U.S. Department of Energy Orders Self-Study Program

DOE-STD-3009-94

PREPARATION GUIDE FOR U.S. DEPARTMENT OF ENERGY
NONREACTOR NUCLEAR FACILITY
DOCUMENTED SAFETY ANALYSES

National Nuclear Security Administration

DOE-STD-3009-94
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FAMILIAR LEVEL

OBJECTIVES

Given the familiar level of this module and the resources listed below, you will be able to answer the following questions:

1. What are five general requirements for contractors who are responsible for a hazard category 1, 2, or 3 nuclear facility, as related to establishing a safety basis?
2. What actions must a contractor take when it is made aware of a potential inadequacy of the documented safety analysis (DSA)?
3. What are the three contractor requirements related to technical safety requirements (TSRs)?
4. What is the safe harbor method used to prepare a DSA for an NNSA nonreactor nuclear facility?
5. What is the purpose of a preliminary DSA for a new facility?
6. What is the purpose of a final DSA?
7. What are the three types of TSRs?
8. What is the purpose of limiting conditions for operations?
9. What is the purpose of action statements as used in TSRs?
10. What is the purpose of an unreviewed safety question (USQ) determination?
11. What is the approval basis for DSAs?

Note: If you think that you can complete the practice at the end of this level without working through the instructional material and/or the examples, complete the practice now. The course manager will check your work. You will need to complete the practice in this level successfully before taking the criterion test.

RESOURCES

10 CFR 830, “Nuclear Safety Management, Subpart B, Safety Basis Requirements.” January 1, 2011.
10 CFR 830.204, “Documented Safety Analysis.” January 1, 2011.
10 CFR 830.205, “Technical Safety Requirements.” January 1, 2011.
29 CFR 1910.119, “Process Safety Management of Highly Hazardous Chemicals.” July 1, 2010.
DOE G 421.1-2, *Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830*. 10/24/01.
DOE G 423.1.1, *Implementation Guide for Use in Developing Technical Safety Requirements.* 11/3/10.
DOE G 424.1-1, *Implementation Guide for Use in Addressing Unreviewed Safety Question Requirements*. 4/8/10.
DOE O 420.1B, *Facility Safety*. 12/22/05.
DOE O 422.1, *Conduct of Operations*. 6/29/10.
DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*. March 2000.
DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. September 1997.
DOE-STD-1104-2009, *Review and Approval of Nuclear Facility Safety Basis Documents (Documented Safety Analyses and Technical Safety Requirements)*. May 2009.
DOE-STD-1186, *Specific Administrative Controls*. August 2004.
DOE-STD-3009-94, chg 3, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facilities Documented Safety Analyses.* March 2006.
U.S. Nuclear Regulatory Commission, NRC Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. February 1983.
U.S. Nuclear Regulatory Commission, NRC Regulatory Guide 1.23, *Meteorological Monitoring Programs for Nuclear Power Plants*. March 2007.
U.S. Nuclear Regulatory Commission, NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licenses*. August 1991.

INTRODUCTION

The familiar level of this module is divided into three sections. The first section is an introduction to DOE-STD-3009-94. In the second section, we will introduce the 17 chapters of a documented safety analysis (DSA). The third section covers the evaluation guideline that appears in appendices 1–5 of the standard. We have provided several examples and practices throughout the module to help familiarize you with the material. The practice will also help prepare you for the criterion test.

Before continuing, you should obtain a copy of the references. You may need to refer to these documents to complete the examples, practices, and criterion test.

SECTION 1, INTRODUCTION TO DOE-STD-3009-94

DOE-STD-3009-94 describes a DSA preparation method that is acceptable to the National Nuclear Security Administration (NNSA). It was developed to assist hazard category 2 and 3 facilities in preparing DSAs that will satisfy the requirements of 10 CFR 830, "Nuclear Safety Management."

Beyond conceptual design and construction, the methodology in the standard is applicable to the spectrum of missions expected to occur over the lifetime of a facility. As the phases of facility life change, suitable methodology is provided to update an existing DSA and to develop a new DSA if the new mission is no longer adequately encompassed by the existing DSA. This integration of the DSA with changes in facility mission and associated updates should be controlled as part of an overall safety management plan.

