237}Np and ending with ^{251}Cf. ANSI/ANS-8.15 constitutes an extension of ANS-8.1. The subcritical limits in ANSI/ANS-8.15 are in some cases substantially less than the estimated minimum critical values. This is to account for uncertainties in calculations. In view of the limited availability of most of the nuclides in the near term, there was no reason to push the limits to higher values.



ANS-8.19, Administrative Practices for Nuclear Criticality Safety

ANS-8.19 provides criteria for the administration of an NCS program for outside-of-reactor operations where 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.

b. Discuss the applicability of the above ANSI/ANS standards to the Department's facilities and processes.

The following is taken from DOE O 420.1C.

DOE O 420.1C, chapter III, "Nuclear Criticality Safety," is applicable to DOE elements and DOE contractors with responsibility for nuclear facilities and activities that involve or will potentially involve radionuclides in such quantities that are equal to, or greater than, the single parameter limits for fissionable materials listed in ANSI/ANS-8.1 and ANSI/ANS-8.15. These limits must be adjusted where process conditions could credibly involve moderators or reflectors that are more effective than light water.

Requirements:

- A criticality safety program (CSP) document must be developed and maintained that describes how the contractor will implement the requirements in DOE O 420.1C, chapter III, including the standards invoked by chapter III.
- The CSP document must describe how the contractor will satisfy the requirements of the ANSI/ANS-8 consensus NCS standards in effect as of the date of DOE O 420.1C (December 4, 2012), unless otherwise modified or approved by DOE. The CSP document must include an explanation as to why any recommendation in applicable ANSI/ANS-8 standards is not implemented.
- The CSP document must be submitted to, and approved by, DOE.
- CSEs must be conducted in accordance with DOE-STD-3007-2007 or by other documented methods approved by DOE.
- Fissile material accumulation control—Facilities that conduct operations using fissionable material in a form that could inadvertently accumulate in significant quantities must include procedures for detecting and characterizing accumulations. The following national standards provide relevant guidance for procedure development: American Society for Testing Materials (ASTM) C1455, *Standard Test Method for Nondestructive Assay of Special Nuclear Material Holdup Using Gamma-Ray Spectroscopic Methods*; and any other nondestructive assay consensus or DOE standards applicable to the measurement technique selected.
- CSEs must show that entire processes involving fissionable materials will remain subcritical under normal and credible abnormal conditions, including those initiated by design basis events.
- The criteria and process for developing the guidelines for firefighting in areas within or adjacent to moderator-controlled areas must be coordinated with firefighting pre-incident plans and procedures.

c. Discuss the role of the Department's nuclear safety specialists in implementing the requirements of these standards.

The following is taken from DOE-STD-1183-2007.

NSS personnel are not necessarily criticality control and safety SMEs. That is the function of the criticality safety engineer SME who has his or her own qualification program. NSSs interface with a variety of technical SMEs and managers to oversee the implementation of nuclear safety requirements and programs through oversight of contractors' implementation

of 10 CFR 830, subpart B and related activities (e.g., ISM, safety-related QA [subpart A], Price-Anderson Amendments Act [PAAA] evaluations, design reviews, accident/incident investigations, emergency response activities, rules, policies, standards, guides, etc., development, review, and implementation).

d. Define the following terms associated with nuclear criticality safety:

- **Criticality incident**
- **Double contingency principle**
- **Geometry control**
- **Nuclear criticality safety**
- **Significant quantity of fissionable material**
- **Temporary exemption**

Criticality Incident

The following is taken from Science Dictionary, *What is Criticality Incident?*

A criticality incident is an unintentional assembly of fissile material that establishes a nuclear chain reaction, resulting in sudden emission of dangerous amounts of neutrons, gamma rays, and heat.

Video 62. The demon core 1945
http://www.youtube.com/watch?v=hh89h8FxNhQ

Double Contingency Principle

The following is taken from 10 CFR 70.4.

Double contingency principle means that process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Geometry Control

The following is taken from IAEA Safety Standard DS407, version 6.

The subcriticality of a system depends on many parameters related to the fissile material, for example, mass, concentration, geometry, enrichment, or density. It is affected by the presence of other materials, for example, moderators, absorbers, and reflectors. Ensuring subcriticality may be realized through the control of an individual parameter or a combination of parameters (e.g., by limiting mass or by limiting mass and moderation). The means for controlling these parameters is ensured either by engineered and/or by administrative measures.

Nuclear Criticality Safety

The following is taken from Wikipedia, *Nuclear Criticality Safety*.

NCS is a field of nuclear engineering dedicated to the prevention of nuclear and radiation accidents resulting from an inadvertent, self-sustaining nuclear chain reaction. NCS is concerned with mitigating the consequences of a nuclear criticality accident.

Significant Quantity of Fissionable Material
The following is taken from DOE G 421.1-1 (Archived).

A significant quantity of fissionable material is the minimum quantity of fissionable material for which control is required to maintain subcriticality under all normal and credible abnormal conditions.

Temporary Exemption
The following is taken from DOE Order 5480.30, Chg 1.

PSOs may grant temporary exemptions with notification of appropriate headquarters-level offices. Temporary exemptions may be granted for durations up to one year.

e. *Discuss the contractor's responsibilities for the following in relation to criticality safety activities:
- **Criticality safety evaluations**
- **Monitoring**
- **Surveillance**
- **Transportation**
- **Storage**

[*Note: In the FAQS, competency 23 has two KSAs identified as d.]

Criticality Safety Evaluations
The following is taken from DOE O 420.1C.

The contractor is responsible for developing and maintaining a nuclear CSP document that describes how the contractor will implement the requirements in DOE O 420.1C, chapter III, "Nuclear Criticality Safety."

CSEs must be conducted in accordance with DOE-STD-3007-2007 or by other documented methods approved by DOE. CSEs must show that entire processes involving fissionable materials will remain subcritical under normal and credible abnormal conditions, including those initiated by design basis events.

Monitoring
The following is taken from DOE O 420.1C.

Fissile material accumulation control—Facilities that conduct operations using fissionable material in a form that could inadvertently accumulate in significant quantities must include procedures for detecting and characterizing accumulations. The following national standards provide relevant guidance for procedure development: ASTM C1455 and any other nondestructive assay consensus or DOE standards application to the measurement technique selected.

Surveillance

The following is taken from DOE O 226.1B.

The contractor must establish an assurance system that includes assignment of management responsibilities and accountabilities and provides evidence to assure DOE and the contractor's management that work is being performed safely, securely, and in compliance with all requirements; risks are being identified and managed; and that the systems of control are effective and efficient.

Transportation

The following is taken from DOE-STD-1135-99.

Nuclear CSEs are performed to technically demonstrate the subcriticality of fissionable material processes, operations, and situations for transportation and storage under all normal and credible abnormal conditions.

Storage

The following is taken from DOE M 441.1-1.

Contractors must ensure the following:

- Nuclear materials stored outside engineered contamination barriers are packaged in accordance with the requirements described in DOE M 441.1-1, *Nuclear Material Packaging Manual*, Attachment 1, "Contractor Requirements Document."
- Chemical, radiological, and physical characteristics of the stored materials are evaluated for the lifetime of the storage of the material and are appropriate for the material package.
- Concerns with corrosion, radiolytic and thermal degradation, oxidative expansion, pressurization, incompatible materials, and usage (handling) that may result in container failure are addressed as part of the package design.
- A surveillance program is established and implemented to ensure the nuclear material storage package continues to meet its design criteria.
- Documentation of the nuclear material storage package design and surveillance is maintained.

24. Nuclear safety specialists must demonstrate a familiarity level knowledge of the following DOE Orders, technical standards, and guides:

- **DOE-STD-3011-2002, *Guidance for Preparation of Basis for Interim Operation (BIO) Documents***
- **DOE-STD-3014-96, *Accident Analysis for Aircraft Crash into Hazardous Facilities***
- **DOE-STD-1120-2005, *Integration of Environment, Safety, and Health into Facility Dispositions Activities***
- **US NRC Guide 1.70, *Standard Format and Content of Safety Analysis Reports of Nuclear Power Plants***
- **29 CFR 1910.120, "Hazardous Waste Operation and Emergency Response"**
- **DOE-STD-1163-2003, *Integration of Multiple Hazard Analysis Requirements and Activities***
- **DOE-NA-STD-3016-2006, *Hazard Analysis Reports for Nuclear Explosive Operations***
- **DOE O 460.1B, *Packaging and Transportation Safety***
- **DOE G 460.1-1, *Implementation Guide for Use with DOE O 460.1A, Packaging and Transportation Safety***
- **DOE O 461.1A, *Packaging and Transfer or Transportation of Materials of National Security Interest***
- **DOE M 461.1-1, Chg 1, *Packaging and Transfer of Materials of National Security Interest Manual***
- **Secretary of Energy Notice (SEN) SEN-35-91, *Nuclear Safety Policy***

[Note: DOE-STD-3014-96 has been superseded by DOE-STD-3014-2006. Presumably DOE-HDBK-1163-2003 is meant instead of DOE-STD-1163-2003. USRC Guide 1.70 has been replaced by US NRC Guide 1.70, revision 3. DOE O 460.1B has been superseded by DOE O 460.1C. DOE O 461.1A has been superseded by DOE O 461.1B, *Packaging and Transportation for Offsite Shipment of Materials of National Security Interest*. The title of 29 CFR 1910.120 is "Hazardous Waste Operation and Emergency Response." DOE M 461.1-1, Chg 1 has been archived and not replaced. SEN-35-91 has been replaced by DOE P 420.1, *Department of Energy Nuclear Safety Policy*.]

a. Describe the contents and requirements of and relationship among the above technical standards.

The following is taken from DOE P 420.1.

DOE's policy is to conduct its operations such that 1) individual members of the public be provided a level of protection from the consequences of DOE operations such that individuals bear no significant additional risk to life and health to which members of the general population are normally exposed; and 2) DOE workers' health and safety are protected to levels consistent with, or better than, that achieved for workers in similar industries. It is the policy of DOE to design, construct, operate, and decommission its nuclear facilities in a manner that ensures adequate protection of workers, the public, and the environment.

DOE-STD-3011-2002, Guidance for Preparation of Basis for Interim Operation (BIO) Documents

DOE-STD-3011-2002 provides guidance for the development of BIO documents that are an acceptable form of DSA under the provision of the 10 CFR 830 rule. In this regard, it supplements the guidance in DOE G 421.1-2A, *Implementation Guide for Developing Documented Safety Analyses to Meet Subpart B of 10 CFR Part 830*.

DOE-STD-3014-2006, Accident Analysis for Aircraft Crash into Hazardous Facilities

DOE-STD-3014-2006 provides the user with sufficient information to evaluate and assess the significance of aircraft crash risk on facility safety without expending excessive effort where it is not required. It provides an approach for performing a conservative analysis of the risk posed by a release of hazardous radioactive or chemical material resulting from an aircraft crash into a facility containing significant quantities of such material. This approach can establish whether a facility has a significant potential for an aircraft impact, and, given an aircraft impact, whether a facility has the potential for an accident producing significant offsite or onsite consequences. The analysis is based on the structural properties of a facility and the inventory at a facility.

DOE-STD-1120-2005/Vol. 1, Integration of Environment, Safety, and Health into Facility Disposition Activities

DOE-STD-1120-2005/vol. 1 provides a DOE-approved methodology for preparing a DSA for decommissioning of nuclear facilities, as well as environmental restoration activities that involve work not done within a permanent structure.

U.S. NRC Guide 1.70, Revision 3, Standard Format and Content of Safety Analysis Reports of Nuclear Power Plants

U.S. NRC Guide 1.70 provides a standard format for SARs considered acceptable to the NRC for nuclear reactor facilities. It is identified as a safe harbor method in 10 CFR 830 for the safety basis documentation requirements for a DOE reactor.

29 CFR 1910.120, Hazardous Waste Operations and Emergency Response

29 CFR 1910.120 covers the following operations: clean-up operations required by a governmental body involving hazardous substances that are conducted at uncontrolled hazardous waste sites; corrective actions involving clean-up operations at sites covered by the Resource Conservation and Recovery Act (RCRA) of 1976; voluntary clean-up operations at sites recognized by Federal, state, local, or other governmental bodies as uncontrolled hazardous waste sites; operations involving hazardous wastes that are conducted at treatment, storage, and disposal facilities regulated by 40 CFR 264, “Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities,” and 40 CFR 265, “Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities,” pursuant to RCRA, or by agencies under agreement with EPA to implement RCRA regulations; and emergency response operations for release of, or substantial threats of releases of hazardous substances without regard to the location of the hazard.

DOE-NA-STD-3016-2006, Hazard Analysis Reports for Nuclear Explosive Operations

DOE-NA-STD-3016-2006 clarifies DOE/NNSA expectations and provides guidance for preparing HARs for nuclear explosive operations (NEOs). This standard addresses operations-specific HARs and their interaction with facility safety basis documents. The general requirements for operation-specific HARs are those contained in DOE-STD-3009-94, Chg 3; however, they are more fully developed in DOE-NA-STD-3016-2006 to account for issues unique to NEOs and interface issues with the facility or special operations.

DOE-HDBK-1163-2003, Integration of Multiple Hazard Analysis Requirements and Activities

DOE-HDBK-1163-2003 is intended to provide DOE and contractor safety personnel with a resource to support the planning, technical review, or conduct of HA activities. Clarifications of requirements and discussions of best practices can be used to help improve cost effectiveness, clarify organizational roles and responsibilities, and provide a basis for enhancing the technical quality of HA activities. The concepts presented in DOE-HDBK-1163-2003 are supportive of ISMS and can be applied to nuclear or hazardous non-nuclear facilities that are operating, shut down, or actively conducting facility disposition activities.

DOE O 460.1C, Packaging and Transportation Safety

DOE O 460.1C establishes safety requirements for the proper packaging and transportation of DOE and NNSA offsite shipments and onsite transfers of radioactive and other hazardous materials and for modal transportation. Offsite is any area within or outside a DOE site where the public has free and uncontrolled access; onsite is any area within the boundaries of a DOE site or facility where access is controlled.

DOE G 460.1-1, Implementation Guide for Use with DOE O 460.1A

DOE G 460.1-1 supplements DOE O 460.1C by providing clarifying material for the implementation of safety procedures for packaging and transportation of hazardous materials.

DOE O 461.1B, Packaging and Transportation for Offsite Shipment of Materials of National Security Interest

DOE O 461.1B makes clear that the packaging and transportation of all offsite shipments of materials of national security interest for DOE must be conducted in accordance with DOT and NRC regulations that would be applicable to comparable commercial shipments, except where an alternative course of action is identified in DOE O 461.1B. The requirements and responsibilities prescribed in DOE O 461.1B are intended to ensure NNSA resources including the transportation safeguards system, are used and managed in an efficient manner.

DOE P 420.1, Department of Energy Nuclear Safety Policy

DOE P 420.1 documents DOE's nuclear safety policy which is to design, construct, operate, and decommission its nuclear facilities in a manner that ensures adequate protection of workers, the public, and the environment.

b. Describe the role of nuclear safety specialists with respect to the requirements in these Orders and standards.

The following is taken from DOE-STD-1183-2007.

DOE-STD-1183-2007 states that the duties and responsibilities of the NSS is to oversee implementation of nuclear safety requirements and programs including participating in the oversight of contractor implementation of 10 CFR 830; evaluating implementation of ISM as related to safety bases; evaluating the nuclear safety posture of nuclear facilities and operations for PAAA compliance, contract performance, operational readiness reviews, readiness assessments, and other periodic assessments; participating in nuclear facility design reviews, and safety system status monitoring; and evaluating design and analysis uncertainties with the functionalities of systems as described in the DSA.

c. Describe the application of DOE-STD-3011, *Guidance for Preparation of Basis for Interim Operations (BIO) Documents*, to nuclear facilities safety basis.

The following is taken from DOE-STD-3011-2002.

10 CFR 830, appendix A to subpart B, states that the contractor responsible for 1) a DOE nuclear facility with a limited operational life; 2) the deactivation of a DOE nuclear facility; or 3) the transition surveillance and maintenance of a DOE nuclear facility may prepare its DSA using the methodology in DOE-STD-3011-2002. Specifically the DSA (BIO) must include the following:

- Facility categorization according to DOE-STD-1027-92
- A description of the facility (including the work to be performed)
- A systematic identification of hazards associated with the facility
- Evaluation of normal, abnormal, and accident conditions (including potential NPH that might be associated with long-term status) that might be associated with the generation or release of radioactive or other hazardous materials, including consideration of the need for analysis of beyond DBAs
- Derivation and classification of hazard controls necessary to protect workers, the public, and the environment
- Definition of the characteristics of safety management programs necessary to ensure safe operation, including criticality safety, when criticality hazards exist

In the context of DOE-STD-3011-2002, these activities are interpreted to be interim operations, since the expected normal lifetime for these activities is ideally anticipated to be the short interim transitional periods immediately prior to, during, or after deactivation. While DOE-STD-3011-2002 allows for an abbreviated and graded approach to development of a safety basis, the expectation exists that the completeness of the analysis will be sufficient so that even though a limited operational life is envisioned, significant hazards will be identified and appropriate controls implemented accordingly. It is important to recognized that the ideal may not be realized. That is, especially in the case of transition surveillance and maintenance, the time interval that a facility may be in that mode may extend many years beyond “short.” When this may be the case, special attention must be paid to hazards that may develop over the extended period of time.

The contractor must develop TSRs as part of the safety basis to satisfy the requirements of 10 CFR 830.205.

DOE G 421.1-2 defines a limited life facility as a facility with an approved deactivation plan calling for cessation of operation within a stated period (5 years or less). The primary rationale for utilizing the BIO approach is that the short remaining operational life of the facility does not justify the increased time and cost required to develop a DSA fully utilizing the DOE-STD-3009-94, Chg 3, methodology.

Maximum advantage should be taken of pertinent existing safety analyses and design information (i.e., requirements and their bases) that are immediately available, or can be retrieved through reasonable efforts. Other information arises from existing sources such as process HAs, fire hazards analyses, explosive safety analyses, HASPs, environmental impact statements, etc. When existing information is not current and correct, cannot be verified, or does not exist, the existing information must be supplemented. Analyses should generally be qualitative, but thorough. When adequate information is not available to fully support the DSA, conservative compensatory approaches to assuring adequate safety should be considered.

Deactivation refers to the process of placing the facility in a stable and known condition and the removal of readily removable hazardous and radioactive materials. Deactivation activities include the removal of energy sources, draining and/or de-energizing nonessential systems, removal of stored radioactive and hazardous materials, and related actions. Deactivation should be a short-term process, measured in months, or at most a few years. The mission is to remove hazardous material to decrease risk during extended surveillance and maintenance, or deactivation and decommissioning.

Transition surveillance and maintenance means activities conducted when a facility is not operating and not during deactivation, decontamination, and decommissioning activities. Ideally, deactivation would precede transition surveillance and maintenance, but often it does not. That is, mission-related operations may have been terminated and the facility placed into a surveillance and maintenance mode, possibly with the expectation of resuming operations at a later date, without removal of hazardous materials. During this phase surveillance and maintenance are the primary activities being conducted at the facility. These activities are necessary for satisfactory containment of hazardous materials and protection of workers, the public, and the environment. Surveillance and maintenance activities include providing periodic inspections and maintenance of structures, systems, and equipment necessary for the satisfactory containment of contamination and for protection of workers, the public, and the environment. Maintenance of the facility in a stable and known condition includes actions to prevent alteration in chemical makeup, physical state, and/or configuration of a hazardous substance or radioactive material. It includes actions taken with regard to physical SSCs.

d. Determine whether aircraft crashes pose an acceptable or unacceptable hazard to safety of nuclear facilities.

The following is taken from DOE-STD-3014-2006.

DOE-STD-3014-2006 provides methodologies that take into consideration items determined to be important to understanding the risk from aircraft crash into hazardous facilities. These items include the number of aircraft operations/flights; crash probabilities; aircraft characteristics; crash kinematics; impacting missiles; local, global, and vibratory structural damage; structure characteristics; source terms; release energy; and meteorological conditions. When applied as a complete approach, the methodologies in DOE-STD-3014-2006 will result in a technically justified, conservative analysis of the risk posed by releases resulting from aircraft crash. The risk will be defined at a sufficient level of detail to document the safety of the facility with respect to aircraft crash, and at the same level of detail as would be expected for other types of accident analyses.

DOE-STD-3014-2006 allows the analysis to proceed along a series of increasingly complex steps; the results at each step are used to determine whether it is necessary to proceed to the next step or whether sufficient information has been provided and the analysis can be stopped and documented. As one proceeds through the steps, the results will get closer to an actual estimate of the risk, but even after fully implementing DOE-STD-3014-2006, the results will still be more conservative than would be expected from a best-estimate risk assessment. In summary, following DOE-STD-3014-2006 will, in the vast majority of cases, provide sufficient information to document facility safety and support sound decision-making for addressing the effects of an aircraft crash in the context of facility safety.