A unique element of DSA documentation is the required provisions for decontamination and decommissioning (D&D). This forward-looking aspect of facility operations is independent of facility mission and is intended to be a means of ensuring that current facility operations take into account D&D operations that will occur in the future.

For facilities transitioning into D&D, the safety basis of the D&D operations is documented throughout a DSA. This DSA, of which the principal emphasis is on the D&D operations themselves, provides the necessary analysis and supporting information to describe the facilities as they undergo shutdown, deactivation, decontamination, and decommissioning or dismantlement. The facility consists of the physical building, its constituent components, and the actual processes of D&D being performed. The DSA also includes the temporary engineering and administrative controls used to maintain the safety basis. This description and evaluation would envelop major configurations during the D&D operations for which the safety basis is sought. This is consistent with the intent of DSAs for operating facilities where not all operations conducted are detailed in the DSA.

Guiding Principles

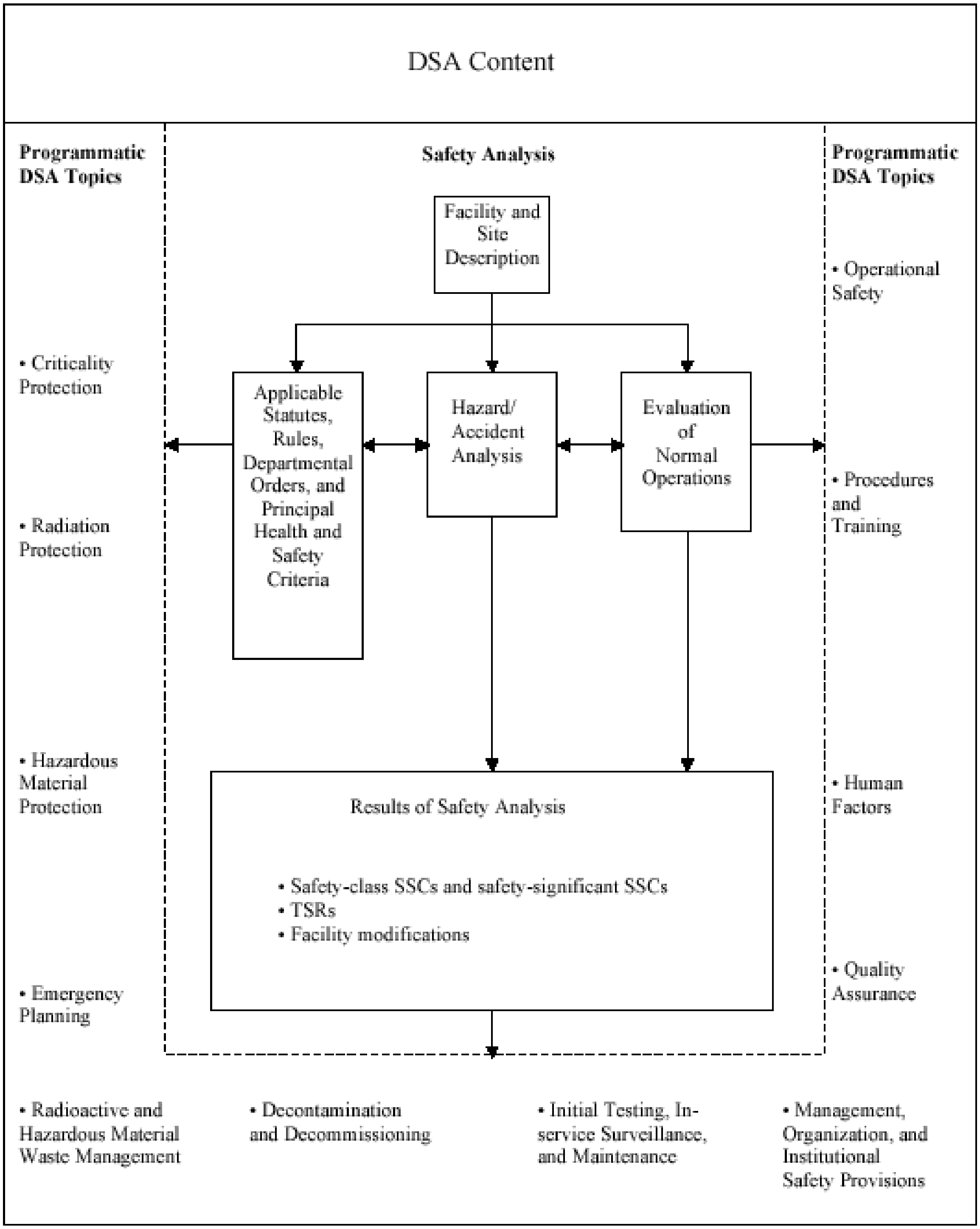
DOE-STD-3009-94 incorporates and integrates approaches regarding DSA format and content. To ensure a consistent application of the standard among users, the following guiding principles are provided:

- The focus of the standard is on hazard category 2 and hazard category 3 facilities.

- Hazard analysis and accident analysis are merged to ensure that the proper emphasis is placed on identification and analysis of hazards. The hazard analysis distinguishes when accident analysis is required as a function of potential offsite consequence.
- Defense in depth, worker safety, and environmental issues are identified in the hazard analysis.
- Defense in depth, as discussed in the standard, consists of two components:
  - 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
  - Integrated safety management programs that control and discipline operations
- Guidance is provided for evaluating the safety of a facility for which documentable, deterministic design basis accidents (DBAs) do not exist to establish bounding accidents that envelope the safety of existing facilities. Guidance is also provided on the treatment of beyond DBAs.
- Distinction is made between safety-class (SC) structures, systems, and components (SSCs), and safety-significant (SS) SSCs, and the balance of facility SSCs. Safety-class SSCs are related to public protection and are defined by comparison with the numerical evaluation guideline (EG). Safety-significant SSCs are identified for specific aspects of defense in depth and worker safety as determined by the hazard analysis.
- 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. 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, of the standard provides a brief, integrated overview of the facility SSCs.
  - Chapter 4, Safety Structures, Systems, and Components, of the standard 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 safety management programmatic requirements identified in 10 CFR 830, and illustrated in figure 1, form the boundaries within which the safety analysis is performed and represent the means of ensuring safe operation of the facility. Hazard analysis and accident analysis are performed to identify specific controls and improvements that feed back into overall safety management. Consequence and likelihood estimates obtained from this process also form the bases for grading the level of detail and control needed in specific programs. The result is documentation of the safety basis that emphasizes the controls needed to maintain safe operation of a facility.

The level of detail provided in the DSA depends on numerous factors. Applying the graded approach assists the preparer in establishing an acceptable level of detail.



Source: DOE-STD-3009-94

Figure 1. DSA scope and integration

The foundation for effectively preparing a DSA is the assembly and integration of an experienced preparation team. The size and makeup of the team depend on the magnitude and type of facility hazards and the complexity of the processes that are addressed in the DSA. In determining the makeup of the preparation team, careful consideration should be given to the key hazard analysis activity. The safety analysis base team should include individuals experienced in process hazard and accident analyses, facility systems engineers, and process operators. Individuals with experience in specific subject matter such as nuclear criticality, radiological safety, fire safety, chemical safety, or process operations may be needed in the hazard analysis on a regular or as-needed basis. Consistent, accurate exchange of information among the team members is at least as important as the makeup of the team itself. This can be ensured through meaningful integration of the required tasks.

Once team makeup is determined, base information needed to support DSA development is gathered. Maximum advantage should be taken of pertinent existing safety analyses and design information that are immediately available, or can be retrieved through reasonable efforts. Other information arises from existing sources such as process hazards analyses, fire hazards analyses, explosive safety analyses, health and safety plans, and environmental impact statements. The need for additional or specific information becomes apparent throughout the hazard analysis process. The remaining key steps for efficient completion of the safety analysis and DSA development process are:

- Identify the DSA project functions, using project information, and ensure the team matches the functions that are required.
- Perform a hazard analysis to provide facility hazard classification, evaluate worker safety and defense in depth, and identify unique and representative accidents to be carried forward to an accident analysis. Safety-significant SSCs and TSRs are designated in hazard analysis as well.
- Perform an accident analysis and assess the results to identify any SC SSCs and accident-specific TSRs that are based on comparison of accident consequences to the evaluation guideline.
- Develop the chapters for the DSA by providing the information necessary to support the results of the safety analysis. These chapters detail the results of the analysis, describe the facility and the safety SSCs and the safety management programs that relate to the facility safety basis.
- Prepare the executive summary.