This is not a criterion-based standard. It does not provide any hard and fast rules prescribing what actions should be taken in response to the results; it does not even prescribe whether any action should be considered. It does provide quantitative guidelines against which the results for each step in the analysis can be measured; however, these are only for the purpose of determining whether further analysis should be performed. Meeting or not meeting these guidelines should not be interpreted as indicating that preventive or mitigative actions either are or are not required.

Video 63. Nuclear power plant aircraft impact simulations and safety structures
http://www.youtube.com/watch?v=X5_NWh8wbR8

e. Discuss the phenomena of aircraft crashes as a mechanism for releasing toxic materials.

The following is taken from DOE-STD-3014-2006.

First, the results of the structural analysis are used to describe the level of damage. This description will provide a conservative estimate of the structural damage that has occurred including the path and location of penetrators; the damage state of walls, barriers, and equipment; the location of the aircraft fuel; and other pertinent information.

Assume that all available fuel burns, as well as any other combustibles that are in the path of the penetrators. Assume that any high explosive material undergoes a high explosive violent reaction. High explosive material includes materials such as dynamite ion exchange resins, but not highly flammable materials that are subject to burning rather than true explosion (e.g., aircraft fuel, hydrogen gas). This assumption only pertains to combustibles and explosives that are directly affected by the penetrators; that is, they are in areas or compartments that are actually breached by the penetrators.

Evaluate the extent to which secondary effects cause the scenario to spread beyond the area directly damaged by the crash. Consider if there is sufficient combustible material to breach additional barriers and spread further through the facility. Credit can be taken for the existence of fire barriers and breaks if they haven't been damaged by the crash. A characterization of fire duration will almost certainly be required. Credit should not be allowed for functioning fire suppression systems unless an analysis can demonstrate they will remain effective following the crash. An evaluation should include a determination of whether the force of any explosion is capable of causing further barriers to be damaged or destroyed or has the potential to cause additional fires and/or explosions in the facility. Again, credit can be taken for the dissipation of explosive energy by existing barriers, if they have not been damaged by the crash. Credit can be taken for diversion of the explosive force through breaches caused by the crash; thus, reducing the shock to intact barriers.

Based on the prior steps, determine if any of the material at risk (MAR) in the facility can be impacted by a release mechanism (e.g., shock, fire, or explosion) as a result of the scenario. This is likely if there is any material that is not separated from the energy available from the release mechanism by an intact barrier capable of dissipating that energy. Consideration should be given to the potential for the primary confinement around any of the material to be breached that can occur if the structural integrity of the primary confinement is degraded below that required to migrate through the breach. Finally, the potential for a direct release to the atmosphere should be determined. This is likely to be the case if there are no intact barriers between the material and the atmosphere, assuming that the primary confinement has failed and there is a driving force capable of causing the material to migrate through the release path.

f. Discuss the application of DOE-STD-1120 and 29 CFR 1910.120 to decommissioning and certain environmental restoration activities, and discuss the content of a safety basis health and safety plan and how it can be used in a dynamic project, including management of hazard controls.

The following is taken from DOE-STD-1120-2005/vol. 1.

DOE-STD-1120-2005 is identified as a safe harbor approach for preparing a DSA for decommissioning and environmental restoration activities. Volume one of DOE-STD-1120-2005 provides DOE an approved methodology for preparing a DSA for decommissioning of nuclear facilities, as well as environmental restoration activities that involve work not done within a permanent structure. Contractors may prepare a DSA by using the method described in DOE-STD-1120-2005 and the provisions of 29 CFR 1910.120 or 29 CFR 1926.65. HAZWOPER requirements specifically focus on provisions for developing a safety and

health (S&H) program and site-specific HASP. HAZWOPER applies to all worker hazards, including physical hazards posed by deconstruction or environmental restoration work (e.g., use of heavy equipment, excavations, confined space entry, and hot work). In the context of the requirements of 10 CFR 830, subpart B, the scope of HAZWOPER is taken to include those hazards, associated with controls, and S&H programs that must be identified and maintained within a hazard category 1, 2, or 3 facility's safety basis.

Volume one of DOE-STD-1120-2005 applies to hazard category 2 or 3 environmental restoration activities and decommissioning projects as defined in 10 CFR 830, subpart B. 29 CFR 1910.120 applies to general worker safety by providing requirements to ensure worker S&H during emergency response for hazardous waste operations, including projects conducted under the Comprehensive Environmental Response, Compensation, and Liability Act. It provides requirements to ensure worker S&H during emergency release of hazardous substances wherever they occur. HAZWOPER requires an S&H program and site-specific S&H plan for cleanup operations involving hazardous substances; operations involving hazardous wastes conducted at treatment, storage, and disposal facilities; and emergency response operations for releases of, or substantial threats of release of, hazardous substances. Derivation of controls is necessary for facility decommissioning projects that involve more than low-level residual fixed radioactivity.

Decommissioning projects that have only low-level residual fixed radioactivity are not expected to have the potential for accidents involving significant radiological consequences. This is reflected in 10 CFR 830.205 that states that TSRs are not required for this type of activity. The DSA format for this type of decommissioning activity may exclude topics related to accident analysis, safety SSCs, and TSR derivation.

g. Discuss the phenomena to which packaging is designed to withstand transportation accidents and the relationship to accident severity.

The following is taken from 10 CFR 71.71.

Evaluation of each package design under normal conditions of transport must include a determination of the effect on that design of the conditions and tests specified in 10 CFR 71.71, "Normal Conditions of Transport." Separate specimens may be used for the free drop test, the compression test, and the penetration test, if each specimen is subjected to the water spray test before being subjected to any of the other tests.

With respect to the initial conditions for the tests in 10 CFR 71.71, the demonstration of compliance with the requirements of 10 CFR 71.71 must be based on the ambient temperature preceding and following the tests remaining constant at that value between -29°C (-20°F) and +38°C (+100°F) which is most unfavorable for the feature under consideration. The initial operating pressure, unless a lower internal pressure consistent with the ambient temperature considered to precede and follow the tests is more unfavorable.

Heat—An ambient temperature of 38°C (100°F) in still air, and insolation according to table 9.

Table 9. Insolation data

Form and location of surface	Total insolation for a 12-hour period (g cal/cm^2)
Flat surfaces transported horizontally:	
Base	None
Other surfaces	800
Flat surfaces not transported horizontally	200
Curved surfaces	400

Source: 10 CFR 71.71

Cold—An ambient temperature of -40°C (-40°F) in still air and shade.

Reduced external pressure—An external pressure of 25 kilopascals (kPa) (3.5 pound force per square inch [lbf/in^2]) absolute.

Increased external pressure—An external pressure of 140 kPa (20 lbf/in^2) absolute.

Vibration—Vibration normally incident to transport.

Water spray—A water spray that simulates exposure to rainfall of approximately 5 cubic centimeter per hour (cm/h) (2 inches per hour [in/h]).

Free drop—Between 1.5 and 2.5 hours after the conclusion of the water spray test, a free drop through the distance specified below onto a flat, essentially unyielding, horizontal surface, striking the surface in a position where maximum damage is expected.

Table 10. Criteria for free drop test (weight/distance)

Package weight		Free drop distance	
Kilograms	(Pounds)	Meters	(Feet)
Less than 5,000	(Less than 11,000)	1.2	(4)
5,000 to 10,000	(11,000 to 22,000)	0.9	(3)
10,000 to 15,000	(22,000 to 33,100)	0.6	(2)
More than 15,000	(More than 33,100)	0.3	(1)

Source: 10 CFR 71.71

Corner drop—A free drop onto each corner of the package in succession, or in the case of a cylindrical package onto each quarter of each rim, from a height of 0.3 m (1 foot [ft.]) onto a flat, essentially unyielding, horizontal surface. This test applies only to fiberboard, wood, or fissile material rectangular packages not exceeding 50 kg (110 lbs) and fiberboard, wood, or fissile material cylindrical packages not exceeding 100 kg (220 lbs).

Compression—For packages weighing up to 5,000 kg (11,000 lbs), the package must be subjected, for a period of 24 hours, to a compressive load applied uniformly to the top and bottom of the package in the position where the package would normally be transported. The compressive load must be the greater of the following:

- The equivalent of five times the weight of the package
- The equivalent of 13 kPa (2 lbf/in^2) multiplied by the vertically projected area of the package

Penetration—Impact of the hemispherical end of a vertical steel cylinder of 3.2 cm (1.25 in) diameter and 6 kg (13 lbs) mass, dropped from a height of 1 m (40 in) onto the exposed surface of the package that is expected to be most vulnerable to puncture. The long axis of the cylinder must be perpendicular to the package surface.

Video 64. Free drop test #1
http://www.youtube.com/watch?v=wW9kAGdFbBc

Video 65. Free drop test #2
http://www.youtube.com/watch?v=GvLVFoIVmMI

Video 66. Fire exposure simulation test
http://www.youtube.com/watch?v=BnSp_2CA3XI

25. Nuclear safety specialists must demonstrate a familiarity level knowledge of the Price-Anderson Amendment Act (PAAA) of 1988 and its relationship to subparts A and B of 10 CFR 830.

a. Describe the purpose of the Price-Anderson Amendment Act.

The following is taken from Wikipedia, *Price-Anderson Nuclear Industries Indemnity Act*.

The main purpose of the PAAA of 1988 is to partially indemnify the nuclear industry against liability claims arising from nuclear incidents while still ensuring compensation coverage for the general public. The Act establishes a no-fault insurance-type system where the first approximately $12.6 billion is industry funded as described in the Act; and any claims above $12.6 billion would be covered by a congressional mandate to retroactively increase nuclear utility liability or would be covered by the Federal government.

b. Discuss the general applicability to the Department's nuclear safety activities.

The following is taken from DOE-STD-1085-95 (Archived).

The PAAA provides indemnification to DOE contractors who manage and conduct nuclear activities in the DOE complex. In a general sense, the government acts as an insurer for these contractors against any findings of liability arising from the nuclear activities of the contractor within the scope of its contract.

In 1988, the PAAA of 1988 was signed into law to extend the government insurance program that was about to expire. It differed from the original act in two principal ways. First, it made

Price-Anderson coverage mandatory for all management and operating (M&O) contractors, subcontractors, and suppliers conducting nuclear activities for DOE. Second, Congress mandated that DOE change its methods of managing nuclear activities at those sites by requiring DOE to undertake enforcement actions against indemnified contractors for violations of nuclear safety requirements. Thus, indemnification risks would be minimized by minimizing the risk to workers and the public. The benefit of indemnification is accompanied by the availability of sanctions to assure compliance with nuclear safety rules.

c. Describe the general indemnity that DOE offers to contractors.

The following is taken from DOE *Safety and Security Enforcement Process Overview*.

The Atomic Energy Act (AEA) provides indemnification to DOE contractors who manage and operate nuclear facilities in the DOE complex; associated subcontractors and suppliers are included under this coverage. In 1988, the PAAA was signed into law to continue this indemnification. The rules that implement the PAAA subject DOE-indemnified contractors, subcontractors, and suppliers to potential civil penalties for violations of DOE rules, regulations, and compliance orders relating to nuclear safety requirements. As part of its agreement to continue the indemnification coverage, Congress required that DOE-indemnified contractors, subcontractors, and suppliers be made subject to civil penalties for violations of DOE's nuclear safety requirements. On August 17, 1993, DOE published its nuclear safety enforcement procedural rules and enforcement policy, which has since been amended several times. The Director of HS-40 has the responsibility to carry out the statutory enforcement authority provided to DOE in the PAAA.

The Bob Stump National Defense Authorization Act for Fiscal Year 2003 extended previously-approved indemnification levels until December 31, 2004, and required DOE to promulgate a final rule to establish and provide for enforcement of worker S&H requirements. The Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 extended indemnification until December 31, 2006. The Energy Policy Act of 2005 extended indemnification of DOE contractors until December 31, 2025, increased liability coverage to approximately $10 billion (plus inflation adjustments) per incident, and repealed waivers or exclusions for remission of civil penalties for nonprofit organizations upon the signing of a new contract.

The National Defense Authorization Act for Fiscal Year 2000 added new Section 234B to the AEA; on January 26, 2005, DOE published 10 CFR 824 to implement this new section. Section 234B provides that a DOE contractor or subcontractor who violates any rule, regulation, or order relating to the safeguarding or security of Restricted Data and/or other classified or sensitive information shall be subject to a civil penalty. Title 10 CFR 824 provides that civil penalties will be assessed for violations of requirements for the protection and control of classified information

On February 9, 2006, DOE issued 10 CFR 851, "Worker Safety and Health Program," subpart E, "Enforcement Process," to be applied to violations of the worker S&H regulation, and, appendix B, "General Statement of Enforcement Policy," for such violations. 10 CFR 851 provides that, beginning May 25, 2007, no work may be performed at a covered workplace unless an approved worker S&H program is in place.

d. Discuss the requirements associated with the topics below:
- **Quality Assurance (QA)**
- **Safety basis**

Quality Assurance (QA)

The following is taken from 10 CFR 830, subpart A.

Contractors conducting activities, including providing items or services, that affect, or may affect, the nuclear safety of DOE nuclear facilities must conduct work in accordance with the QA criteria in 10 CFR 830.122, "Quality Assurance Criteria," which states that the QAP must address the following management, performance, and assessment criteria:

- Criterion 1—Management/Program
  - Establish an organizational structure, functional responsibilities, levels of authority, and interfaces for those managing, performing, and assessing the work
  - Establish management processes, including planning, scheduling, and providing resources for the work.
- Criterion 2—Management/Personnel Training and Qualification
  - Train and qualify personnel to be capable of performing their assigned work
  - Provide continuing training to personnel to maintain their job proficiency.
- Criterion 3—Management/Quality Improvement
  - Establish and implement processes to detect and prevent quality problems
  - Identify, control, and correct items, services, and processes that do not meet established requirements
  - Identify the causes of problems and work to prevent recurrence as a part of correcting the problem
  - Review item characteristics, process implementation, and other quality-related information to identify items, services, and processes needing improvement.
- Criterion 4—Management/Documents and Records
  - Prepare, review, approve, issue, use, and revise documents to prescribe processes, specify requirements, or establish design
  - Specify, prepare, review, approve, and maintain records.
- Criterion 5—Performance/Work Processes
  - Perform work consistent with technical standards, AC, and other hazard controls adopted to meet regulatory or contract requirements, using approved instructions, procedures, or other appropriate means
  - Identify and control items to ensure their proper use
  - Maintain items to prevent their damage, loss, or deterioration
  - Calibrate and maintain equipment used for process monitoring or data collection.
- Criterion 6—Performance/Design
  - Design items and processes using sound engineering/scientific principles and appropriate standards
  - Incorporate applicable requirements and design bases in design work and design changes
  - Identify and control design interfaces
  - Verify or validate work before approval and implementation of the design.

- Criterion 7—Performance/Procurement
  - Procure items and services that meet established requirements and perform as specified
  - Evaluate and select prospective suppliers on the basis of specified criteria
  - Establish and implement processes to ensure that approved suppliers continue to provide acceptable items and services.
- Criterion 8—Performance/Inspection and Acceptance Testing
  - Inspect and test specified items, services, and processes using established acceptance and performance criteria
  - Calibrate and maintain equipment used for inspections and tests.
- Criterion 9—Assessment/Management Assessment
  - Ensure managers assess their management processes and identify and correct problems that hinder the organization from achieving its objectives.
- Criterion 10—Assessment/Independent Assessment
  - Plan and conduct independent assessments to measure item and service quality, to measure the adequacy of work performance, and to promote improvement
  - Establish sufficient authority, and freedom from line management, for the group performing independent assessments
  - Ensure persons who perform independent assessments are technically qualified and knowledgeable in the areas to be assessed.

Safety Basis

The following is taken from 10 CFR 830.201.

A contractor must perform work in accordance with the safety basis for a hazard category 1, 2, or 3 DOE nuclear facility and, in particular, with the hazard controls that ensure adequate protection of workers, the public, and the environment.

e. Discuss the role of the Department's nuclear safety specialists with respect to implementing the requirements of the Price-Anderson Amendment Act.

The following is taken from DOE-STD-1183-2007.

The NSS must participate in the oversight of contractor implementation of 10 CFR 830, the Nuclear Safety Management Rule, subpart B. The NSS must evaluate the nuclear safety posture of nuclear facilities and operations for PAAA compliance, contract performance, operational readiness reviews, readiness assessments, and other periodic assessments. They must participate in enforcement of PAAA requirements.

The following is taken from 10 CFR 820, appendix A.

The DOE goal in the compliance arena is to enhance and protect the radiological health and safety of the public and workers at DOE facilities by fostering a culture among the DOE line organizations and the contractors that actively seeks to attain and sustain compliance with DOE nuclear safety requirements. The enforcement program and policy has been developed with the express purpose of achieving safety inquisitiveness and voluntary compliance. DOE will establish effective administrative processes and positive incentives to the contractors for

the open and prompt identification and reporting of noncompliances, and the initiation of comprehensive corrective actions to resolve the noncompliance conditions and the program or process deficiencies that led to noncompliance.

26. Nuclear safety specialists must demonstrate a working level knowledge of the requirements in DOE technical standard DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis*, and DOE-STD-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*.

[Note: DOE-STD-3009-94 has been superseded by DOE-STD-3009-94, Chg 3, and the title is now *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*. Presumably DOE-HDBK-3010-94 is meant instead of DOE-STD-3010-94.]

a. Discuss the conceptual basis and process for preparation of a facility/activity DSA.

The following is taken from DOE-STD-3009-94, Chg 3.

DOE-STD-3009-94, Chg 3, incorporates and integrates many different approaches regarding DSA format and content. To ensure a consistent application of DOE-STD-3009-94, Chg 3 among users, the following guiding principles are provided:

- Focus is primarily on hazard category 2 and 3 facilities.
- Hazard and accident analyses are merged into one chapter (chapter 3) to ensure that the proper emphasis is placed on identification and analysis of hazards. The HA distinguishes when accident analysis is required as a function of potential offsite consequences. Guidance for hazard and accident analysis is not based on PRA.
- DID, work safety, and environmental issues are identified.
- DID as discussed in DOE-STD-3009-94, Chg 3, consists of two components:
  1. Equipment and administrative features providing preventive or mitigative functions so that multiple features are relied on for prevention or mitigation to a degree proportional to the hazard potential
  2. ISM programs that control and discipline operation
- Guidance is provided for evaluating the safety of a facility where documentable DBAs do not exist to establish bounding accidents (derivative DBAs) that envelope the safety of existing facilities. Guidance is provided on the treatment of beyond DBAs.
- Distinction is made between SC SSCs and SS SSCs and the balance of facility SSCs. SC SSCs are related to public protection and are defined by comparison with the numerical EG. SS SSCs are identified for specific aspects of DID and worker safety as determined by the HA. Specific definitions are provided for these two terms.
- Guidance is provided identifying ACs that are major contributors to DID that are designated as SACs. DOE-STD-3009-94, Chg 3, and DOE-STD-1186-2004 provide guidance applicable to these types of controls. SACs provide preventive and/or mitigative functions for specific potential accident scenarios that have safety

importance equivalent to engineered controls that would be classified as SC or SS if the engineered controls were available and selected.

- Consequences from normal operations are addressed in the radiation protection, hazardous material protection, and waste management chapters.
- Guidance is provided in each chapter on the application of the graded approach.
- A common DSA format for all nonreactor nuclear facilities is desirable but not essential. A table is to be provided by the preparer that indicates where the DSA requirements of 10 CFR 830 are addressed. Content needs to be flexible to allow for different facility types, hazard categories, and other grading factors.
- Facility descriptive material is intentionally split to emphasize SSCs of major significance:
  - Chapter 2, "Facility Description," provides a brief, integrated overview of the facility SSCs.
  - Chapter 4, "Safety Structures, Systems, and Components," provides detailed information only for those SSCs that are SC and SS. This application of the graded approach will provide for a significant reduction of DSA volume, while maintaining a focus on safety.
- The programmatic chapters, including chapters 6–17, provide a summary description of the key features of the various safety programs as they related to the facility being analyzed. These chapters are not meant to be used as the vehicle for the determination of adequacy of these programs.

b. Discuss the following in relation to the preparation of the DSA:

- **Worker safety**
- **Defense-in-depth**
- **Programmatic commitments**
- **TSRs**
- **SSCs**
- **Hazard analysis**
- **Accident analysis**
- **Application of the graded approach**
- **Safe harbor methods**

Worker Safety

The following is taken from DOE-STD-3009-94, Chg 3.

Workers, typically those in proximity to operations, are the population principally at risk from potential consequences associated with hazard category 2 and 3 facilities. The DOE recognizes, via 10 CFR 830, the importance of including worker safety in safety analyses by specifically noting the worker as a population of concern. Developing a conceptual basis for the methodology used in DOE-STD-3009-94, Chg 3, requires answering the fundamental question of how worker safety is most appropriately addressed in the DSA.

OSHA has published 29 CFR 1910.119. OSHA defines the purpose of this regulation in summary fashion as, "Employees have been and continue to be exposed to the hazards of toxicity, fires, and explosions from catastrophic releases of highly hazardous chemicals in their workplaces. The requirements in DOE-STD-3009-94, Chg 3, are intended to eliminate or mitigate the consequences of such releases." Many of the topics requiring covering in this

Federal regulation, such as design codes and standards, process HA, human factors, training, etc., are directly parallel to the requirements in 10 CFR 830.

DOE O 440.1B, Admin Chg 1, *Worker Protection Program for DOE (Including the National Nuclear Security Administration) Federal Employees*, and 10 CFR 851 address the issue of work safety from process accidents by requiring the performance of HAs for processes (exclusive of standard industrial hazards) in conjunction with implementation of basic safety programs that discipline operations and ensure judgments made in HAs are supported by actual operating conditions. These requirements effectively integrate programs and analyses into an overall safety management structure without requiring quantitative risk assessment. This integration and the basic concepts of PSM described by OSHA regulations and the manuals and codes of practice described in DOE O 440.1B, Admin Chg 1, are philosophically accepted as appropriate for DSAs. DOE-STD-3009-94, Chg 3, effectively merges PSM principles with traditional DSA precepts.

Defense-in-Depth

The following is taken from DOE-STD-3009-94, Chg 3.

DID as an approach to facility safety has extensive precedent in nuclear safety philosophy. It builds in layers of defense against release of hazardous materials so that no one layer by itself, no matter how good, is completely relied on. To compensate for potential human and mechanical failures, DID is based on several layers of protection with successive barriers to prevent the release of hazardous material to the environment. This approach includes protection of the barriers to avert damage to the plant and to the barriers themselves. It includes further measures to protect the public, workers, and the environment from harm in case these barriers are not fully effective.

The DID philosophy is a fundamental approach to hazard control for nonreactor nuclear facilities even though they do not possess the catastrophic accident potential associated with nuclear power plants. In keeping with the graded-approach concept, no requirement to demonstrate a generic, minimum number of layers of DID is imposed. However, defining DID as it exists at a given facility is crucial for determining a safety basis. Operators of DOE facilities need to use the rigorous application of DID thinking in their designs and operations. Such an approach is representative of industrial operations with an effective commitment to public and worker safety and the minimization of environmental releases.

For high hazard operations, there are typically multiple layers of DID. The inner layer of DID relies on a high level of design quality so that important
SSCs will perform their required functions with high reliability and high tolerance against degradation. The inner layer also relies on competent operating personnel who are well trained in operations and maintenance procedures. Competent personnel translate into fewer malfunctions, failures, or errors; thus, minimizing challenges to the next layer of defense.
In the event that the inner layer of DID is compromised from either equipment malfunction (from whatever cause) or operator error and there is a progression from the normal to an abnormal range of operation, the next layer of DID is relied upon.

It can consist of 1) automatic systems; or 2) means to alert the operator to take action or manually activate systems that correct the abnormal situation and halt the progression of events toward a serious accident.

Mitigation of the consequences of accidents is provided in the outer layer of DID. Passive, automatically, or manually activated features (e.g., containment or confinement system, deluge systems, filtered exhaust), and/or safety management programs (i.e., emergency response) minimize consequences in the event that all other layers have been breached. The contribution of emergency response actions to minimizing consequences of a given accident cannot be neglected as they represent a truly final measure of protection for releases that cannot be prevented.

SSCs that are major contributors to DID are designated as SS SSCs. Additionally, DOE-STD-3009-94, Chg 3, provides guidance on grading the safety management programs (e.g., radiation protection, hazardous material protection, maintenance, procedures, training) that a facility must commit to compliance in order to establish an adequate safety basis. The discipline imposed by safety management programs goes beyond merely supporting the assumptions identified in the HA and is an integral part of DID.

ACs that are major contributors to DID are designated as SACs that are required for safety because they are the basis for validity of the hazard or accident analyses, or they provide the main mechanisms for hazard control. DOE-STD-3009-94, Chg 3, and DOE-STD-1186-2004 provide guidance applicable to these types of controls. SACs provide preventive and/or mitigative functions for specific potential accident scenarios that have safety importance equivalent to engineered controls that would be classified as SC or SS if the engineered controls were available and selected. The established hierarchy of hazard controls requires that engineering controls with an emphasis on safety-related SSCs be preferable to ACs or SACs due to the inherent uncertainty of human performance. SACs may be used to help implement a specific aspect of a program AC that is credited in the safety analysis and therefore has a higher level of importance.

In accordance with nuclear safety precepts, a special level of control is provided through use of TSRs. DOE G 423.1-1A provides screening criteria for converting existing technical specifications and operational safety requirements into TSRs. For the purposes of this DOE-STD-3009-94, Chg 3, the screening criteria are considered a generally reasonable set of criteria to designate TSRs for DID. The safety items identified in the HA are examined against those criteria to identify a subset of the most significant controls that prevent uncontrolled release of hazardous materials and nuclear criticality. These TSR controls may be captured in operational limits or in ACs, including those on safety management programs. This collection of TSRs formally acknowledges features that are of major significance to DID.

Programmatic Commitments

10 CFR 830.204 requires that the DSA define the characteristics of the safety management programs necessary to ensure the safe operation of the facility. Program commitments (e.g., radiation protection, maintenance, QA) encompass a large number of details that are more appropriately covered in specific program documents (e.g., plans and procedures) external to

the DSA. The cumulative effect of these details, however, are recognized as being important to facility safety that is the rationale for a top-level program commitment becoming part of the safety basis.

As appropriate to the hazard, the safety basis may identify specific controls (e.g., hazardous material inventory limits) that are required for safety. These controls should be considered for designation as an SAC as discussed in DOE-STD-3009-94, Chg 3, and DOE-STD-1186.

The importance of the program commitments that can be incorporated in TSRs as ACs cannot be overestimated. The safety basis, however, includes only the top-level summary of program elements, not the details of the program or its governing documents. Inspection discrepancies in a program would not constitute violation of the safety basis unless the discrepancies were so gross as to render premises of the summary invalid.

By virtue of application of the graded approach, the majority of the engineered features in a facility will not be identified in the categories of SC or SS SSCs even though they may perform some safety functions. However, such controls noted as a barrier or preventive or mitigative feature in the hazard and accident analyses must not be ignored in managing operations. Such a gross discrepancy would violate the safety basis documented in the DSA even if the controls are not designated SC or SS, because programmatic commitments extend to these SSCs as well. For example, the commitment to a maintenance program means that the preventive and mitigative equipment noted as such in the DSA HA is included in the facility maintenance program. As a minimum, all aspects of DID identified must be covered within the relevant safety management programs (e.g., maintenance, QA) committed to in the DSA. The details of that coverage, however, are developed in the maintenance program as opposed to in the DSA. Facility operators are expected to have noted the relative significance of these engineered features and have provided for them in programs, in keeping with standard industrial practice, based on the importance of the equipment. It is the fact of coverage that is relevant to the facility safety basis. The details of this programmatic coverage (i.e., exact type of maintenance items and associated periodicities) are not developed in or part of the DSA.

An overall commitment made in a DSA is that the contractor will not change the facility configuration underlying the documented safety basis without implementing and completing the USQ process. However, situations do occur where a USQ process is not necessary. For example, a stipulation to have a radiation protection program in the AC section of the TSR is a commitment; however, changes to specific program provisions do not require going through the USQ process. Further clarification of such interpretations can be found in DOE G 424.1-1B, Admin Chg 2.

TSRs and SSCs

The following is taken from DOE-STD-3009-94, Chg 3.

To comply with 10 CFR 830, specific safety controls are to be developed in the DSA. In keeping with the graded-approach principle, distinctions are made to avoid wasting effort by providing detailed descriptions of all facility SSCs. While a basic descriptive model of the facility and its equipment must be provided in chapter 2, “Facility Description,” highly

detailed descriptions are reserved for two categories of SSCs comprising the most crucial aspects of facility safety. These two categories are SC and SS SSCs.

Detailed descriptions are provided for SC and SS SSCs and SACs in chapter 4 of the DSA because of the importance of their safety functions. Descriptions result in the definition of functional requirements and associated performance criteria used to derive TSRs. TSRs are safety controls developed in accordance with the precepts of 10 CFR 830. TSR and SSC commitments encompass the following:

- Technical safety requirements—TSRs comprise 1) SLs; 2) LCSs, LCOs, and associated SRs; 3) ACs; 4) SACs; 5)use and application provisions; 6) design features; and 7) bases appendix. Based on the results of hazard and accident analysis TSRs are designated for 1) S SSCs and controls established on the basis of application of the EG; 2) SS SSCs; 3) DID in accordance with the screening criteria of DOE G 423.1-1B, Admin Chg 2; and 4) safety management programs for DID or worker safety. The bases appendix provides the linkage to the DSA. It is important to develop TSRs judiciously. TSRs should not be used as a vehicle to cover the many procedural and programmatic controls inherent in any operation. Excessive use of TSR limits to manage operations will result in distortion of the regulatory structure DOE is attempting to develop and will dilute the emphasis intended for the most critical controls. SLs should be limited in number and designated with caution. In accordance with table 4 of 10 CFR 830, appendix A to subpart B, SLs are generally reserved for limits on process variables associated with those SC 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. The associated OLs apply to active SSCs that prevent exceeding SLs. The only candidates for SLs should be SC SSCs and any non-SSC controls established on the basis of the application of the EG. Nuclear industry precedent is only a limited subset of SC SSCs, if any, require definition of associated SLs that are intended to prevent significant accidents as opposed to mitigating their effects. TSRs assigned for DID or SS SSCs do not have SLs and are not required to use operational limits (i.e., LCSs, LCOs). They should, however, receive coverage in the AC section of TSRs as a minimum. Judgment should be used to determine what controls warrant use of operational limits. When TSR ACs are used for purposes other than generic coverage of safety management programs (e.g., SAC), descriptions should be sufficiently detailed that a basic understanding is provided of what is controlled and why. Beyond SS SSCs designated for worker safety and their associated TSR coverage, additional worker safety issues should be covered in TSRs only by ACs on overall safety management programs.
- SC SSCs—The rule defines SC designation for SSCs that are established on the basis of application of the EG. This designation carries with it the most stringent requirements (e.g., enhanced inspection, testing and maintenance, and special I&C systems). 10 CFR 830, appendix A to subpart B, provides guidance for implementing the EG to classify SSCs as SC SSCs.
- SS SSCs—This category of SSCs is provided to ensure that important SSCs will be given adequate attention in the DSA and facility operations programs. SS SSCs are those of particular importance to DID or worker safety as determined in the HA. Control of such SSCs does not require meeting the level of stringency associated with

SC SSCs. The EG is not used for designating SS SSCs. SC SSCs are designated to address public risk that makes a dose guideline at the site boundary a useful tool. SS SCs address risk for all individuals within the site boundary as well as additional DID for the public, making a dose guideline at any one point an artificial distinction distorting the process of systematically evaluating SSCs. TSRs covering SSCs ensuring DID should generally correlate with SS SSC designation for DID, but exact one-to-one correlation is not required.

- Specific ACs—This category of ACs is provided to ensure that controls important to safety that are needed to prevent or mitigate an accident scenario will be given equivalent attention in the safety basis documents had that safety function been provided by an SC or SS SSC. Safety analyses shall establish the identification and functions of SACs and the significances to safety of the functions of the SAC. The established hierarchy of hazard controls requires that engineering controls with an emphasis on safety-related SSCs be preferable to ACs or SACs due to the inherent uncertainty of human performance. SACs may be used to help clarify and implement an AC.

Hazard Analysis

The following is taken from DOE-STD-3009-94, Chg 3.

The initial analytical effort for all facilities is an HA that systematically identifies facility hazards and accident potentials through hazard identification and hazard evaluation. The focus of the HA is on thoroughness and requires evaluation of the complete spectrum of hazards and accidents. This largely qualitative effort forms the basis for the entire safety analysis effort, including specifically addressing DID and protection of workers and the environment. Basic industrial methods for HA, its interface with more structured quantitative evaluations, and the basis for both have been described in references such as the American Institute of Chemical Engineers *Guidelines for Hazard Evaluation Procedures*. OSHA has accepted these guidelines as the standard for analytical adequacy in characterizing commercial chemical processes that perform the same type of unit operations conducted at DOE nonreactor nuclear facilities. Appropriately applied, they help fulfill the requirements of DSAs for hazard category 2 and 3 facilities as specified in 10 CFR 830.

The largely qualitative techniques described in the above reference on HA provide methodologies for comprehensive definition of the accident spectrum for workers and the public. The basic identification of hazards inherent in the process provides a broad, initial basis for identification of safety programs needed (e.g., radiation protection, hazardous chemical protection). The HA then moves beyond basic hazard identification to evaluation of the expected consequences and estimation of likelihood of accidents, an activity that in no way connotes the level of effort of a probabilistic or quantitative risk assessment.

Throughout the evaluation process, preventive and mitigative SSCs and SACs and pertinent elements of programmatic controls are identified. This identification establishes functional requirements that will subsequently delineate the technical information needed to establish performance criteria. The DSA summarizes these requirements and criteria for SC and SS SSCs and SACs only. Refinement of the information obtained in hazard evaluation leads to overall definition of DID, worker safety, and environmental protection.

The most significant aspects of DID and worker safety are subject to designation as SC SSCs and coverage by TSRs. Other items noted are encompassed by the details of safety management programs that can be captured in top-level fashion in TSR ACs. However, programmatic ACs should not be used to provide preventive or mitigative functions for accident scenarios identified in the safety basis where the safety function has importance similar to, or the same as, the safety function of SC or SS SSCs. The classification of SAC was specifically created for this safety function. The hazard evaluation conducted to assess the accident spectrum associated with hazards germane to the DSA indicates the adequacy of programmatic efforts and provides input to programmatic activities whose discipline provides a significant margin of safety.

The process outlined above is self-grading for analytical effort. Analytical effort can be limited to a simple, resource efficient HA geared to facility needs, unless events are noted that are of sufficient complexity to require more detailed, quantitative evaluations to understand the basis for safety assurance. Implicit in this methodology is the statement of DOE-STD-1027-92, Chg 1, that the largely qualitative level of effort in HA is appropriate and sufficient for accident analysis of hazard category 3 facilities. It is again noted that the HA effort is not a qualitative risk assessment. Preparers (and subsequent reviewers) cannot expect the level of detail associated with a quantitative risk assessment in a HA, as the HA is focused on systematically assessing what can go wrong in a facility as opposed to deriving mathematical expressions of risk.

The final purpose of HA is to identify a limited subset of accidents to be carried forward to accident analysis.

Accident Analysis

The following is taken from DOE-STD-3009-94, Chg 3.

The complete spectrum of accidents is examined in HA. A limited subset of accidents (i.e., DBAs and derivative DBAs) that bound the envelope of accident conditions to which the operation could be subjected are carried forward to accident analysis where SC SSCs are designated by comparison of accident consequences to the EG. Identification of DBAs in safety analysis and use of DBAs is appropriate in defining a facility safety basis. DBAs are accidents that are used to provide the design parameters for release barriers and mitigating systems. DBAs are a “front-end” device for designing individual equipment or systems to meet functional requirements, as evidenced by use of the phrase “utilized to provide the design parameters.” An accident can be defined as a DBA if relevant SSCs were specifically designed to function during that accident and appropriate documentation of this fact exists.

The range of accident scenarios analyzed in a DSA should be such that a complete set of bounding conditions to define the envelope of accident conditions where the operation could be subjected are evaluated and documented. This necessitates the consideration of accidents other than DBAs for two cardinal reasons. First, even if DBAs exist, they may not adequately cover the range of accident scenarios needed to establish the facility safety basis. Secondly, DBAs may not cover a complete set of bounding conditions. Either of these conditions may arise for a number of reasons, such as the original design not being related to bounding conditions, the criteria for determining facility safety basis having significantly changed,

operations or types of hazards having changed, or magnitude of hazards having increased. Any one of these reasons may make the DBA inadequate for determining a facility safety basis.

The most obvious and extreme reason for examining accidents other than DBAs for existing facilities is a lack of design documentation. If appropriate design documentation is not available, postulated accidents are not DBAs. The front-end purpose of a DBA cannot be meaningfully addressed even if existing design parameters are estimated and used to develop an accident scenario. The reconstructed accident would not determine design parameters. It would be determined by them. The need to analyze a range of scenarios that bound conditions would not clearly be met by such an exercise. This potential lack of relevance is one of the reasons that the DSA is not the proper vehicle for formally filling gaps in existing design documentation.

Where DBAs do not exist, or do not adequately cover the range of scenarios or bounding conditions, surrogate evaluation bases are needed. These derivative DBAs are used to estimate the response of SSCs to the range of accident scenarios and stresses that bound the envelope of accident conditions to which the facility could be subjected in order to evaluate accident consequences. The derivative DBAs should take maximum advantage of the pertinent existing design information (i.e., requirements and bases) that is immediately available or can be retrieved through reasonable efforts. To the extent necessary, this information can be supplemented by testing, extrapolation, and engineering judgments.

Existing facilities, like all industrial facilities, were generally built with standard process and utility SSCs with a high consideration for basic safety. For the majority of these facilities, adequate facility design and process information exist that, while not of the quality and detail expected for current conceptual design, is typical of many commercial processing operations that comprise the majority of industrial practices. This information can be used in estimating SSC response to derivative DBAs whose evaluation will satisfy the requirements of safety analysis.

For operational accidents, a derivative DBA is defined based on the physical possibility of phenomena as defined in the HA. Use of a lower binning threshold such as 10^{-6}/yr is generally appropriate, but should not be used as an absolute cutoff for dismissing physically credible low probability operational accidents without any evaluation of preventive and mitigative features in HA. This distinction is made to prevent “pencil sharpening” at the expense of objective evaluation of hazards. Examples of a candidate derivative DBA would be an ion exchange column or a red oil explosion at a facility where the phenomena is physically possible and documentation is not available substantiating ventilation and building confinement systems were specifically designed for such an occurrence. For natural event accidents, derivative DBAs are defined by a frequency of initiator based on DOE O 420.1 and its associated implementation standards. For external man-made accidents, derivative DBAs are assumed if the event can occur with a frequency $>10^{-6}$/yr as conservatively estimated, or $>10^{-7}$/yr as realistically estimated. Use of a frequency cutoff for external events represents a unique case for external events only, based on established NRC precedents. For simplicity, use of the term DBA throughout DOE-STD-3009-94, Chg 3, is inclusive of DBAs and derivative DBAs.

Application of the Graded Approach

The following is taken from DOE-STD-3009-94, Chg 3.

10 CFR 830 prescribes the use of a graded approach for the effort expended in safety analysis and the level of detail presented in associated documentation. The graded approach applied to DSA preparation and updates is intended to produce cost-efficient safety analysis and DSA content that provides adequate assurance to DOE that a facility has acceptable safety provisions without providing unnecessary information. As described in 10 CFR 830, the graded approach adjusts the magnitude of the preparation effort to the characteristics of the subject facility based on the following seven factors:

1. The relative importance to safety, safeguards, and security
2. The magnitude of any hazard involved
3. The life-cycle stage of a facility
4. The programmatic mission of a facility
5. The particular characteristics of a facility
6. The relative importance of radiological and non-radiological hazards
7. Any other relevant factor

The rule provides for developing the DSA based on judgment of the facility in relation to these seven factors. For example, simple hazard category 3 facilities or facilities that have a short operational life may only require a limited, but adequate, analysis documented to a level less than that required for a hazard category 2 facility. In addition, facilities with short operational lives should consider the appropriateness of using DOE-STD-3011-2002 to meet the requirements of 10 CFR 830. On the opposite end of the spectrum, a complex hazard category 1 facility that is just going in operation requires extensive analysis and highly detailed documentation.

The application of the graded approach may allow for much simpler analysis and documentation for some of these facilities. For facilities of little hazard, or hazards at the hazard category 3 level, for which only a modest reduction of risk is required, the DSA may be simple and short. In such cases, all the topics for the DSA listed in DOE-STD-3009-94, Chg 3, may not be necessary and with proper technical bases some topics may be omitted or reduced in the detail that would otherwise be required of hazard category 1 or 2 facilities.

Thus, with application of the graded approach, DSAs for hazard category 3 facilities or facilities with short operational lives will normally require more simplified DSA analysis and documentation. Specific minimum levels of detail for these facilities are given in options #3 and #8 in 10 CFR 830, appendix A to subpart B, table 2 and the graded approach section of each chapter in DOE-STD-3009-94, Chg 3. As a minimum, a DSA would be found acceptable for a simple hazard category 3 facility if it used the methods in DOE-STD-3009-94, Chg 3, chapters 2, 3, 4, and 5 to address in a simplified fashion

- the basic description of the facility and its operations, including SC SSCs;
- a qualitative hazards analysis; and
- the hazard controls (consisting primarily of inventory limits and safety management programs) and their bases.