Several specific topics are directly relevant to understanding the conceptual basis of the standard. These topics are worker safety, defense in depth, programmatic commitments, SSC and TSR commitments, and correlation of the standard to 10 CFR 830 requirements.

Worker Safety

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 requires answering the fundamental question of how worker safety is most appropriately addressed in the DSA.

The Occupational Health and Safety Administration (OSHA) has published 29 CFR 1910.119, “Process Safety Management of Highly Hazardous Chemicals.” 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 this standard are intended to eliminate or mitigate the consequences of such releases." Many of the topics requiring coverage in this Federal regulation, such as design codes and standards, process hazard analysis, human factors, training, etc., are directly parallel to the requirements in 10 CFR 830.

Defense in Depth

Defense in depth 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 upon. To compensate for potential human and mechanical failures, defense in depth 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 defense-in-depth 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 defense in depth is imposed. However, defining defense in depth 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 defense-in-depth 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 defense in depth. The inner layer of defense in depth relies upon 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 and, thus, minimize challenges to the next layer of defense.

In the event that the inner layer of defense in depth 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 defense in depth 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 defense in depth. Passive, automatically or manually activated features, and/or safety management programs 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.

Structures, systems, or components that are major contributors to defense in depth are designated as SS SSCs. Additionally, this standard provides guidance on grading the safety management programs 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 hazard analysis and is an integral part of defense in depth.

Administrative controls (AC) that are major contributors to defense in depth are designated as specific administrative controls (SAC) 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, along with DOE-STD-1186-2004, *Specific Administrative Controls*, provides guidance applicable to these types of controls. SACs provide preventive and/or mitigative functions for specific potential accident scenarios, which also 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 423.1-1, *Implementation Guide for Use in Developing Technical Safety Requirements*, provides screening criteria for converting existing technical specifications and operational safety requirements into TSRs. The screening criteria are considered a generally reasonable set of criteria to designate TSRs for defense in depth. The safety items identified in the hazard analysis 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 defense in depth.

Safety Management Program Commitments

10 CFR 830.204, "Documented Safety Analysis," requires that the DSA define the characteristics of the safety management programs necessary to ensure the safe operation of the facility.

Program commitments encompass a large number of details that are more appropriately covered in specific program documents external to the DSA. The cumulative effect of these details, however, is recognized as being important to facility safety, which is the rationale for a top-level program commitment becoming part of the safety basis.

NNSA facilities that use and rely on site-wide, safety support services, organizations, and procedures, may summarize the applicable site-wide documentation provided its interface with the facility is made clear. The DSA then notes whether the reference applies to a specific commitment in a portion of the referenced documentation or is a global commitment to maintaining a program for which a number of details may vary without affecting the global commitment. Any documents referenced in the DSA must be available upon request.

TSR and SSC Commitments

To comply with 10 CFR 830, specific safety controls must 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, highly detailed descriptions are reserved for two categories of SSCs comprising the most crucial aspects of facility safety. These two categories are SC SSCs 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) safety limits (SLs); 2) operational limits consisting of limiting control settings (LCSs) and limiting conditions for operation (LCOs) and associated surveillance requirements (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) SC SSCs and controls established on the basis of application of the EG; 2) SS SSCs; 3) defense in depth in accordance with the screening criteria of DOE G 423.1-1; and 4) safety management programs for defense in depth or worker safety. The bases appendix provides the linkage to the DSA. It is important to develop TSRs judiciously.
- Safety-class SSCs. 10 CFR 830 defines SC designation for SSCs that are established on the basis of application of the EGs. This designation carries with it the most stringent requirements. Appendix A provides guidance for implementing the EG to classify SSCs as SC SSCs.
- Safety-significant SSCs. This category of SSCs is provided to ensure that important SSCs will be given adequate attention in the DSA and facility operations programs. Safety-significant SSCs are those of particular importance to defense in depth or worker safety as determined in hazard analysis. Control of such SSCs does not require meeting the level of stringency associated with SC SSCs. Safety-class SSCs are designated to address public risk, which makes a dose guideline at the site boundary a useful tool. Safety-significant SSCs address risk for all individuals within the site boundary as well as additional defense in depth for the public, making a dose guideline at any one point an artificial distinction distorting the process of systematically evaluating SSCs.
- Specific administrative controls. 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.