The following is taken from DOE O 232.2

Occurrences must be investigated and analyzed using a graded approach in accordance with locally approved quality and issues management procedures. Facility managers must consider the significance or potential significance (e.g., significance category) of the event when choosing the scope and tools to use in the investigation. The investigation and analysis methodology(ies) must be included in the report's "Description of Cause" field.

Safe Harbor Methods

The following is taken from DOE G 421.1-2A.

Table 11. Safe harbor methods

The contractor responsible for	May prepare its DSA by
(1) a DOE reactor	using the method in U.S. NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*, or successor document.
(2) a DOE nonreactor nuclear facility	using the method in DOE-STD-3009-94, Chg 3, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, dated July 1994, or successor document.
(3) a DOE nuclear facility with a limited operational life[1]	using the method in either (1) DOE-STD-3009-94, Chg 3, or successor document; or (2) DOE-STD-3011-94, *Guidance for Preparation of DOE 5480.22 (TSR) and DOE 5480.23 (SAR) Implementation Plans*, dated November 1994, or successor document.
(4) the deactivation or the transition surveillance and maintenance of a DOE nuclear facility	using the method in either (1) DOE-STD-3009-94, Chg 3, or successor document; or (2) DOE-STD-3011-94 or successor document.
(5) the decommissioning of a DOE nuclear facility	(1) using the method in DOE-STD-1120-2005 or successor documents; (2) using the provisions in 29 CFR 1910.120 (or 29 CFR 1926.65 for construction activities) for developing S&H programs, work plans, HASPs, and emergency response plans to address public safety, as well as worker safety; and (3) deriving hazard controls based on the S&H programs, the work plans, the HASPs, and the emergency response plans.
(6) a DOE environmental restoration activity that involves either work not done within a permanent structure or the decommissioning of a facility with only low-level residual fixed radioactivity.	(1) using the method in DOE-STD-1120-2005 or successor document; and (2) using the provisions of 29 CFR 1910.120 (or 29 CFR 1926.65 for construction activities) for developing a S&H program and a site-specific HASP (including elements for emergency response plans, conduct of operations, training and qualifications, and maintenance management).
(7) a DOE nuclear explosive facility and the nuclear explosive operations conducted therein	developing its DSA in two pieces: (1) an SRA for the nuclear facility that considers the generic NEOs and is prepared in accordance with DOE-STD-3009-94, Chg 3, or successor document; and (2) an HAR for the specific NEOs prepared in accordance with DOE-NA-STD-3016-2006 or successor document.

(8) a DOE hazard category 3 nonreactor nuclear facility	using the methods in DOE-STD-3009-94, Chg 3, chapters 2, 3, 4, and 5, or successor document to address in a simplified fashion: (1) the basic description of the facility/activity and its operations, including safety SSCs; (2) a qualitative hazards analysis; and (3) the hazard controls (consisting primarily of inventory limits and safety management programs) and their bases.
(9) transportation activities	(1) preparing an SAR for packaging in accordance with DOE O 460.1C, *Packaging and Transportation Safety*, or successor document; and (2) preparing a transportation safety document in accordance with DOE G 460.1-1, or successor document.
(10) transportation and onsite transfer of nuclear explosives, nuclear components, Naval nuclear fuel elements, category I and category II special nuclear materials, special assemblies, and other materials of national security	(1) preparing an SARP in accordance with DOE O 461.1B, or successor document; and (2) preparing a transportation safety document in accordance with DOE O 461.2, *Onsite Packaging and Transfer of Materials of National Security Interest.*

[1]A facility with a limited operational life is one that has an approved deactivation plan (removal of hazards) calling for cessation of operations within a stated period (five years). This plan should include required funding action and plan change control to ensure relevancy.
Source: DOE G 421.1-2A

Each of the safe harbors has a methodology specific to the application to satisfy the requirements for the development of a DSA as described in 10 CFR 830.204 for the hazards identification, safety analysis, and derivation of hazard controls. DOE-STD-3009-94, Chg 3, is a safe harbor for any of the specialized areas covered by the other safe harbors (with the exception of hazard category 1 nuclear reactors) and can be used in lieu of any of them. An expectation associated with any of the safe harbors is that the safety classification guidance for SC SSCs and SACs of DOE-STD-3009-94, Chg 3, will be used in developing the DSA.

As long as a facility is a category 1, 2, or 3 nuclear facility, in accordance with 10 CFR 830, subpart B, it must have a safety basis, including a DSA, hazard controls (including TSRs), and a USQ process. In its life-cycle, a nuclear facility generally has a mission-oriented or production phase, after which it is shut down and either devoted to another mission or is declared excess and enters in a disposition process. Transition activities occur between operations and ultimate disposition.

c. Discuss the relationship between the safe harbor methods for a DSA in 10 CFR 830, Appendix A with regard to completeness.

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 (including the design of safety SSCs) 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 that 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 criticality safety 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.

d. Describe the objectives of requiring accident analyses in safety basis documents.

The following is taken from DOE-STD-3009-94, Chg 3.

Accident analysis has historically consisted of the formal development of numerical estimates of the expected consequences and probability of potential accidents associated with a facility. For the purposes of implementing DOE-STD-3009-94, Chg 3, accident analysis is a follow-on effort to the HA, 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 HA and performance of a formally documented consequence analysis. Consequences are compared with the EG to identify SC SSCs.

e. Identify and discuss the use of the source term five-factor formula in accident analyses.

The following is taken from DOE-HDBK-3010-94.

The source term is the MAR (as derived from the hazard identification), the release fraction or rate, and the overall facility LPFs that determine the fraction of material released external to the facility.

The airborne pathway is of primary interest for nonreactor nuclear facilities. DOE-STD-1027-2002 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.”

$$\text{Source term} = \text{MAR} \times \text{DR} \times \text{ARF} \times \text{RF} \times \text{LPF}$$

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 = Leak path 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 LPF of the specific stage.

DOE-HDBK-3010-94 assesses ARF and RF values separately for sources of airborne material generated from accidents involving gases, liquids, solids, and surface contamination. All of the above factors may need to be determined for particulate releases. Some of them, however, will collapse to values of one for special cases (e.g., gaseous releases).

Material-at-Risk (MAR)

The MAR is the amount of radionuclides (in grams or curies of activity for each radionuclide) available to be acted on by a given physical stress. For facilities, processes, and activities, the MAR is a value representing some maximum quantity of radionuclide present or reasonably anticipated for the process or structure being analyzed. Different MARs may be assigned for different accidents as it is only necessary to define the material in those discrete physical locations that are exposed to a given stress. For example, a spill may involve only the contents of a tank in one glovebox. Conversely, a seismic event may involve all of the material in a building.

Damage Ratio (DR)

The DR is the fraction of the MAR actually impacted by the accident-generated conditions. A degree of interdependence exists between the definitions of MAR and DR. If it is predetermined that certain types of material would not be affected by a given accident, some analysts will exclude this material from the MAR.

The DR is estimated based on engineering analysis of the response of structural materials and materials-of-construction for containment to the type and level of stress/force generated by the event. Standard engineering approximations are typically used. These approximations often include a degree of conservation due to simplification of phenomena to obtain a useable model, but the purpose of the approximation is to obtain, to the degree possible, a realistic understanding of potential effects.

Airborne Release Fraction (ARF)

The ARF is the coefficient used to estimate the amount of a radioactive material suspended in air as an aerosol and thus available for transport due to physical stresses from a specific accident. For discrete events, the ARF is a fraction of the material affected. For mechanisms that continuously act to suspend radionuclides (e.g., aerodynamic entrainment/resuspension), a release rate is required to estimate the potential airborne release from postulated accident conditions. Generally, accident release rates are based on measurements over some extended

period to encompass most release situations for a particular mechanism. The rates are average rates for the broad spectrum of situations and, as such, the most typically meaningful time unit to reflect average conditions is one hour. There is evidence that in some situations (e.g., aerodynamic entrainment of sparse powder deposits on a heterogeneous surface), the rate of release is not uniform with time. Even in the situations where the rates are relatively uniform, the source is depleted by the removal of particles from the surface by aerodynamic forces, and the amount of material airborne decreases with time unless the source is continuously replenished. The ARFs are based primarily on experimentally measured values for the specific material (e.g., plutonium, uranium, mixed fission products) or surrogates subjected to the particular type of stress under controlled conditions. Attention is given to the parameters, if known, that may have a significant influence on suspension by the specific mechanism and the uncertainty in the measurement as indicated by the variability of the results.

Respirable Fraction (RF)

The RF is the fraction of airborne radionuclides as particles that can be transported through air and inhaled into the human respiratory system and is commonly assumed to include particles 10-µm AED and less. Other definitions of "respirable particles" have been presented by various groups at different times. Use of a 10-µm AED cut-size for respirable particles is considered conservative, and may even be overly conservative since the mass is a cube function of particle diameter.

Leak Path Factor (LPF)

The LPF is the fraction of the radionuclides in the aerosol transported through some confinement deposition of filtration mechanism. There can be many LPFs for some accident conditions (e.g., the fraction transported from the package, such as a shipping container, to the cell or enclosure; the fraction leaked from the enclosure, cell, or glovebox to the operating area around the enclosure or room; the fraction leaked from the room to the building-atmosphere interface). Where multiple leak paths are involved, their cumulative effect is often expressed as one value that is the product of all leak path multiples. The LPF is a calculated or standard value based on 1) established relationships between size of the particulate material, airborne transport mechanisms, and losses by deposition mechanisms; or 2) specified filtration efficiencies.

f. Given an accident scenario, determine a reasonably bounding estimate of the airborne release fraction (ARF) and respirable fraction (RF), material-at-risk (MAR), leak path factor (LPF), and damage ratio (DR) to determine the product (MAR x ARF x RF x DR x LPF).

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

The information provided in element e of this competency may be helpful.

g. Identify and discuss the methods, conventions, and data sources used in developing estimates of the five factors for use in accident analyses.

See element e of this competency for a discussion of the methods, conventions, and data sources used in developing estimates of the five factors for use in accident analyses.

h. Identify and discuss methods/codes used to determine the environmental dispersion and delivered doses from accidental releases of hazardous materials.

The following is taken from DOE-STD-3009-94, Chg 3.

The 9th percentile of the distribution of doses to the MOI, accounting for variations in distance to the site boundary as a function of direction, is the comparison point for assessment against the EG.

The method used should be consistent with the statistical treatment of calculated X/Q values described in NRC Regulatory Guide 1.145, regulatory position 3, for the evaluation of consequences along the exclusion area boundary. The determination of distance to the site boundary should be made in accordance with the procedure outlined in NRC Regulatory Guide 1.145, regulatory position 1.2. NRC Regulatory Guide 1.23, revision 1, *Meteorological Monitoring Programs for Nuclear Power Plants*, describes acceptable means of generating the meteorological data on which dispersion is based. Accident phenomenology may be modeled assuming straight-line Gaussian dispersion characteristics, applying meteorological data representing a one-hour average for the duration of the accident. Accident duration is defined in terms of plume passage at the location of dose calculation, for a period not to exceed eight hours.

Prolonged effects, such as resuspension, need not be modeled. The accident progression should not be defined so that the MOI is not substantially exposed (i.e., using a release rate that is specifically intended to expose the MOI to only a small fraction of the total material released, or defining time and wind speed so that the plume has not reached the MOI). The exposure period begins from the time the plume reaches the MOI.

For ground releases, the calculated dose equates to the centerline dose at the site boundary. For elevated, thermally buoyant, or jet releases, plume touchdown may occur beyond the site boundary. These cases should locate the dose calculation at the point of maximum dose beyond the site boundary that is typically at the point of plume touchdown.

Accidents with unique dispersion characteristics, such as explosions, may be modeled using phenomenon-specific codes more accurately representing the release conditions. Discussion should be provided justifying the appropriateness of the model to the specific situation. For accident phenomena defined by weather extremes, actual meteorological conditions associated with the phenomena may be used for comparison to the EG.

i. Discuss the effect of prevailing weather, building wake effects, and plume buoyancy upon the magnitude and distribution of doses from hazardous releases into the atmosphere.

The following is taken from Environmental Protection Agency, *Air Toxics Risk Assessment Reference Library*, volume 1, "Technical Resource Manual," chapter 8, "Quantification of Exposure: Dispersion, Transport, and Fate of Air Toxics in the Atmosphere."

EPA and others have developed several air dispersion models to predict the often complex behavior of air pollution releases. Most air dispersion models take into account a number of

characteristics of the source and pollutants released. The most common of these characteristics are described below.

Release Rate and Volume

The rate of release (exit velocity) strongly influences the behavior of the pollutant plume as it moves through the atmosphere. In the case of stack and vent releases, sources can release pollutants as pure vapors, as dilute solutions of vapor in air or other gases, or as suspended particles. Large volumes are often released at a relatively high velocity from stacks or vents driving pollutants higher in the atmosphere. Air quality models often calculate volume from data on exit area and exit velocity. In case of fugitive releases or volatilization, the volume of release has less meaning, and often does not receive consideration in fate and transport modeling.

Concentration

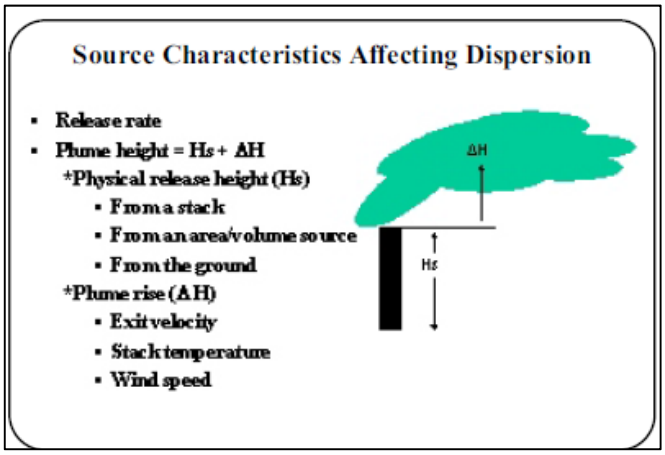
Concentration (the mass of pollutant per unit volume of released gases) is the other half of the equation that determines the amount (mass) of pollutant released. Pollutants at higher concentrations may be more likely to condense onto particles or liquid droplets.

Temperature

The temperature of a plume emitted from a stack or vent influences the dispersion and transport of pollutants. A plume that is warmer than the surrounding air will generally rise, tending to increase the distance over which pollutants will be transported. The combination of temperature and vertical velocity of stack emissions combine to affect the height to which the plume will rise and the layer of the atmosphere in which it will initially be transported. As with concentration, the temperature of the plume affects the physical form of pollutants, with less volatile pollutants condensing faster from cooler plumes.

Height

Pollutants may be released into the atmosphere at different heights, and the height of release can strongly affect dispersion and transport. Greater release heights generally result in increased pollutant dilution in the atmosphere, lower ground-level concentrations, and a greater distance to peak ground-level concentrations. Release height is important in evaluating local effects on air transport, such as building downwash. While power generation or industrial activities may release combustion products from stacks that are hundreds of feet tall, volatilization releases or suspension of particulate pollutants often occur at or near the ground surface. (See figure 75.)



Source: EPA, Air Toxic Risk Assessment Reference Library

Figure 74. Source characteristics affecting dispersion

Timing and Duration

Multiplying the release duration by the release rate products the total mass of pollutant released. The timing of release relative to specific meteorological conditions determines the particular dispersion and transport of pollutants. Unfortunately, only total annual or average daily release data are available for most sources, making it difficult to fully characterize time varying releases. Fortunately, chronic exposure assessments usually focus on the average long-term (annual) concentrations. Acute exposure assessments, however, usually focus on the maximum short-term (twenty-four hours or less) concentration. Acute exposures are derived from conservative meteorological factors that lead to the highest short term peak values for a screening exercise; for a more detailed exercise, the actual meteorology should determine the short term peaks.

Physical Form

The physical form of pollutant releases greatly affects the dispersion, transport, and chemical reactions that pollutants undergo. Generally, pollutants are characterized as being vapors (not bound to particles, but existing as single molecules or very small aggregates "dissolved" in air—called gaseous), particle-bound (reversibly absorbed or condensed onto the surface of particles), or particulate (irreversibly incorporated into airborne particles). The distribution of pollutants in these three phases is known as partitioning. Partitioning is a function of the chemical and physical properties of the pollutants and the temperature and pressure of the atmosphere into which the chemicals have been released (e.g., the partitioning behavior of pollutants can vary greatly with temperature). As noted above, sources can emit chemical pollutants in the vapor phase at relatively high temperatures, and these pollutants can condense into or onto particulates as the emitted gases cool in the atmosphere. Sources generally emit most metals (with the important exception of mercury) as particles in the atmosphere.

Particle Size

When sources release pollutants as particles (or if released as gases, if these pollutants condense into particles or absorb onto the surface of existing particles), the rate of pollutant removal from the atmosphere to surfaces (e.g., plants, soils, surface water) depends on particle size. The typical size of particles that different activities and processes emit into the air can vary by many orders of magnitude (powers of ten). As the size of particles increases, the rate at which particles fall due to gravity (the settling velocity) increases. Thus, fine particles (approximate diameter less than a few microns) may remain suspended in air indefinitely, but particles larger than about 20 microns in diameter settle rapidly and may not transport far from sources of release.

Particles less than 10 microns in diameter are of primary concern because they are small enough to be taken into and deposited in the lung after inhalation. These particles are divided into two size ranges: fine particles less than 2.5 micrometers in size ($PM_{2.5}$), and coarse particles covering the range from 2.5 to 10 micrometers in diameter (PM_{10}). (Thus, monitoring analyses for metals in particulates, for example, would commonly collect particulate samples of $PM_{2.5}$ and PM_{10} for analysis, not total suspended particulate, because such samples contain particles that are too large to be effectively respired—thereby leading assessors to overestimate inhalation risk.) Particles emitted from combustion and high-temperature chemical processes can be very fine, on the order of 0.01 to 1.0 microns in

diameter. Such fine particles tend to condense to form larger aggregates up to the limit of perhaps a few microns, and participate in a wide range of chemical reactions.

Chemical Form

Chemical form is generally more of a concern for inorganic pollutants, because organic chemicals tend to have well-defined chemical compositions and properties. The most important chemical properties of inorganic metal compounds, for example, include the oxidation or valence state of the cationic metal, the identity of the anionic counterion, and the chemical and physical properties of the compound that the cation and anion comprise. As an example of the importance of valence state, consider the metal chromium. When emitted in the hexavalent form (with six positive charges – Cr^{+6}), chromium is highly reactive chemically and is readily reduced under certain conditions to the trivalent form that is Cr^{3+}. Cr^{6+} can cause respiratory irritation and cancer in humans. Cr^{3+}, on the other hand, is much more stable, is much less toxic to humans and animals (and is actually an essential mineral), and is not thought to cause cancer.

The following describes some of the most important physical and meteorological factors that affect the movement of air pollutants after their release. Although this material focuses on releases from stationary point sources (i.e., stacks), most of the factors may apply to releases from other source types as well. Stacks come in all sizes, from a small vent on a roof to stacks hundreds of feet in height. The function of a stack is to remove pollution of high concentrations and to discharge it to the atmosphere for dispersion and transport. Stacks release pollutants high enough above the earth's surface that pollutants can sufficiently disperse in the atmosphere before reaching ground level. All else being equal, taller stacks disperse pollutants more effectively than shorter stacks because the pollutants release into higher wind speeds and travel through a greater depth of the atmosphere before reaching ground level.

The air space the stack pollutant occupies can be described as a plume. As the plume travels, it spreads and disperses, reducing ambient pollutant concentrations even though the cross-sectional mass of the plume remains the same. Eventually, the plume may intercept the ground. The combination of emission velocity, emission temperature, vertical air movement, and horizontal airflow all influence how high a plume will rise and how fast and far it travels. Another factor is wind direction (i.e., changes in wind direction during light wind speed conditions) that can cause the plume to deviate in the horizontal direction due to turbulence and wind fluctuation. (See figure 76.)

Wind Speed and Direction Affect Plume Dispersion

The wind will determine which direction the plume goes and how fast it gets there. To look at long-term impacts (chronic exposure) a wind-rose (a distribution of winds around a compass) can be used to determine the areas of persistent wind; downwind of the largest persistent winds will generally be the areas to expect the maximum long-term impacts.

Source: EPA, Air Toxic Risk Assessment Reference Library

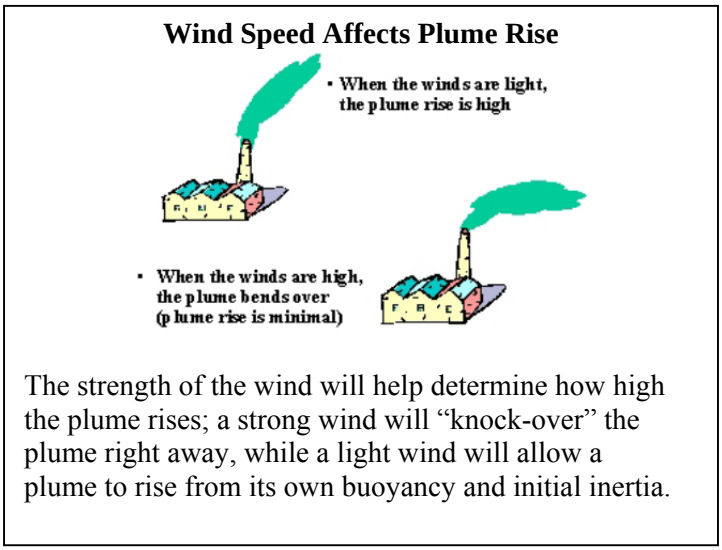
Figure 76. Wind speed and direction affect plume dispersion

As the gases exit the stack, they mix with ambient air. This mixing of ambient air into the plume is called entrainment. The plume grows in volume as it entrains ambient air and travels downwind. Because stack gases are often warmer than the ambient outdoor air, the gases may be less dense than the outdoor air and are therefore buoyant (like a helium-filled balloon). Gases that stacks emit are often pushed out by fans giving the gas momentum as it enters the atmosphere. The combination of this momentum and the buoyancy of the stack gases that are warmer than the ambient air cause the gas to rise.

This plume rise allows air toxics emitted in this stack stream to be lofted higher in the atmosphere. Since the plume is higher in the atmosphere where the winds are generally stronger, the plume will generally disperse more before it reaches ground level. Plume rise depends on the stack's physical characteristics and on the effluent's exit temperature and velocity.

The condition of the atmosphere (i.e., the vertical profile of the winds and temperature) along the path of the plume determines how far the plume rises in the atmosphere. An inversion layer (formed when a layer of warm air traps a layer of cold air beneath) may act as a barrier to vertical mixing. The height of a stack in relation to the height of the inversion layer may often influence ground-level pollutant concentrations.

The initial velocity of the plume (stack exit velocity) reduces quickly as the plume entrains ambient air and acquires horizontal momentum from the wind. This momentum causes the plume to bend over. The greater the wind speed, the more horizontal momentum the plume acquires. Wind speed usually increases with height above the earth's surface. Therefore, as the plume continues upward the stronger winds tilt the plume even further. This process continues until the plume may appear to be horizontal to the ground. The point where the plume looks level may be a considerable distance downwind from the stack. This is illustrated in figure 77.



Source: EPA, Air Toxic Risk Assessment Reference Library

Figure 75. Wind speed affects plume rise

Due to configuration of the stack or adjacent buildings, the plume may not rise freely into the atmosphere. The way that the wind moves around adjacent buildings and the stack can force the plume toward the ground instead of allowing it to rise in the atmosphere. Stack-tip downwash can occur where the ratio of the stack exit velocity to horizontal wind speed is small. In this case, low pressure in the wake of the stack may cause the plume to draw downward behind the stack. Pollutant plume rise reduces when this occurs and elevates pollutant concentrations immediately downwind of the source. As air moves over and around buildings and other structures, it forms turbulent wakes. Depending on the stack height, it may be possible for the plume to be pulled down into this wake area. The reduction in plume height is known as aerodynamic or building downwash.

Once air toxics have equilibrated with ambient conditions (e.g., temperature, velocity), atmospheric and meteorologic factors primarily influence dispersion and transport of air toxics. In particular, the rate of dispersion is influenced by the thermal structure of the atmosphere and mechanical agitation of the air as it moves over the different surface features of the earth.

Video 67. Spectacular smoke plume—borex dispersion experiment
http://www.bing.com/videos/search?q=Plume+Dispersion&Form=VQFRVP#view=detail&mid=122FAC3D4801802F197C122FAC3D4801802F197C

j. Identify and discuss the treatment of uncertainty and the realistic effects in accident analyses.

The following is taken from DOE-HDBK-3010-94.

The ARFs are based primarily on experimentally measured values for the specific material (e.g., plutonium, uranium, mixed fission products) or surrogates subjected to the particular type of stress under controlled conditions. Attention is given to the parameters, if known, that may have a significant influence on suspension by the specific mechanism and the uncertainty in the measurement as indicated by the variability of the results. Those applying the data must be aware of the range of stress represented by the measured ARFs, and seek to define the accident conditions to determine, in a gross sense, whether or not the stresses induced by the postulated events are bounded by the experimental parameters.

No attempt is made to precisely characterize total airborne material in terms of individual mechanisms acting within an overall given release. To obtain useful data outside the immediate physical chaos of the stress-inducing event itself, the experimental apparatus cited almost uniformly relied on designs to channel air to some contamination collection mechanism a short distance from the point of generation. The need to keep this distance small to avoid introducing new distortion in the form of aerosol deposition effects is one of the reasons for the relatively small scale of many experiments.

In keeping with this experimental design, the interpretation of experimental data does not consider material momentarily airborne from substrate mass ejection due to physical stresses acting on the substrate mass as an aerosol suspended in air. For example, in fire and boiling experiments, fuel mass and volumes of solution were observed to splatter or launch from the experimental substrate and land on surfaces in the local vicinity. The radioactive contamination carried with this material that deposits and is measured on the adjacent surfaces is not an aerosol suspended in air, and does not travel on air currents. It represents a source of highly localized migration that is not amenable to meaningful prediction and is not relevant to the issue of how much material might be expected to escape the immediate area and disperse in air.

Measured experimental data for RFs are much more limited but are from the same general sources used for the ARFs. To keep RFs at a reasonable bounding rather than an ultraconservative level, the RF associated with the measured bounding ARFs is generally selected rather than the highest RF value measured. The highest RF values are often associated with the smallest ARFs, and when used in conjunction with the bounding ARF, result in ultraconservative estimates of the RF released. When no measured RF is associated with the maximum measured ARF, but other measured RFs are available for the experimental set, the greatest RFs are generally used. In some cases where significant uncertainty may exist, RFs are arbitrarily set to a value of 1.0 for conservatism.

There appear to be very large differences in suspension rates under experimental test conditions as well as an order of magnitude uncertainty in measurements for individual conditions. On this basis, conservative values are applied.

k. Identify the purpose and relationship between chapters 3, 4, and 5 and the TSRs of the DSA.

The following is taken from DOE-STD-3009-94, Chg 3.

Section 3.4.2X.5 identifies the SC SSCs (or equivalent SAC) and assumptions judged to require TSR coverage. Any TSR assumption not directly related to exceeding of the EG should be defined in section 3.3.2.3.2, "Defense-in-Depth."

Sections 4.3X5 and 4.5.X.5 identify those assumptions requiring TSRs to ensure performance of the safety function. SACs are implemented in TSRs generally by either of the following two forms:

- LCO/SR—SACs can often be written in the format of an LCO.
- Specific "directive action" AC—A specific directive action AC TSR can be in the AC section of the TSRs.

DOE-STD-1186-2004, section 4, discusses the treatment of SACs in TSRs.

DOE-STD-3009-94, Chg 3, chapter 5, "Derivation of Technical Safety Requirements," builds on the control functions determined to be essential in chapter 3, "Hazard and Accident Analyses," and chapter 4, "Safety Structures, Systems, and Components," to derive TSRs. This chapter is meant to support and provide the information necessary for the separate TSR document required by 10 CFR 830.205. Derivation of TSRs consists of summaries and references to pertinent sections of the DSA where design (i.e., SSCs) and administrative features (i.e., non-SSCs) are needed to prevent or mitigate the consequences of accidents. Design and administrative features addressed include ones that 1) provide significant DID; 2) provide for significant worker safety; or 3) provide for the protection of the public. Expected products of this chapter, as applicable based on the graded approach, include the following:

- Information with sufficient basis from which to derive, as appropriate, any of the following TSR parameters for individual TSRs:
  - SLs
  - LCSs
  - LCOs
  - SRs
- Information with sufficient basis from which to derive TSR ACs for SACs or to specify programs necessary to perform institutional safety functions
- Identification of passive design features addressed in the DSA
- Identification of TSRs from other facilities that affect the facility's safety basis

l. Complete a review of a hazards analysis including walking down the scope of the work area.

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

<u>Mandatory Performance Activities</u>:

a. Participate on a combination of at least two safety basis documents or amendment reviews and/or review of safety basis documents whose scope includes evaluation of the development of a source term or radiological dose consequences.

Mandatory performance activities are performance-based KSA. The Qualifying Official will evaluate its completion.

27. Nuclear safety specialists must demonstrate a working level knowledge of DOE-STD-1186-2004, *Specific Administrative Controls*, with respect to its impact on the Department's nuclear safety.

a. Describe the relationship of DOE-STD-1186-2004 to 10 CFR 830, DOE G 423.1-1, and DOE G 421.1-2.

[Note: DOE G 423.1-1 has been superseded by DOE G 423.1-1A and DOE G 421.1-2 has been superseded by DOE G 421.1-2A.]

The following is taken from DOE-STD-1186-2004.

10 CFR 830, subpart B, "Safety Basis Requirements," requires contractors responsible for hazard category 1, 2, and 3 nuclear facilities to develop safety bases for those facilities. The safety bases consist of DSAs and hazard controls in TSRs derived from the DSA hazard analyses. Various guides and technical standards, such as DOE-STD-1186-2004, provide guidance to help interpret and implement requirements, including the DSA safe harbor methodologies listed in 10 CFR 830, appendix A, table 2. (See table 11.)

The methodology in DOE-STD-3009-94 is an acceptable method for preparation of a DSA for nonreactor nuclear facilities. DOE-STD-3009-94 provides detailed guidance for preparation of SARs including the derivation of TSRs. The general guidance of DOE-STD-3009-94 in methodologies for HA and specification of hazard controls and their classification is applicable to all safe harbor methodologies as is the guidance for SACs in DOE-STD-1186-2004.

Dispositioning activities such as decommissioning and environmental restoration provide unique challenges. In these types of activities, it is common that the hazards and the hazard control sets change as the work progresses. More application-specific guidance for dispositioning activities can be found in DOE-STD-1120-2002.

DOE-STD-3009-94, Chg 3, addresses derivation of ACs relative to the anticipated application of ACs with major significance to DID, or worker safety. These ACs are typically implemented through safety management programs. Inclusion of these ACs in the TSRs formally acknowledges the importance of programmatic commitments (e.g., radiation protection, maintenance, QA) to overall facility safety, but usually do not specify key aspects of each program as providing specific safety functions. The cumulative effect of these safety management programs is recognized as being important to overall facility safety, as opposed to specific accident risk reduction.

DOE G 423.1-1A provides detailed guidance for developing TSR content, including ACs. DOE G 423.1-1A, section 4.7, "Administrative Controls," recognizes that ACs may be applied for risk reduction of individual accident scenarios. When ACs specifically state a limit or specific requirements rather than a generic programmatic reliance, failure to meet such statements can result in a TSR violation. In contrast, a TSR violation of a safety management program can only result from a gross program failure, significant enough to render the DSA assumptions invalid.

DOE G 423.1-1A and DOE-STD-3009-94, Chg 3, continue to provide relevant guidance on the application of ACs as part of the DSA-required controls. However, DOE-STD-1186-2004 provides additional amplification and clarification for the appropriate development and implementation of ACs.

b. Define and discuss how SACs are identified.

The following is taken from DOE-STD-1186-2004.

To focus attention on the unique issues associated with this type of AC, DOE-STD-1186-2004 introduces a classification of AC to be known as an SAC. An SAC exists when an AC

- is identified in the DSA as a control needed to prevent or mitigate an accident scenario
- has a safety function that would be SS or SC if the function were provided by an SSC

c. Discuss the position of SACs in the preferred hierarchy of hazard controls.

The following is taken from DOE-STD-1186-2004.

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.

d. Describe how SACs are treated in DSAs and TSRs.

The following is taken from DOE-STD-1186-2004.

DSA

The provisions of 10 CFR 830.204 require a DSA to "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."

All hazard controls are identified and characterized in support of the DSA. Judgments must be made regarding what constitutes appropriate controls. These judgments should consider the level of the hazard and potential consequences, the practicality and effectiveness of possible control options, the importance of the mission of the facility, and other relevant factors, if any. These are all elements of the graded approach.

Hazard controls in the DSA are selected to reduce the risks of hazardous activities. Controls are classified by comparison to an EG in the case of SC SSCs for protection of the public, and by criteria described in DOE-STD-3009-94, Chg 3, for SS SSCs for worker protection and DID. SC and SS SSCs are expected to be addressed in TSRs.

While SACs may be acceptable for ensuring safe operation, they must be evaluated carefully when choosing safety measures for long-term hazardous activities because of their generally lower reliability compared with engineered controls. The actual design and selection process should consider the ensemble of controls used to address a hazard, such as cost, availability, required reliability, and consequence of mechanical or human failure for each potential control. SACs have elevated safety significance, and have more stringent implementation and verification requirements to ensure their effectiveness and dependability, as described in DOE-STD-1186-2004.

Controls identified as part of a safety management program (e.g., fire, criticality, radiation protection, etc.) 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. Hazard controls should be identified on a case-by-case basis and should be graded according to the guidance in DOE G 421.1-2A; DOE-STD-3009-94, Chg 3; and DOE-STD-1186-2004, with regard to the classification of hazard controls.

For site-wide safety management programs (e.g., radiation protection), 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 (e.g., hazardous material inventory limits) that are required for safety. These controls should be considered for designation as an SAC as discussed in DOE-STD-1186-2004.

TSRs

The following is taken from DOE-STD-1186-2004.

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 DID, or worker safety. Failure to maintain all aspects of one of these programs will not result in a safety basis violation unless there is a gross failure significant enough to render the DSA assumptions invalid (i.e., a programmatic breakdown). 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 SC or SS SSCs. The classification of SAC, as defined in DOE-STD-1186-2004, 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 in accordance with the guidance in DOE-STD-1186-2004.

e. Discuss how the concepts of validation and verification apply to SACs.

The following is taken from DOE-STD-1186-2004.

The concepts of V&V are important to the formulation of SACs. These concepts, as they apply to SACs, are discussed below.

Validation

The functional requirements and performance criteria for safety SSCs are identified to support the safety functions identified in the DSA and to support subsequent derivation of TSRs. The formulation of SACs should include a similar process that validates plant operators can perform the task(s) called for in an SAC within the timeframes assumed in the DSA. If SACs require operator action and perform a function similar to a safety SSC, assurance should be provided that the operators can adequately perform their required tasks by analyzing the following human performance factors at a minimum:

- Adequacy of the description of the task in facility procedures
- Level of difficulty of the task
- Design of the equipment and feedback (e.g., indicators, alarms, etc.)
- Time available to do the task or recover an error
- Stress levels induced by the external environment (e.g., noise, heat, light, and protective clothing worn)

Formal engineering calculations may be necessary to ensure plant operators have the appropriate time and resources to carry out the required tasks. For example, if it is assumed that operators will take actions to detect and isolate a leak, flow rate calculations will need to be performed to substantiate the available time interval necessary to accomplish the task. Consequences of incorrect implementation of the control should be evaluated and measures to prevent control failure should be factored into the control formulation.

If SACs require operator action and perform a function similar to an SC SSC, a human reliability assessment (HRA) should be performed as part of the SAC formulation.

The HRA validates the dependability of an SAC and can identify weaknesses in the proposed procedures to implement an SAC and suggest additional measures to improve the overall dependability.

Verification

SACs implemented by TSRs must be initially (prior to operation) and periodically verified to perform their intended safety function. In the context of SACs, this may involve “dry runs,” procedure walkdowns, tabletop exercises, or actual hazard/casualty exercises. Additionally, the verification process should be performed by knowledgeable individuals who were not part of the formulation of the control to assure an unbiased assessment of the effectiveness of the control. In addition, the control should be formulated so that it is easily and readily verifiable through appropriate and ongoing testing, examination, or surveillance activities. Periodic re-verification that SACs are performing, or capable of performing, their intended safety function should be addressed through LCO SR for SACs written as LCOs, or through

facility operations and maintenance procedures if the specific AC is incorporated in the AC section of the TSRs.

f. Discuss how SACs are implemented and maintained.

The following is taken from DOE-STD-1186-2004.

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:

1. LCO/SR—SAC TSRs can often be written in the format of an LCO.
2. Directive action SAC—A directive action SAC TSR can be in the AC section of the TSRs.

These are discussed in DOE-STD-1186-2004, section 4, "Treatment of Specific Administrative Controls in Technical Safety Requirements (TSR)."

g. Describe measures used to ensure the dependability of SACs.

The following is taken from DOE-STD-1186-2004.

The measures to ensure the dependability of SACs are discussed below.

Lessons Learned on Human Actions Used for Safety Controls in Accident Scenarios

Human actions, either taken in response to an event or taken proactively to establish desired conditions, are subject to errors of omission or commission. Experience shows that ACs are prone to CCF. Attributes used to improve worker performance in using ACs should be carefully evaluated for improving the dependability of SACs. Implementation of each of these attributes may not be practical or necessary for every SAC. DOE O 422.1, Admin Chg 1, *Conduct of Operations*, and DOE-STD-1186-2004 provide guidance for improving the dependability of SACs based on the safety function performed by the SAC.

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 O 422.1, Admin Chg 1. Proper conduct of operations is a key safety management program and should be addressed in the facility DSA as such. The guidelines in DOE O 422.1, Admin Chg 1, form a compendium of good practices and describe key elements of programs that support operations at DOE facilities. These may be applied to improve the dependability of SACs.

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 O 422.1, Admin Chg 1, for each of these program elements. The

following discussion provides specific guidance relative to improving the dependability of SACs.

Independent Verification

SACs that require operation of components, or verification of components, condition, or position should be included in the facility's independent verification program. As such, these verifications should be identified explicitly in facility procedures or other official documents.

Lockouts and Tagouts

A lockout/tagout program as described in DOE O 422.1, Admin Chg 1, should be used to support implementation of SACs where the SACs require that equipment, components, or equipment controls be placed in a specific position or condition during operations to support the safety basis. Use of this program to support an SAC further ensures that the requirements of the SAC are implemented using detailed administrative procedures, training of personnel, and uniquely identifiable tags.

Instrumentation, Controls, and Support Equipment for SACs

Operators often must rely on effective I&Cs and support equipment to implement SACs. For this reason, I&Cs and equipment that support an SAC should meet performance requirements consistent with the importance of the safety function of the SAC.

Training and Qualification for SACs

Training requirements for contractor personnel are generally stated under the QA requirements of 10 CFR 830, as discussed below. These requirements are applicable to all contractor personnel involved with nuclear facilities, including management and supervisory personnel, technical staff, and operations personnel. As a minimum, hazard analysts, personnel involved with formulation of SACs, and TSR writers should receive training on the guidance of DOE-STD-1186-2004. Training on TSRs for operations personnel should include specific training on attributes of the SACs as identified in the safety basis. Training should include training on the implementing procedures for SACs.

Establishing a Safety Culture

There are many aspects, organizationally and operationally, to establishing a safety culture in facilities involving hazardous operations. The Institute of Nuclear Power Operations (INPO) Excellence in Human Performance Initiative 2001 identified the key principles in developing an appropriate safety culture to improve human performance.

h. Discuss SAC violation reporting requirements.

The following is taken from DOE-STD-1186-2004.

Violations of SACs in the TSRs must be reported to DOE in accordance with DOE O 231.1B, Admin Chg 1.

DOE O 232.2, Admin Chg 1, *Occurrence Reporting and Processing of Operations Information*, provides guidance on how to determine the apparent cause(s) of specific reportable occurrences including TSR violations, and to explain the structure and nodes of the causal analysis tree for use in occurrence reporting and failure analysis.

Identifying the causes for SAC violations is often difficult. The identification of human error as a root or contributing cause of violations provides little information about how to prevent similar problems from recurring. Recognizing human performance problems when they occur and accurately identifying their causes are necessary first steps to developing effective corrective actions. The investigator(s) should be experts in human performance and the process of the facility involved in the violation. See NUREG/CR-6751, *The Human Performance Evaluation Process: A Resource for Reviewing the Identification and Resolution of Human Performance Problems*.

TSR violations, including SAC violations, that may occur during operation of the facility, must be investigated to determine their specific or generic cause(s) and generic implications, corrective actions recommended, and those violations reported to the DOE as required by 10 CFR 830.205 and DOE O 231.1B, Admin Chg 1.

MANAGEMENT, ASSESSMENT, AND OVERSIGHT

28. Nuclear safety specialists 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 nuclear security specialists in the assessment of government-owned, contractor-operated facilities.

The following is taken from DOE-STD-1183-2007.