Note: You do not have to do example 1 on the following pages, but it is a good time to check your skill and knowledge of the information covered. You may do example 1 or go to section 2.

EXAMPLE 1

Using the familiar level of this module and the resources, answer the following questions.

1. What are the two components that make up defense in depth?

2. What are the conditions in which an accident can be defined as a DBA?

3. What is the purpose of DOE-STD-3009-94?

Note: When you are finished, compare your answers to those contained in the example 1 self-check. When you are satisfied with your answers, go to section 2.

EXAMPLE 1 SELF-CHECK

1. What are the two components that make up defense in depth?
   - Defense in depth, as discussed in the standard, consists of two components: 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.
   - Integrated safety management programs that control and discipline operations.

2. What are the conditions in which an accident can be defined as a DBA?
   An accident can be defined as a DBA if relevant SSCs are specifically designed to function during that accident and appropriate documentation exists.

3. What is the purpose of DOE-STD-3009-94?
   DOE-STD-3009-94 was developed to assist hazard category 2 and 3 facilities in preparing DSAs that will satisfy the requirements in 10 CFR 830, “Nuclear Safety Management.”

SECTION 2, PREPARATION GUIDANCE

This section summarizes the contents of the seventeen chapters of a DSA. The standard includes preparation guidance to ensure consistent and appropriate treatment of all DSA requirements for the various NNSA nonreactor nuclear facilities. Please refer to the standard for detailed guidance for each of the chapters. You may need this information to answer questions in the practice and in the criterion test.

Chapter 1, Site Characteristics

Chapter one should provide a description of site characteristics necessary for understanding the facility environs important to the safety basis. The chapter covers the following topics and issues that are typically included in a DSA:

- Description of the location of the site, location of the facility within the site, its proximity to the public and to other facilities, and identification of the point where the evaluation guideline are applied
- Specification of population sheltering, population location and density, and other aspects of the surrounding area of the site that relate to assessment of the protection of the health and safety of the public
- Determination of the historical basis for site characteristics in meteorology, hydrology, geology, seismology, volcanology, and other natural events
- Identification of design basis natural events
- Identification of sources of external accidents
- Identification of nearby facilities impacting, or impacted by, the facility
- Validation of site characteristic assumptions common to safety analysis that were used in prior environmental analyses and impact statements, or of the need to revise and update such assumptions used in facility environmental impact statements

Chapter 2, Facility Description

Chapter two should provide 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 hazard analysis and accident analysis, not just safety SSCs. Information in this chapter typically includes the following:

- An overview of the facility, its inputs, and its outputs, including mission and history
- A description of the facility structure and design basis
- A description of the facility process systems and constituent components
- Instrumentation, controls, operating parameters, and relationships of SSCs
- A description of confinement systems
- A description of the facility safety support systems
- A description of the facility utilities
- A description of facility auxiliary systems and support systems

Chapter 3, Hazard and Accident Analysis

The purpose of chapter three is to provide information that will satisfy the requirements of 10 CFR 830 to evaluate normal, abnormal, and accident conditions, including consideration of natural and man-made external events, identification of energy sources or processes that might contribute to the generation or uncontrolled release of radioactive and other hazardous materials, and consideration of the need for analysis of accidents which may be beyond the design basis of the facility.