Roles of the NSSs:

- Oversee implementation of nuclear safety requirements and programs including:
  - Participate in the oversight of contractor implementation of the Nuclear Safety Management Rule (10 CFR 830, subpart B) (e.g., preparation, review and/or recommendation for approval of nuclear basis safety documents: DSAs, TSRs, USQ, SERs, etc.).
  - Evaluate implementation of ISM as related to safety bases: identification, surveillance, and maintenance of safety SSCs, safety-related QA, selection and implementation of safety-related standards, related nuclear safety management programs, etc.
  - Evaluate the nuclear safety posture of nuclear facilities and operations for PAAA compliance, contract performance, operational readiness reviews, readiness assessments, and other periodic assessments. Participate in enforcement of PAAA requirements.
  - Participate in nuclear facility design reviews, safety system status monitoring, etc.
  - Evaluate design and analysis uncertainties with the functionalities of systems as described in the DSA.
- Communicate nuclear safety issues through the site/office management to Department and contractor management and other stakeholders and assist in the resolution of these issues.

- Participate in the development, review, approval, and interpretation of nuclear safety rules, Orders, policies, standards, guides, and documents.
- Participate in departmental self-assessments in the area of nuclear safety.
- Participate in nuclear facility accident/incident investigations.
- Participate in emergency response activities.
- Maintain and increase professional knowledge and expertise related to the field of nuclear safety.
- With respect to independent assessment, especially performance-based, the NSS is an SME for the safety basis. This is an area (and areas related to 10 CFR 830 implementation) where the NSS contributes most effectively to an assessment program.

b. Describe the assessment requirements and limitations associated with the interface with contractor employees.

The following is taken from NNSA/Service Center, Environment, Safety and Health Department, *Assessment Handbook*, June 2006.

The assessor must be independent of the particular area being assessed with respect to the contractor. The assessor must not have a vested interest in the outcome of an assessment with respect to his/her area of the assessment.

The assessor has no line management oversight over the contractor and particular assessment area and is not in any way a supervisor over those contractors with whom he/she interfaces during the course of the assessment. If a contractor fails to give the assessor information or documents requested, the issue should be turned over to the team leader who will then interface with appropriate DOE/NNSA line management and their contractor counterparts.

The assessment results are a government work product. Aside from the contractor response during factual accuracy processing, the results are not subject to negotiation by the contractor.

c. Discuss the essential elements of a performance-based assessment including:
- **Investigation**
- **Fact finding**
- **Exit interview**
- **Reporting**
- **Follow-up**
- **Closure**

Investigation

The following is taken from DOE G 414.1-1B (Archived).

An investigation is an examination or inquiry into something, especially a detailed one that is undertaken officially, or the act of undertaking an examination. Its purpose is to collect facts that, when taken as a coherent whole, enable the assessor to come to a conclusion.

The following is taken from DOE G 414.1-1C.

Performance-based assessments focus first on the adequacy of management behaviors and the processes that produce a product or service, and then on the product itself. The assessor is expected to determine whether a noncompliance or series of noncompliances with procedures could result in a failure to satisfy top-level requirements.

In performance-based assessments, greater emphasis is placed on the impact of issues discovered rather than on simply the existence of noncompliance issues. The assessor addresses the impact of the issues against the adequacy of what is being assessed. While the assessor should be familiar with requirements and procedures, in performance-based assessments the assessor's experience and knowledge play an integral part in determining whether requirements are satisfied.

Fact Finding

The following is taken from DOE G 414.1-1C.

To maximize the effectiveness of the assessment, productivity of the assessment team, and resources assessment teams should use a combination of tools and techniques. Assessment techniques include the following:

- 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.
- Document reviews—Provide the objective evidence to substantiate compliance with applicable requirements.
- Observation—Is the viewing of actual work activities. This is often considered the most effective technique for determining whether performance is in accordance with requirements.
- Inspections—Are performed in accordance with acceptance criteria to verify the condition of physical facilities, systems, equipment, and components.
- Performance testing—Is used to observe the response of personnel or equipment by creating a specific situation and noting the resulting performance.

Exit Interview

The following is taken from DOE G 414.1-1C.

The exit 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. Prior to the exit meeting, the assessment team should consider combining related weaknesses or performance issues into a smaller number of well-supported findings to help focus management's follow-up actions. A written summary of the assessment conclusions and results should be provided at the exit meeting.

Reporting
The following is taken from DOE G 414.1-1C.

Assessment reports are required for documentation of assessment results. Assessment team leaders have the overall responsibility for preparing the report and obtaining appropriate approval for its release as applicable. The report may be formal or informal, depending on the level of assessment performed, but should provide a clear picture of the results in terms of the programs, systems, and processes assessed. 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
The following is taken from DOE G 414.1-1C.

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.

Closure
The following is taken from DOE O 425.1D, Admin Chg 1.

Closure of DOE operational readiness review findings must include the following:

- Development of corrective action plans approved by DOE to correct the findings. Action plans must provide evaluation of, and address, any overall programmatic deficiencies and causes.
- Creation of a finding closure package that must include a brief description of actual corrective actions taken, evidence of completion, and reasons for concluding that closure has been achieved.
- DOE verification of closure of prestart findings. The organization verifying the closure must be designated by the startup authorization authority.

d. Describe the following assessment methods and the advantages or limitations of each method:

- **Document review**
- **Observation**
- **Interview**

The following is taken from DOE G 414.1-1C.

Document Review
Document reviews provide the objective evidence to substantiate compliance with applicable requirements. A drawback is that the accuracy of many 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 other documents should be selected carefully to ensure that they adequately characterize the program, system, or process being assessed.

Observation

Observation is the viewing of actual work activities. This 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 that should then be put in 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 observed.

e. 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 NNSA/Service Center Environment, Safety, and Health Department, *Assessment Handbook*, June 2006.

All reports of assessment results must be reviewed for factual accuracy. The contractor being assessed must concur that the reported results are factual, or detailed objective evidence must be assembled to provide indisputable evidence that reported information is factual. Factual accuracy review can be performed in parallel with assessment performance, or a period of time following completion of the data-gathering portion of the assessment may be allotted to the contractor to provide formal comments. Comments should be restricted to those of a technical nature. The contractor being assessed must provide documented objective evidence to address discrepancies documented within the report. Inclusion of a disputed non-discrepancy within the report is solely at the discretion of the team leader. It is very important that the review plan identify the process to be used for factual accuracy verification, and that the process is clearly understood by the contractor and site office. Failure to adequately verify factual accuracy of information contained within the final report often results in corrective action weaknesses, and greatly reduces the effectiveness of any type of assessment.

Since the goal of any assessment is to identify and implement actions to improve the assessed activity, the contractor being assessed must fully understand any identified problems, and must agree that the problem identification was based on valid information. It is not necessary that the assessed contractor agree that the problem identified is valid, since requiring this agreement can result in contentious relations between the contractor and the team.

29. Nuclear safety specialists must demonstrate a working level knowledge of the DOE facility contract provisions necessary to provide oversight of a contractor's operations.

a. Describe the role of nuclear safety specialists in contractor oversight.

The following is taken from 10 CFR 830 and DOE-STD-1183-2007.

In general, an NSS, as a representative of a DOE/NNSA site office's line management, is to ensure that

- government nuclear facilities are operated safely and efficiently
- the M&O system is effectively controlling its conduct of operations (within the NSS's area of expertise)
- effective lines of communication between the DOE/NNSA site office and the M&O are maintained regarding operational matters within the NSS's area of expertise

The NSS role is to ensure that the site contractor has developed an up-to-date, comprehensive, compliant, documented nuclear facility safety basis and associated implementing mechanisms and procedures for all required nuclear facilities and activities pursuant to 10 CFR 830, subpart B.

The NSS should determine whether or not the following elements have been incorporated in the contractor's safety basis:

- Requirements for developing and implementing a nuclear facility safety basis are included in contract documents and include all applicable requirements in 10 CFR 830, subpart B.
- Identification and categorization of all nuclear facilities and activities (category 1, 2, 3, and below hazard category 3) per 10 CFR 830, subpart B, using the current version of DOE-STD-1027-92, Chg 1.
- Preparation of an authorization agreement that identifies all nuclear facilities/activities and the approved safety basis for these facilities/activities and its submittal to the appropriate DOE/NNSA site office.
- Preparation and submission for approval of a DSA for each nuclear facility/activity that complies with the requirements of 10 CFR 830, subpart B.
- Preparation and submission for approval of TSRs for each nuclear facility/activity that complies with the requirements of 10 CFR 830, subpart B.
- Submission of a compliant USQ process that meets the requirements of 10 CFR 830, subpart B.
- The contractor has a process in place to develop a DSA for each nuclear facility/activity that complies with the requirements of 10 CFR 830, subpart B, and DOE/NNSA-directed standards.
- A process in place to ensure safety basis documentation is updated in accordance with the requirements of 10 CFR 830, subpart B and NNSA-directives and standards.

b. Compare and contrast the following:
- **DOE's expectations of a contractor**
- **A contractor's expectations of the DOE**

The following is taken from 10 CFR 830 and DOE-STD-1183-2007.

DOE's Expectations of a Contractor

The M&O contractor's statement of work and supporting documentation (e.g., work smart standards, etc.) will convey the mutual agreement between DOE and the M&O.

DOE expectations are communicated by means of a performance evaluation plan (PEP) or similar document. PEPs should delineate clearly and communicate effectively appropriate milestones and performance objectives for upcoming rating periods. The award-fee criteria for each contract must reflect DOE's objectives, priorities, and focused initiatives. The criteria must be meaningful and must contain challenging objectives that are prioritized and weighted in importance as appropriate. Goals and objectives that result from the corrective action plans of various assessments, investigations, etc., must be incorporated in the award-fee criteria.

A Contractor's Expectations of the DOE

The M&O contractor's statement of work and supporting documentation (e.g., work smart standards, etc.) will convey the mutual agreement between DOE and the M&O.

c. Identify the key elements and features of an effective DOE and contractor relationship.

The following is taken from DOE Order 5480.19, Chg 2 (Archived).

An effective relationship between DOE/NNSA and the M&O contractor should be based on the site's conduct of operations. An effective DOE/NNSA and M&O contractor relationship should incorporate the following key elements:
- High operating standards established in writing by M&O management through effective implementation and control of operations activities
- Communication of those operating standards to the working levels
- Sufficient resources provided to operations
- Guarantee that personnel at all levels are well-trained according to the requirements of their duties
- Close monitoring of performance in operations (i.e., performance assessments)
- Accountability of workers and supervisors for performance in conducting duties and responsibilities

d. Describe the responsibility nuclear safety specialists have associated with contractor compliance under the Price-Anderson Amendments Act.

The following is taken from 10 CFR 820 and DOE-STD-1183-2007.

The PAAA of 1988 added section 234A to the AEA to give DOE the authority to assess civil penalties for violations of rules, regulations, or Orders that related to nuclear safety by contractors and subcontractors who are indemnified by DOE pursuant to the AEA, section 170.

Any risks that are indemnified are not considered as risk to the contractor. The net result of the PAAA is to provide for greater accountability of contractors, subcontractors, and suppliers in the performance of their duties under contract with DOE for indemnified nuclear activities. The availability and careful exercise of this authority by DOE management and its SME professionals can reduce the likelihood of serious nuclear incidents. For the contractor, there are opportunities as well to identify improvements to existing operations and facilities, and modifications in practices that can improve the health and safety of workers, the public, and the environment.

To achieve the above, the NSS should

- provide oversight and direction to the M&O contractor for the development and approval of the safety basis for nuclear facilities under the cognizance of the contractor;
- monitor M&O contractor conformance with required regulations, DOE Orders, standards, etc., pursuant to agreements in the DOE-M&O contract;
- identify potential PAAA noncompliance actions and activities to one's supervisor and/or the PAAA coordinator; and
- assist in determinations of the M&O contractor's liability and penalty (civil and/or criminal) for a PAAA-relevant violation.

e. Describe the role of nuclear safety specialists in the cost-plus-award fee process.

The following is taken from 48 CFR 16.405-2.

A cost-plus-award-fee contract is a cost-reimbursement contract that provides for a fee consisting of 1) a base amount fixed at inception of the contract, if applicable, and at the discretion of the contracting officer; and 2) an award amount that the contractor may earn in whole or in part during performance and that is sufficient to provide motivation for excellence in the areas of cost, schedule, and technical performance.

The following is taken from 48 CFR 16.401.

A cost-plus-award fee contract is suitable for use when

- the work to be performed is such that it is neither feasible nor effective to devise predetermined, objective incentive targets applicable to cost, technical performance, or schedules;
- the likelihood of meeting acquisition objectives will be enhanced by using a contract that effectively motivates the contractor toward exceptional performance and

provides the government with the flexibility to evaluate actual performance and the conditions under which it was achieved; and
- any additional administrative effort and cost required to monitor and evaluate performance are justified by the expected benefits.

The amount of award fee earned shall be commensurate with the contractor's overall cost, schedule, and technical performance as measured against contract requirements in accordance with the criteria stated in the award-fee plan. Award fee shall not be earned if the contractor's overall cost, schedule, and technical performance is below satisfactory. The basis for all award-fee determinations shall be documented in the contract file to include, at a minimum, a determination that overall cost, schedule, and technical performance is or is not at a satisfactory level. This determination and the methodology for determining the award fee are unilateral decisions made solely at the discretion of the government.

The following is taken from DOE-STD-1183-2007.

The NSS should assist the contracting officer and the contracting officer's representative in evaluating the contractor's safety programs—vis-à-vis nuclear safety—for the cost-plus-award-fee process. This role should help ensure the following:
- Competitive sources are solicited, evaluated, and selected.
- Quality standards are prescribed and met.
- Performance and delivery are timely.
- Prices, estimated costs, and fees are reasonable.
- Files are documented to substantiate the judgments, decisions, and actions taken.

f. Explain the responsibilities of nuclear safety specialists for DOE O 442.1A, *Department of Energy Employee Concerns Program*, and the identification, reporting, reviewing, and documentation of employee concerns.

The following is taken from NNSA/Service Center, Environment, Safety and Health Department, *Assessment Handbook*, June 2006.

The NSS should
- support the effective implementation of the DOE Employee Concerns Program;
- assist line management in the resolution of employee concerns in a manner that protects the health and safety of employees and the public and ensures the effective and efficient operation of DOE/NNSA-related activities under the NSS's cognizance;
- report concerns related to environment, safety, and health (ES&H) to one's supervisor and/or the appropriate NNSA site office manager or official; and
- assist in assessments used to verify that corrective actions as applicable and pursuant to an employee's concerns minimize, correct, or prevent the recurrence of the situation that precipitated an employee's valid concern.

g. Describe the differing professional opinions process used in your office.

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

30. Nuclear safety specialists must demonstrate a familiarity level knowledge of problem analysis principles and the techniques necessary to identify problems, potential causes, and corrective action(s) associated with nuclear safety issues at DOE nuclear facilities.

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**

Root Cause Analysis

The following is taken from DOE-HDBK-1208-2012, volume I.

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 (ECFA), barrier analysis, and change analysis—provide answers to an investigator's questions regarding what, when, where, who, and how. Root cause analysis is primarily performed to resolve the question, "Why?"

Causal Factor Analysis

The following is taken from SCIE-DOE-01-TRAC-14-95, *Events and Causal Factors Analysis*.

ECFA is an important component in the accident investigation repertoire of methods. ECFA serves three main purposes in investigations: 1) assists the verification of causal chains and event sequences; 2) provides a structure for integrating investigation findings; and 3) assists communication during and on completion of the investigation.

The following is taken from DOE-HDBK-1208-2012.

A causal factor is an event or condition in the accident sequence that contributes to the unwanted result. ECFA includes charting that depicts the logical sequence of events and conditions (causal factors that allowed the accident to occur), and the use of deductive reasoning to determine the events or conditions that contributed to the accident.

Change Analysis

The following is taken from DOE-HDBK-1208-2012.

Change analysis is an analytical technique used for accident investigations, wherein accident-free reference bases are established, and changes relevant to accident causes and situations are systematically identified. In change analysis, all changes are considered, including those initially considered trivial or obscure.

Change analysis is performed to determine points where changes are needed to correct deficiencies in the safety management system and to pinpoint changes and differences that may have an effect on the accident.

Barrier Analysis

The following is taken from DOE-HDBK-1208-2012.

Barrier analysis is an analytical technique used to identify energy sources and the failed or deficient barriers and controls that contributed to an accident. Barrier analysis is based on the premise that hazards are associated with all accidents. Barriers are developed and integrated in a system or work process to protect personnel and equipment from hazards. Investigators use barrier analysis to identify hazards associated with an accident and the barriers that should/could have prevented it.

Management Oversight Risk Tree Analysis

The following is taken from Risk Management LTD, *MORT and All That: A 40-Year Old Technique Still of Value*.

The management oversight and risk tree (MORT) analysis is a generic analytical procedure used for inquiring into causes and contributing factors of events and to analyze risks. The MORT method is a logical expression of the functions needed by an organization to manage risks effectively. These functions are set out in a generic chart that is essentially a generic FTA used to help analyze the causes of undesired events. It can be used to analyze identified points of uncertainty in a system. MORT reflects a philosophy that holds that the most effective way of managing risk is to integrate business management and operational risk controls.

b. Describe the following types of investigations and discuss an example of the application of each:
- **Type A**
- **Type B**

[Note: Investigation types A and B are no longer used. They have been replaced by DOE O 225.1B, appendix A, "Accident Investigation Criteria," which is reproduced below in its entirety.]

The following is taken from DOE O 225.1B.

The following criteria must be considered to determine whether any accident resulting from DOE, contractor, or subcontractor operations requires the appointment of an accident investigation board (AIB). Accidents must be analyzed expeditiously to determine whether an AIB must be appointed based on the criteria indicated below, and other relevant factors.

Determination Criteria

Human Effects

- Any injury or chemical or biological exposure that results in, or is likely to result in, the fatality of an employee or member of the public. Fatal injury is defined as any injury that results in death within thirty calendar days of the accident.
- Any single accident that results in the hospitalization for more than five calendar days, commencing within seven calendar days of the accident, of one or more DOE, contractor, or subcontractor employees or members of the public due to a serious personal injury or acute chemical or biological exposure. Serious personal injury means any injury that 1) results in a fracture of any bone (except simple fracture of fingers, toes, or nose); 2) causes severe hemorrhages or nerve, muscle, or tendon damage; 3) involves any internal organ; or (4) involves second or third degree burns or any burns affecting more than five percent of the body surface.
- Any single accident resulting in three or more DOE, contractor, or subcontractor employees having lost-workday cases.
- Accidents involving Federal or contractor employees driving vehicles while on official government business, on or off government property, if the consequences result in meeting any of the criteria above.

Loss of Control of Radioactive Material

Any single accident that results in the following:

- A general employee exceeds any of the external dose limits in 10 CFR 835.202 by a factor of two or more.
- The embryo/fetus of a declared pregnant worker, minor, or a member of the public in a controlled area exceeds an external dose of one rem effective dose.
- Any confirmed monitoring result (workplace or individual) indicates an intake (via inhalation, ingestion, wound, or absorption) of radioactive material by a general employee equivalent to two or more times the ALI.
- Any confirmed monitoring result (workplace or individual) indicating an intake of radioactive material to a declared pregnant worker, a minor, or a member of the public in a controlled area equivalent to twenty percent or more of an ALI.
- Notes:
  - Dose thresholds for the embryo/fetus apply after declaration of pregnancy.
  - Confirmation must be made within three working days following identification of monitoring results (workplace or individual monitoring) indicating an exposure exceeding one or more of the criteria in DOE O 225.1B, appendix A.
  - Monitoring results are those obtained prior to medical intervention to reduce or otherwise mitigate dose.
  - ALIs for an inhalation are 10 CFR 835, appendix A, values, in microcuries per milliliter (uCi/ml), multiplied by 2.4E9, ALIs for ingestion are ingestion dose coefficients from ICRP Publication 68; ALIs for wounds should use dose coefficients published in a consensus or refereed report.
  - Planned special exposures (10 CFR 835.204) or authorized emergency exposures (10 CFR 835.1301, “General Provisions”) to general employees are excluded.
  - “Confirmed” means a monitoring result confirmed by follow-up radiobioassay, by association with a known incident, or by investigation.

Environmental Release of Hazardous Material

- An accident that results in the environmental release of a hazardous material from a DOE facility (onsite or offsite), in an amount greater than five times the reportable quantities specified in 40 CFR 302.
- An accident that results in the release of a hazardous material from a DOE facility (onsite or offsite) that meets the criterion for classification as a site area or general emergency in DOE O 151.1C, *Comprehensive Emergency Management System*.
- Any offsite transportation incident involving hazardous materials that requires immediate notice pursuant to 40 CFR 302.
- For facilities in which 29 CFR 1910.119 is applicable, an incident that results in, or could reasonably have resulted in, a catastrophic release of a highly hazardous chemical in the workplace.

Property Effects

- Estimated loss of, or damage to, DOE property, including aircraft, equal to or greater than $2.5 million or requiring estimated costs equal to or greater than $2.5 million for cleaning, decontaminating, renovating, replacing, or rehabilitating property. DOE facility damage is estimated within 72 hours of the accident based on comparison with the facility replacement value in the facility information management system database maintained by the HQ Office of Administration, Office of Engineering and Construction Management.
- Any unplanned nuclear criticality.

Other Effects

Any accident or series of accidents for which an AIB is deemed appropriate by the secretary or deputy secretary. In such circumstances, the secretary or deputy secretary may direct the head of the HQ element or Chief Health, Safety and Security Officer, to conduct an accident investigation.

c. Compare and contrast immediate, short-term, and long-term actions taken as the result of problem identification or an occurrence.

The following is taken from DOE G 231.1-2 (Archived).

According to DOE G 231.1-2 (Archived), *Occurrence Reporting Causal Analysis Guide*, corrective actions should be identified to remedy the problem (immediate actions), prevent recurrence of the specific problem (short-term actions), and preclude occurrence of similar problems (long-term actions). The Institute of Nuclear Power Operations has developed a list of error precursors that are useful in preventing events from occurring. These error precursors are listed in DOE G 231.1-2 (Archived), attachment 5, "INPO Error Precursors (Short List) Versus Causal Analysis Tree Level C Nodes." It provides extensive information on the development, implementation, and follow-up of corrective action programs. The guidance may be considered and applied using a graded approach based on the significance, criticality, sensitivity, risk, and/or impact of each finding to the mission, safety, and security of the site, the public, and regulatory requirements. The guidance may assist managers and contractors during the course of organizational self-assessments, internal and external oversight,

investigations, regulatory actions, audits, inspections, worker safety concerns, design reviews, analyses, and other types of incidents or assessments.

d. Given a nuclear safety 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 KSA. The Qualifying Official will evaluate its completion.

e. Describe various data gathering techniques and the use of trending/history when analyzing problems.

Performance Indicators (PIs)

The following is taken from DOE-HDBK-1028-2009, Volume 1.

PIs or metrics are parameters measured to reflect the critical success factors of an organization. The purpose of these measures is to provide facility personnel with a way of knowing whether planned activities are occurring as originally intended as well as warning of developing problems.

The following is taken from DOE G 151.1-3.

PIs represent an evaluation tool for monitoring, tracking, and analyzing specific parameters that reflect the characteristics of a program's preparedness activities. They can provide a timely indication of problems developing in program readiness, if key parameters have been identified that might predict performance degradation. PIs that reflect response performance obtained directly from observations during exercises can be used to track and identify unacceptable trends associated with responder activities.

PIs are usually discussed in terms of deterministic processes that produce products or services with identifiable characteristics that meet the needs of the customer. Whether the product meets the specific requirements is either directly apparent or becomes apparent through deterministic testing. In contrast, the "products" are not necessarily objects that can be directly observed and whose characteristics are readily apparent and well-defined. Actual tests are not comprehensive, involve a snapshot of performance in response to a single event scenario, test only selected functions, and usually involve only one segment of the program. Hence, the ongoing activities that maintain the program must be evaluated to provide assurances that the program satisfies the comprehensive objectives associated with the range of hazards on the facility/site or associated with an activity and maintained at a level of readiness to protect workers, the public, and the environment.

PIs reflect critical quantitative or qualitative performance characteristics of products, services, and processes. They represent a useful tool for helping to understand, manage, and improve how organizations perform. In general, PIs can reflect the following:

- How well the organization is doing
- If it is meeting its goals
- If customers are satisfied
- If processes are in statistical control
- If and where improvements are necessary

A logical selection of PIs can provide an organization with the information necessary to make intelligent decisions related to achieving the organization's mission and goals.

Trending and Analysis

The following is taken from DOE Order 5480.26 (Archived).

Contractors for each facility, groups of facilities, or site shall review and assess their PIs and other operations information, such as reportable occurrences.

Facility managers shall assess their facility operating information for trends and indications of deteriorating/improving conditions and identify lessons learned and good practices that should be used in their facilities to prevent occurrences or to improve safety and/or operations.

Each level of DOE line management shall adopt the use of trending and analysis of PIs and other operations information, such as reportable occurrences, at all levels of operations to provide ongoing feedback to operators, support personnel, and managers of the condition and performance of their operations with the intent of identifying deficiencies/good practices and opportunities for improvement in safety and performance of their operations.

History

There is a natural tendency for people to think, "It can't happen here" or "That won't happen to me." Humans underestimate risk and overestimate their ability to maintain control. This sense of invulnerability is an unsafe attitude. The use of operating experience (using feedback acquired from previously operating equipment or a system, internal and external to the facility) has proven effective in improving performance and keeping facilities safer. Operating experience helps ground individuals to the risks and vulnerabilities associated with specific activities. Lessons learned can be reinforced during various training forums and through day-to-day activities such as pre-job briefings, coaching, and reinforcement by supervisors, as well as through engineering design reviews.

31. Nuclear safety specialists must demonstrate a familiarity level knowledge of DOE Policy 450.4, *Safety Management System Policy*; Policy 226.1A, *Department of Energy Oversight Policy*; and DOE O 226.1, *Implementation of Department of Energy Oversight Policy*, as applied to nuclear safety.

[Note: DOE Policy 450.4 has been superseded by DOE Policy 450.4A, *Integrated Safety Management Policy*; DOE Policy 226.1A has been superseded by DOE Policy 226.1B; and DOE O 226.1 has been superseded by DOE O 226.1B.

a. Discuss the fundamentals of ISM and its direct application to nuclear safety.

The following is taken from DOE P 450.4A.

It is the Department's policy that work be conducted safely and efficiently and in a manner that ensures protection of workers, the public, and the environment. To achieve this policy, effective safety requirements and goals are established; applicable national and international consensus standards are adopted; and where necessary to address unique conditions,

additional standards are developed and effectively implemented. Implementing ISM requirements for Federal organizations are established through directives, and for contractor organizations through contract clauses.

The Department will implement ISM systems to systematically integrate safety in management and work practices at all levels in the planning and execution of work. All organizations will develop, maintain, and implement ISM systems for their operations and work practices, based on ISM guiding principles and core functions. To improve effectiveness and efficiency, organizations are expected to tailor their safety management system to the hazards and risks associated with the work activities supporting the mission; including using established mechanisms to tailor requirements. Decisions impacting safety are made by technically qualified managers with knowledge of the operations and after consideration of hazards, risks, and performance history. To complement these systems and mechanisms, the Department expects all organizations to embrace a strong safety culture where safe performance of work and involvement of workers in all aspects of work performance are core values that are deeply, strongly, and consistently held by managers and workers. The Department encourages a questioning attitude by all employees and a work environment that fosters such attitude.

The guiding principles of ISM include the following:

- Line management responsibility for safety—Line management is directly responsible for the protection of the workers, the public, and the environment.
- Clear roles and responsibilities—Clear and unambiguous lines of authority and responsibility for ensuring safety are established and maintained at all organizational levels within the Department and its contractors.
- Competence commensurate with responsibilities—Personnel possess the experience, knowledge, skills, and abilities that are necessary to discharge their responsibilities.
- Balanced priorities—Resources are effectively allocated to address safety, programmatic, and operational considerations. Protecting the workers, the public, and the environment is a priority whenever activities are planned and performed.
- Identification of safety standards and requirements—Before work is performed, the associated hazards are evaluated and an agreed-upon set of safety standards and requirements is established which, if properly implemented, will provide adequate assurance that the workers, the public, and the environment are protected from adverse consequences.
- Hazard controls tailored to work being performed—Administrative and engineering controls to prevent and mitigate hazards are tailored to the work being performed and associated hazards.
- Operations authorization—The conditions and requirements to be satisfied for operations to be initiated and conducted are clearly established and agreed upon.

These five core safety management functions provide the necessary structure for any work activity that could potentially affect the workers, the public, and the environment. The functions are applied as a continuous cycle with the degree of rigor appropriate to address the type of work activity and the hazards involved.

1. Define the scope of work—Missions are translated into work, expectations are set, tasks are identified and prioritized, and resources are allocated.
2. Analyze the hazards—Hazards associated with the work are identified, analyzed, and categorized.
3. Develop and implement hazard controls—Applicable standards and requirements are identified and agreed-upon, controls to prevent/mitigate hazards are identified, the safety envelope is established, and controls are implemented.
4. Perform work within controls—Readiness is confirmed and work is performed safely.
5. Provide feedback and continuous improvement—Feedback information on the adequacy of controls is gathered; opportunities for improving the definition and planning of work are identified and implemented.

b. Describe the key elements of an effective contractor self-assessment nuclear safety program.

The following is taken from DOE P 226.1B.

To provide strong assurance that the workers, the public, the environment, and national security assets are adequately protected, the Department expects that 1) robust assurance systems are effectively implemented by site contractors and, for DOE operated activities, by the responsible DOE line management organizations; and 2) DOE oversight is performed effectively by line management, DOE HQ and field, as well as by independent oversight organizations.

Attributes of effective assurance and oversight processes include the following:

- Assurance systems are tailored to meet the needs and unique risks of each site or activity, include methods to perform rigorous self-assessments, conduct feedback and continuous improvement activities, identify and correct negative performance trends, and share lessons learned.
- DOE oversight programs are designed and conducted commensurate with the level of risk of the activities.
- The oversight of activities with potentially high consequences is given high priority and greater emphasis.

The following is taken from DOE O 226.1B.

The contractor assurance system, at a minimum, must include rigorous, risk-informed, and credible self-assessment and feedback and improvement activities. Assessment programs must be risk-informed, formally described and documented, and appropriately cover potentially high consequence activities.

The following is taken from DOE G 450.4-1C.

Self-assessments and management assessments are done to determine whether work is properly scoped and whether work controls and work performance adequately meet agreed-upon standards. The criteria, indicators, and measures used in assessments are best developed at the time the work and the ES&H expectations are agreed on. Such up-front work ensures that the criteria, indicators, and measures are developed using the same factors that were used to develop the work expectations.

Contractors should develop assessment programs and protocols that are tailored to the details of their safety management systems to meet agreed-upon expectations. DOE assessment programs and protocols should ensure that contractors' self-assessment programs are effective and produce valid results. A history of effective self-assessment and continuous performance improvement by contractors can be a basis for decreased DOE assessment efforts.

c. Discuss the following nuclear safety assessments/surveillance activities:
- **Determination of assessment/surveillance requirements**
- **Operation/area/site office and contractor notification**
- **Assessment/surveillance agenda**

Determination of Assessment/Surveillance Requirements
The following is taken from DOE O 226.1B.

Oversight processes must be implemented by applicable DOE line management organizations and must include written plans and schedules for planned assessments, focus areas for operational oversight, and reviews of the contractor's self-assessment of processes and systems. DOE line management must evaluate the contractor and DOE programs and management systems, including site assurance systems, for effectiveness of performance (including compliance with requirements). Such results must be based on the results of operational awareness activities; assessments of facilities, operations, and programs; and assessments of the contractor's assurance system. The level and/or mix of oversight may be tailored based on considerations of hazards, the maturity and operational performance of the contractor's programs, and management systems.

Operation/Area/Site Office and Contractor Notification
The following is taken from DOE O 226.1B.

DOE line management must have effective processes for communicating oversight results and other issues in a timely manner up the line management chain, and to the contractor as appropriate, sufficient to allow senior managers to make informed decisions. DOE line management must ensure relevant findings are effectively communicated to the contractors, and ensure that problems are evaluated and corrected on a timely basis.

Assessment/Surveillance Agenda

The following is taken from DOE O 226.1A (Archived).

DOE oversight programs and assurance systems will evaluate performance against requirements and performance objectives, including laws, regulations, national standards, DOE directives, DOE-approved plans and program documents, site-specific procedures/manuals, criteria review and approach documents, other contractually mandated requirements, and contractual performance objectives. Requirements and performance objectives are established and interpreted through approved processes so that they are relevant to the site and mission.

SELECTED BIBLIOGRAPHY AND SUGGESTED READING

Code of Federal Regulations

10 CFR 20.1001-20.2401, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewage." January 1, 2014.
10 CFR 20.1003, "Definitions." January 1, 2014.
10 CFR 30, "Rules of General Applicability to Domestic Licensing of Byproduct Materials." January 1, 2014.
10 CFR 30.11, "Specific Exemptions." January 1, 2014.
10 CFR 30.19, "Self-Luminous Products Containing Tritium, Krypton-85, or Promethium-147." January 1, 2014.
10 CFR 30.72, Schedule C, "Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release." January 1, 2014.
10 CFR 70.4, "Definitions." January 1, 2014.
10 CFR 71, "Packaging and Transportation of Radioactive Materials." January 1, 2014.
10 CFR 71.71, "Normal Conditions of Transport." January 1, 2014.
10 CFR 820, Appendix A, "General Statement of Enforcement Policy." January 1, 2014.
10 CFR 830, "Nuclear Safety Management." January 1, 2014.
10 CFR 830.2, "Exclusions." January 1, 2014.
10 CFR 830.3, " Definitions." January 1, 2014.
10 CFR 830, Subpart A, "Quality Assurance Requirements." January 2013.
10 CFR 830, Subpart B, "Safety Basis Requirements." January 2013.
10 CFR 830, Appendix A to Subpart B, "General Statement of Safety Basis Policy." January 1, 2014.
10 CFR 830.120, "Scope." January 1, 2014.
10 CFR 830.122, "Quality Assurance Criteria." January 1, 2014.
10 CFR 830.201, "Performance of Work." January 1, 2014.
10 CFR 830.202, "Safety Basis." January 1, 2014.
10 CFR 830.203, "Unreviewed Safety Question Process." January 1, 2014.
10 CFR 830.204, "Documented Safety Analysis." January 1, 2014.
10 CFR 830.205, "Technical Safety Requirements." January 1, 2014.
10 CFR 830.206, "Preliminary Documented Safety Analysis." January 1, 2014.
10 CFR 830.207, "DOE Approval of Safety Basis." January 1, 2014.
10 CFR 835, "Occupational Radiation Protection." January 1, 2014.
10 CFR 835, Appendix A, "Derived Air Concentrations (DAC) for Workers From External Exposure During Immersion in a Cloud of Airborne Radioactive Material." January 1, 2014.
10 CFR 835, Appendix E, "Values for Establishing Sealed Radioactive Sources Accountability and Radioactive Material Posting and Labeling Requirements." January 1, 2014.
10 CFR 835.2, "Definitions." January 1, 2014.
10 CFR 835.202, "Occupational Dose Limits for General Employees." January 1, 2014.
10 CFR 835.204, "Planned Special Exposures." January 1, 2014.
10 CFR 835.205, "Technical Safety Requirements." January 1, 2014.

10 CFR 835.207, “Occupational Dose Limits for Minors.” January 1, 2014.
10 CFR 835.401, “General Requirements.” January 1, 2014.
10 CFR 835.502, “High and Very High Radiation Areas.” January 1, 2014.
10 CFR 835.603, “Radiological Areas and Radioactive Material Areas.” January 1, 2014.
10 CFR 835.1301, “General Provisions.” January 1, 2014
10 CFR 835.1302, “Emergency Exposure Situations.” January 1, 2014.
10 CFR 835.1304, “Nuclear Accident Dosimetry.” January 1, 2014.
10 CFR 851, “Worker Safety and Health Program.” January 1, 2014.
10 CFR 851, Subpart E, “Enforcement Process.” January 1, 2014.
29 CFR 1910.111, “Storage and Handling of Anhydrous Ammonia.” July 1, 2013.
29 CFR 1910.119, “Process Safety Management of Highly Hazardous Chemicals.” July 1, 2013.
29 CFR 1910.120, “Hazardous Waste Operations and Emergency Response.” July 1, 2013.
29 CFR 1910.1200, “Hazard Communication.” July 1, 2013.
29 CFR 1910.1450, “Occupational Exposure to Hazardous Chemicals in Laboratories.” July 1, 2013.
29 CFR 1926.65, “Hazardous Waste Operations and Emergency Response.” July 1, 2013.
40 CFR 264, “Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities.” July 1, 2013.
40 CFR 265, “Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities.” July 1, 2013.
40 CFR 302, “Designation, Reportable Quantities and Notification.” July 1, 2013.
40 CFR 355, “Emergency Planning and Notification.” July 1, 2013.
40 CFR 355, Appendix A, “The List of Extremely Hazardous Substances and Their Threshold Planning Quantities.” July 1, 2013.
48 CFR 16.401, “General.” October 1, 2013.
48 CFR 16.405-2, “Cost-Plus-Award-Fee Contracts.” October 1, 2013.
49 CFR 171, “General Information, Regulations, and Definitions.” October 1, 2013.
49 CFR 173.469, “Tests for Special Form Class 7 (Radioactive) Materials. October 1, 2013.
49 CFR 970.5223-1, “Integration of Environmental Safety and Health into Work Planning and Exclusions.” October 1, 2013.

American Industrial Hygiene Association. *Emergency Response Planning Guidelines,* “ERPG Levels.”

American Institute of Chemical Engineers. *Guidelines for Hazard Evaluation Procedures*, third edition. 2008.

American National Standards Institute (ANSI)/American Nuclear Society (ANS)
ANSI/ANS-1-2000, *Conduct of Critical Experiments.* 2000.
ANSI/ANS-2.26-2004, *Categorization of Nuclear Facility Structures, Systems and Components for Seismic Design*. December 2, 2004.
ANSI/ANS-8.1, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*. 1998 reaffirmed May 16, 2007.
ANSI/ANS-8.3, *Criticality Accident Alarm System*. 1997 reaffirmed 2012.

ANSI/ANS-8.5, *Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material*. 1996 reaffirmed in 2002 and 2007.
ANSI/ANS-8.7, *Guide for Nuclear Criticality Safety in the Storage of Fissile Material*. 1998 reaffirmed in 2007.
ANSI/ANS-8.15, *Nuclear Criticality Control of Special Actinide Elements*. 1981 reaffirmed in 1987, 1995, and 2005.
ANSI/ANS-8.19, *Administrative Practices for Nuclear Criticality Safety*. 2005.
ANSI/ANS-14.1-2004, *Operation of Fast Pulse Reactors*. 2004.
ANSI N43.6, *Sealed Radioactive Sources, Categorization*. September 1, 1998.
ANSI/ASHRAE Standard 134-1005. *Graphic Symbols for Heating, Ventilating, Air-Conditioning, and Refrigerating Systems*. February 2005.
ANSI N317, *American National Standard Performance Criteria for Instrumentation Used for Inplant Plutonium Monitoring*. 1980

Ask.com. *What is the Physics of an Explosion?*

American Society of Mechanical Engineers (ASME)
ASME/NQA-1-2012, *Quality Assurance Requirements for Nuclear Facilities Applications*. 2012.
NQA-1a-2009, Addenda, *Quality Assurance Requirements for Nuclear Facility Applications*. 2009.

American Society for Testing Materials. C-1455, *Standard Test Method for Nondestructive Assay of Special Nuclear Material Holdup Using Gamma-Ray Spectroscopic Methods*. 2007

Berkeley University. Department of Nuclear Engineering. *Berkeley Radiological Air and Water Dose Calculation*.

BrightPlanIT, Inc. *Optimize Your Data Center*.

Cambridge Energy Alliance. *Whole House Ventilation Systems*.

Canadian Nuclear Safety Commission. GD-327, *Guidance for Nuclear Criticality Safety*. September 19, 2011.

Cember, Herman and Thomas E. Johnson, *Introduction to Health Physics*, fourth edition. 2009.

Cleaningpro.com. *Cleaning Products from A to Z: M*.

Controlled Environments. *Filtration Monitoring Lags Behind Advances in HVAC Technology*. November 1, 2004.

Dai, Weiping. *Applying Proper Dispersion Models for Industrial Accidental Releases*. July 26, 2004.

Defense Nuclear Facilities Safety Board (DNFSB). Technology (TECH)-20, *Protection of Collocated Workers at the Department of Energy's Defense Nuclear Facilities and Sites*. February 1999.

eHow. *How Does a HVAC Damper Work?*

Energy Consumption Characteristics of Commercial Building HVAC Systems, *Volume II: Thermal Distribution, Auxiliary Equipment, and Ventilation*. October 1999.

Environmental Protection Agency (EPA)

Acute Exposure Guidelines Levels (AEGLs), "Definitions." January 11, 2012.

Air Toxics Risk Assessment Reference Library, volume 1, *Technical Resource Manual*, chapter 8, "Qualification of Exposure: Dispersion, Transport, and Fate of Air Toxics in the Atmosphere." April 2004.

Chemical Emergency Prevention and Planning Newsletter, *Process Hazard Analysis (PHA)*. March-April 2013.

Control of Gaseous Emissions, chapter 3, "Air Pollution Control Systems." January 2000.

EPA Probabilistic Risk Assessments, chapter 31. April 2004.

Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*. September 1988.

Federal Guidance Report No. 13, *Cancer Risk Coefficients for Environmental Exposure to Radionuclides*. September 1999.

Indoor Air Quality in Large Buildings, appendix B, "HVAC Systems and Indoor Air Quality."

Manual of Protective Action Guides and Protective Actions for Nuclear Incidents. May 1992.

Technical Guidance for Hazard Analysis—Emergency Planning for Extremely Hazardous Substances. December 1987.

Technical Panel of the EPA Risk Assessment Forum, *Using Probabilistic Methods to Enhance the Role of risk Analysis in Decision-Making with Case Study Examples*.

The EPA Manual, *Protective Action Guides and Protective Actions for Nuclear Incidents*. 1991.

ESCO. *A Guide to Laboratory Fume Hoods*.

Executive Orders (EO)

EO 12699, "Seismic Safety of Federal and Federally Assisted or Regulated New Building Construction." January 5, 1990.

EO 12941, "Seismic Safety of Existing Federally Owned or Leased Buildings." December 1, 1994.

Fullwood, Ralph, *Lecture Notes for Criticality Safety*. March 1992.

GlobalSpec

HVAC Fans and Blowers Information.

Air Scrubbers, Wet Scrubbers, and Gas Scrubbers Information.

Illinois Department of Public Health, Environmental Health Fact Sheet. *Air Quality in the Home*.

INTECH. *Nanoindentation Based Analysis of Heterogeneous Structural Materials*. 2012.

Interagency Committee on Seismic Safety in Construction (ICSSC)

ICSSC RP-4, *Standards of Seismic Safety for Existing Federally Owned or Leased Buildings and Commentary*. December 1, 1994.

ICSSC RP-5, *ICSSC Guidance on Implementing Executive Order 12941 on Seismic Safety of Existing Federally Owned or Leased Buildings*. December 1995.

ICSSC RP-6, *Standards of Seismic Safety for Existing Federally Owned and Leased Buildings*. January 18, 2002.

ICSSC RP-8, *Standards of Seismic Safety for Existing Federally Owned and Leased Buildings*. December 2011.

International Agency for Research on Cancer (IARC) Monographs, volume 78. *IARC Monographs on the Evaluation of Carcinogenic Risks to Humans*, 2001.

International Atomic Energy Agency (IAEA)

Safety Series No. NS-G-1.3, *Instrumentation and Control Systems Important to Safety in Nuclear Power Plants*. 2002.

Safety Series No. 35-G1, *Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report*. 1994.

Standard DS407, version 6, *Criticality Safety for Facilities and Activities Handling Fissile Material*. April 2012.

International Code Consortium. *International Building Code 2000*. March 2000.

International Commission on Radiological Protection (ICRP)

ICRP 23, *Report on the Task Group on Reference Man*. 1975.

ICRP 30, *Limits for Intakes of Radionuclides by Workers*. 1979.

ICRP 39, *Principles for Limiting Exposure of the Public to Natural Sources of Radiation*. 1984.

ICRP 56, *Age-Dependent Doses to Members of the Public from Intake of Radionuclides—Part 1*. 1990.

ICRP 61, *Annuals Limits on Intake of Radionuclides by Workers Based on the 1990 Recommendations*, (Archived). 1990.

ICRP 66, *Human Respiratory Tract Model (HRTM)*. 1994.

ICRP 67, *Age-Dependent Doses to Members of the Public from Intake of Radionuclides—Part 2 Ingestion Dose Coefficients*. 1992.

ICRP 68, *Dose Coefficients for Intakes of Radionuclides by Workers*. 1994.

ICRP 69, *Age-Dependent Doses to Members of the Public from Intake of Radionuclides—Part 3 Ingestion Dose Coefficients*. 1995.

ICRP 71, *Age-Dependent Doses to Members of the Public from Intake of Radionuclides—Part 4 Inhalation Dose Coefficients*. 1995.

ICRP 72, *Age-Dependent Doses to the Members of the Public from Intake of Radionuclides—Part 5 Compilation of Ingestion and Inhalation Coefficients*. 1995.

ICRP 103, *The 2007 Recommendations of the International Commission on Radiological Protection*. 2007.

International Thermonuclear Experimental Reactor. *G A0 General Design Requirements Document 3 01-17-19 R1.0*, chapter 21, "Design Requirements and Guidelines Level 2."

IPS-E-PR-170, *Engineering Standard for Process Flow Diagram*. January 1996.

Laboratory Newwork.com. *Determining the Right Glovebox for Accurate Level of Containment*. January 9, 2013.

Los Alamos National Laboratory (LANL). LANL/BEAR Conference, *Uranium Lung Solubility Class Section at Bechtel Jacobs Company LLC-Operated Facilities*.

Maurya, Dr. D.K., *Types of Damages Induced After Radiation Exposure (Effect of Radiation)*. 2011.

Muldoon's Introduction to Health Physics Page. *Time, Distance, and Shielding*.

National Aeronautics and Space Administration (NASA)
Reliability and Data Analytics—Viewpoints, Tips, Examples.
NASA/SP-2011-3421, *Probabilistic Risk Assessment Procedures Guide for NASA Managers and Practitioners*. December 2011.

National Fire Protection Association (NFPA)
NFPA 58, *Liquefied Petroleum Gas Code*. 2014.
NFPA 704, *Identification of the Fire Hazards of Materials*. 2012.

National Institute of Standards and Technology (NIST). Special Publication 1026, *CFAST—Consolidated Model of Fire Growth and Smoke Transport*, version 6. April 2009.

National Renewable Energy Laboratory/TP-550-46698, *Understanding Fault Characteristics of Inverter-Based Distribution Energy Resources*. January 2010.

ncidavid.blogspot.com. *Magnehelic Basics*. September 2, 2013.

OneMine.org., National Institute for Occupational Safety and Health. *A Comparative Evaluation of the Differential-Pressure-Based Respirable Dust Dosimeter with the Personal Gravimetric Respirable Dust Sampler in Underground Coal Mines*. September 10, 2008.

Public Law 101-614, "An Act to Authorize Appropriations for the Earthquake Hazards Reduction Act of 1977, and for Other Purposes." November 16, 1990.

Radiation Resources. *How Particles Deposit in the Respiratory Tract*.

Reference Handbook, EG&G Rocky Flats, *Nuclear Criticality*, Revision 3. December 6, 1991.

Risk Management LTD. *MORT and All That: A 40-Year Old Technique Still of Value*.

Rocky Mountain Arsenal, Woodward Clyde Consultants, Draft, *Public Health Risk Assessment Report Submerged Quench Incinerator Task IRA-2 Basin F Liquids Treatment Design*. January 1990.

Science Daily. *Earthquake*. 2013.

Science Dictionary. *What is Criticality Incident?*

Stockholm Environment Institute. Foundation on Air Quality Management in Asia, **"**Modelling." 2008.

Technical Panel of the EQP Risk Assessment Forum, *Using Probabilistic Methods to Enhance the Role of Risk Analysis in Decision-Making with Case Study Examples*.

Terrauniversal.com. *Differential Pressure Gauges/Controller*, "Photohelic Pressure Control."

Universidad De Los Andes, Venezuela. *Biological Effects of Radiation*, "Cellular Damage and Possible Cellular Processes."

U.S. Department of Commerce. National Oceanic and Atmospheric Administration, National Ocean Service, *Temporary Emergency Exposure Limits (TEELs)*. January 3, 2014.

U.S. Department of Energy Directives (Guides, Manuals, Orders, and Policies)

DOE Guide 151.1-3, *Programmatic Elements Emergency Management Guide*. July 11, 2007.

DOE Guide 200.1-1, Chapter 4, "Requirements Definition Stage" (Archived). May 21, 1997.

DOE Guide 231.1-2, *Occurrence Reporting Causal Analysis Guide* (Archived). August 20, 2003.

DOE Guide 414.1-1B, *Management and Independent Assessment Guide for Use with 10 CFR Part 830, Subpart A, and DOE O 414.1C, Quality Assurance; DOE M 450.4-1, Integrated Safety Management System Manual; and DOE O 226.1A, Implementation of DOE Oversight Policy* (Archived). September 27, 2007.

DOE Guide 414.1-1C, *Management and Independent Assessments Guide*. March 27, 2014.

DOE Guide 414.1-2B, Admin Chg 1, *Quality Assurance Program Guide*. May, 8, 2013.

DOE Guide 414.1-4, *Safety Software Guide for Use with 10 CFR 830 Subpart A, Quality Assurance Requirements, and DOE O 414.1C, Quality Assurance*. June 17, 2005.

DOE Guide 420.1-1, *Nonreactor Nuclear Safety Criteria and Explosives Safety Criteria Guide for use with DOE O 420.1, Facility Safety* (Archived). March 28, 2000.

DOE Guide 420.1-1A, *Nonreactor Nuclear Safety Criteria and Explosives Safety Criteria Guide for use with DOE O 420.1C, Facility Safety*. December 4, 2012.

DOE Guide 421.1-1, *Criticality Safety Good Practices Program Guide for DOE Nonreactor Nuclear Facilities* (Archived). August 25, 1999.

DOE Guide 421.1-2, *Implementation Guide for use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830* (Archived). October 24, 2001.

DOE Guide 421.1-2A, *Implementation Guide for use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830*. December 19, 2011.
DOE Guide 423.1-1A, *Implementation Guide for use in Developing Technical Safety Requirements*. November 3, 2010.
DOE Guide 424.1-1A, *Implementation Guide for Use in Addressing Unreviewed Safety Question Requirements*, (Archived). July 24, 2006.
DOE Guide 424.1-1B, Admin Chg 2, *Implementation Guide for Use in Addressing Unreviewed Safety Question Requirements*. June 12, 2013.
DOE Guide 441.1-1C, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835, Occupational Radiation Protection*. November 18, 2010.
DOE Guide 450.4-1B, *Integrated Safety Management System Guide for use with Safety Management System Policies (DOE P 450.4, DOE P 450.5, and DOE P 450.6); The Functions, Responsibilities, and Authorities Manual; and The Department of Energy Acquisition Regulation,* (Archived). March 1, 2001.
DOE Guide 450.4-1C, *Integrated Safety Management System Guide*. September 29, 2011.
DOE Guide 460.1-1, *Implementation Guide for Use with DOE O 460.1A, Packaging and Transportation Safety*. June 5, 1997.
DOE Manual 441.1-1, *Nuclear Material Packaging Manual*. March 7, 2008.
DOE Order 151.1C, *Comprehensive Emergency Management System*. November 2, 2005.
DOE Order 225.1B, *Accident Investigations*. March 4, 2011.
DOE Order 226.1A, *Implementation of Department of Energy Oversight Policy*. July 31, 2007.
DOE Order 226.1B, *Implementation of Department of Energy Oversight Policy*. April 25, 2011.
DOE Order 231.1B, Admin Chg 1, *Environment, Safety and Health Reporting*. June 27, 2011.
DOE Order 232.2, Admin Chg 1, *Occurrence Reporting and Processing of Operations Information*. August 30, 2011.
DOE Order 410.1, *Central Technical Authorities Responsibilities Regarding Nuclear Safety Requirements*. August 28, 2007.
DOE Order 413.3B, *Program and Project Management for the Acquisition of Capital Gains*. November 29, 2010.
DOE Order 414.1C, Admin Chg 1, *Quality Assurance*, (Archived). June 17, 2005.
DOE Order 414.1D, Admin Chg 1, *Quality Assurance*. April 25, 2011.
DOE Order 420.1C, *Facility Safety*. December 4, 2012.
DOE Order 422.1, Admin Chg 1, *Conduct of Operations*. June 29, 2010.
DOE Order 425.1D, Admin Chg 1, *Verification of Readiness to Start Up or Restart Nuclear Facilities*. April 2, 2013.
DOE Order 440.1B, Admin Chg 1, *Worker Protection Program for DOE (Including the National Nuclear Security Administration) Federal Employees*. May 17, 2007.
DOE Order 442.1A, *Department of Energy Employee Concerns Program*. June 6, 2001.
DOE Order 460.1C, *Packaging and Transportation Safety*. May 14, 2010.
DOE Order 461.1B, *Packaging and Transportation for Offsite Shipment of Materials of National Security Interest*. December 16, 2010.

DOE Order 461.2, *Onsite Packaging and Transfer of Materials of National Security Interest*. November 1, 2010.
DOE Order 5480.19, Chg 2, *Conduct of Operations Requirements for DOE Facilities*, (Archived). July 9, 1990.
DOE Order 5480.22, Chg 2, *Technical Safety Requirements* (Archived). September 15, 1992.
DOE Order 5480.23, *Nuclear Safety Analysis Reports* (Archived). April 30, 1992.
DOE Order 5480.26, *Trending and Analysis of Operations Information Using Performance Indicators* (Archived). January 15, 1993.
DOE Order 5480.30, Chg 1, *Nuclear Reactor Safety Design Criteria*. November 18, 2010.
DOE Policy 226.1B, *Department of Energy Oversight Policy*. April 25, 2011.
DOE Policy 420.1, *Department of Energy Nuclear Safety Policy*. February 8, 2011.
DOE Policy 450.4A, *Integrated Safety Management Policy*. April 25, 2011.

U.S. Department of Energy Handbooks and Standards

DOE-HDBK-1011/1-92, *Electrical Science,* volume 1. June 1992.
DOE-HDBK-1013/1-92, *Instrumentation and Control,* volume 1. *June 1992.*
DOE-HDBK-1014/2-92, *Mathematics*, volume 2. June 1992.
DOE-HDBK-1016-93, *Engineering Symbology, Prints, and Drawings*, volumes 1 and 2. January 1993.
DOE-HDBK-1019/1-93, *DOE Fundamentals Handbook, Nuclear Physics and Reactor Theory*, *v*olume 1. January 1993.
DOE-HDBK-1028-2009, *Human Performance Improvement Handbook,* volume 1, "Concepts and Principles." June 2009.
DOE-HDBK-1081-94, *Primer on Spontaneous Heating and Pyrophoricity*. December 1994.
DOE-HDBK-1081-YR, *Primer on Spontaneous Heating and Pyrophoricity*. (Draft).
DOE-HDBK-1085-95, *DOE Enforcement Program Roles and Responsibilities.* (Archived). September 1998.
DOE-HDBK-1108-2002, Chg 2, *Radiological Safety Training for Accelerator Facilities*. July 2013.
DOE-HDBK-1122-99, *Radiological Control Technician Training* module 2.04 (Archived). June 1999.
DOE-HDBK-1129-99, *Tritium Handling and Safe Storage* (Archived). March 1999.
DOE-HDBK-1130-2008, *Radiological Worker Training*. December 2008.
DOE-HDBK-1132-99, *Design Considerations*. April 1999.
DOE-HDBK-1141-2008, *Radiological Assessor Training*. August 2008.
DOE-HDBK-1163-2003, *Integration of Multiple Hazard Analysis Requirements and Activities*. October 2003.
DOE-HDBK-1169-2003, *Nuclear Air Cleaning Handbook*. 2003.
DOE-HDBK-1208-2012, *Accident and Operational Safety Analysis*. July 2012.
DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*. December 1994.
DOE-STD-1020-2002, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities* (Archived). January 2002.

DOE-STD-1020-2012, *Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities*. December 2012.
DOE-STD-1021-93, Chg 1, *Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components*. July 1993.
DOE-STD-1023-95, *Natural Phenomena Hazards Assessment Criteria*. March 1995.
DOE-STD-1027-92, Chg 1, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. September 1997.
DOE-STD-1027-92, Chg 1, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, Attachment 1, "Hazard Categorization of DOE Facilities." September 1997.
DOE-STD-1029-92, Chg 1, *Writer's Guide for Technical Procedures*. December 1998.
DOE-STD-1085-95, *DOE Enforcement Program Roles and Responsibilities* (Archived). August 1995.
DOE-STD-1098-2008, *Radiological Control*. October 2008.
DOE-STD-1104-2009, *Review and Approval of Nuclear Facility Safety Basis and Safety Design Basis Documents*. May 2009.
DOE-STD-1120-2005, *Integration of Environment, Safety, and Health into Facility Disposition Activities*. April 2005.
DOE-STD-1128-2008, *Good Practices for Occupational Radiological Protection in Plutonium Facilities* (Archived). December 2008.
DOE-STD-1128-2013, *Good Practices for Occupational Radiological Protection in Plutonium Facilities*. April 2013.
DOE-STD-1134-99, *Review Guide for Criticality Safety Evaluations*. September 1999.
DOE-STD-1135-99, *Guidance for Nuclear Criticality Safety Engineer Training and Qualification*. September 1999.
DOE-STD-1183-2007, *Nuclear Safety Specialist Functional Area Qualification Standard*. November 2007.
DOE-STD-1186-2004, *Specific Administrative Controls*. August 2004.
DOE-STD-1189-2008, *Integration of Safety into the Design Process*. March 2008.
DOE-STD-3007-2007, *Guidelines for Preparing Criticality Safety Evaluations at Department of Energy Nonreactor Nuclear Facilities*. February 2007.
DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analysis* (Archived). July 1994.
DOE-STD-3009-94, Chg 3, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analysis*. March 2006.
DOE-STD-3011-2002, *Guidance for Preparation of Basis for Interim Operation (BIO) Documents*. December 2002.
DOE-STD-3014-2006, *Accident Analysis for Aircraft Crash into Hazardous Facilities*. May 2006.
DOE-NA-STD-3016-2006, *Hazard Analysis Reports for Nuclear Explosive Operations*. May 2006.
DOE-STD-6003-96, *Safety of Magnetic Fusion Facilities: Guidance*. May 1996.

U.S. Department of Energy Other References

DOE/EH-0173T, *Environmental Regulatory Guide for Radiological Effluent Monitoring and Environmental Surveillance*. January 1991.

DOE/EH-0711, *Gap Analysis for IMBA and DOE Safety Software Central Registry Recommendation*. August 2006.

DOE/EH-4.2.1.2-Criteria, *Software Quality Assurance Plan and Criteria for the Safety Analysis Toolbox Codes*. November 2003.

DOE/EM-5503-94, *EM Health and Safety Plan Guidelines* (Archived). December 1994.

DOE PNNL-18348, *Masters' Thesis—Criticality Alarm System Design Guide with Accompanying Alarm System Development for the Radioisotope Production L*. December 2009.

Energy Facility Contractors Group, *EM Hazard Categorization Guidance*.

Lawrence Livermore National Laboratory, UCRL-AR-133352 Rev 1, *HVAC Systems in Nonreactor Nuclear Facilities*. May 1999.

NNSA/Service Center, Environment, Safety and Health Department, *Assessment Handbook*. June 2006.

Oak Ridge Operations Office. *Nuclear Facility Safety Basis Fundamentals Self-Study Guide*. November 2002.

Office of Enforcement and Oversight, Office of Health, Safety and Security U.S. Department Of Energy, *Safety and Security Enforcement Process Overview*. August 2012.

Savannah River Site, *DNFSB Recommendation 2004-2, Ventilation System Evaluation*, revision 3. March 2009.

SCIE-DOE-01-TRAC-14-95, *Events and Causal Factors Analysis*. August 1995.

U.S. Department of Health and Human Services. Centers for Disease Control and Prevention, Emergency Preparedness and Response, *Acute Radiation Syndrome: A Fact Sheet for Clinicians*.

U.S. Department of Labor. OSHA, *Process Safety Management Guidelines for Compliance*. 1994.

U.S. Nuclear Regulatory Commission (U.S. NRC)

Glossary. December 11, 2013.

Human Resource Training Development, *Westinghouse Technology System Manual*, Section 16.0, "Radiation Monitoring System."

NUREG 9080, *Nuclear Regulatory Legislation*, September 2013.

NUREG/CR-6751, *The Human Performance Evaluation Process: A Reviewing the Identification and Resolution of Human Performance Problems*. March 2002.

Regulatory Guide 1.23, Revision 1, *Meteorological Monitoring Programs for Nuclear Power Plants*. March 2007.

Regulatory Guide 1.145, Revision 1, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. February 28, 1983.

Regulatory Guide 1.70, Revision 3, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*. November 1978.

Regulatory Guide 7.9, Revision 2, *Standard Format and Contents of Part 71 Applications for Approval of Packages for Radioactive Materials*. March 15, 2005.

Wikipedia

Air Purifier. February 5, 2014.
Criticality Accident. November 1, 2013.
Data Logger. October 30, 2013.
Failure Mode and Effects Analysis. January 4, 2014.
Fissile vs Fissionable. November 9, 2013.
Fume Hood. January 19, 2014.
Glossary of HVAC Terms. November 18, 2013.
Heat Exchanger. January 25, 2014.
HEPA. December 25, 2013.
HVAC Control System. September 22, 2013.
Neutron Poison. September 18, 2013.
Nuclear Criticality Safety. March 6, 2014.
Nuclear Fission Product. March 14, 2014.
Price-Anderson Nuclear Industries Indemnity Act. May 11, 2014.
Raschig Ring. July 31, 2013.
Respiratory Rate. December 2, 2013.
Seismic Analysis. November 8, 2013.
Severe Weather. January 9, 2014.

Nuclear Safety Specialist Qualification Standard
Reference Guide
August 2014