Chapter three 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 also presents the results of this hazard identification and assessment process. Hazard analysis 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. Chapter three 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.
- Identification in the hazard analysis of the spectrum of potential accidents at the facility in terms of largely qualitative consequence and frequency estimates. The summary of this activity will also include
  - identification of planned design and operational safety improvements;
  - summary of defense in depth, including identification of SS SSCs, SACs and other items needing TSR coverage in accordance with10 CFR 830;
  - summary of the significant worker safety features, including identification of SS SSCs and any relevant programs to be covered under TSR and administrative controls, 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 hazard analysis. 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 hazard analysis
  - Documentation of accident assumptions and identification of SC SSCs based on the EG

Chapter 4, Safety Structures, Systems, and Components

The purpose of chapter four is to provide information that is necessary to support the safety basis requirements of 10 CFR 830 for derivation of hazard controls. Chapter four provides details on those facility SSCs that are necessary for the facility to protect the public, provide defense in depth, or contribute to worker safety. Descriptions are provided of the attributes that are 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, including safety functions
- Identification of support systems that safety SSCs depend on to carry out safety functions
- Identification of the functional requirements that are necessary for the safety SSCs to perform their safety functions, and the general conditions caused by postulated accidents under which the safety SSCs 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

Chapter 5, Derivation of Technical Safety Requirements

The purpose of chapter five is to provide information that is necessary to support the safety basis requirements in 10 CFR 830 for the derivation of hazard controls.

Chapter five builds on the control functions determined to be essential in chapter three, and chapter four, to derive TSRs. Chapter five is meant to support and provide the information necessary for the separate TSR document required by 10 CFR 830.205, “Technical Safety Requirements.”

Derivation of TSRs consists of summaries and references to pertinent sections of the DSA in which design and administrative features are needed to prevent or mitigate the consequences of accidents. Design and administrative features addressed include ones that: 1) provide significant defense in depth; 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 specific control features 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

Chapter 6, Prevention of Inadvertent Criticality

The purpose of chapter six is to provide information that will support the development of a safety basis in compliance with the provisions of 10 CFR 830.204 regarding the definition of a criticality safety program.

Expected products of this chapter include:

- 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 nuclear criticality safety 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 administrative controls used by the criticality safety program

Chapter 7, Radiation Protection

Chapter seven summarizes provisions for radiation protection. Summaries focus on radiation protection based on facility hazards to provide a basic understanding of the scope of the radiation protection program. Expected products of this chapter, as applicable based on the graded approach, include the following:

- Description of the overall radiation protection program and organization
- Description of the radiological as low as reasonably achievable (ALARA) policy and program
- Description of radiation exposure control including administrative limits, radiological practices, dosimetry, and respiratory protection
- Identification of radiological monitoring to protect workers, the public, and the environment
- Discussion of radiological protection instrumentation
- Description of the plans and procedures for maintaining records of radiation sources, releases, and occupational exposures

Chapter 8, Hazardous Material Protection

Chapter eight summarizes provisions for hazardous material protection other than radiological hazards. Summaries focus on hazardous material protection based on facility hazards to provide a basic understanding of the scope of the hazardous material protection program. Expected products of this chapter, as applicable based on the graded approach, include the following:

- A description of the overall hazardous material protection program and organization
- A description of the hazardous material ALARA policy and program
- A description of hazardous material exposure control
- An identification of hazardous material monitoring to protect workers, the public, and the environment
- A discussion of hazardous material protection instrumentation
- A description of the plans and procedures for maintaining hazardous material records, hazard communications, and occupational exposures

Chapter 9, Radioactive and Hazardous Waste Management

Chapter nine describes the provisions for radioactive and hazardous waste management. Expected products of this chapter, as applicable based on the graded approach, include the following:

- Description of the overall radioactive and hazardous waste management program and organization

- Description of the site-specific radioactive, mixed, and hazardous material waste management policy, objectives, and philosophy
- Identification of hazardous waste streams
- Description of the waste management process, and waste treatment and disposal systems

Chapter 10, Initial Testing, In-Service Surveillance, and Maintenance

Chapter ten describes the essential features of the testing, surveillance, and maintenance programs. Expected products of this chapter, as applicable based on the graded approach, include the following:

- Description of the facility initial testing program
- Description of the facility in-service surveillance program
- Description of the planned, predictive, preventive, and corrective facility maintenance programs

Chapter 11, Operational Safety

Chapter eleven discusses general aspects of operational safety. It specifically focuses on the bases for the conduct of operations program. Chapter eleven is intended to acknowledge the intent of conduct of operations, indicate the aspects of conduct of operations directly applicable to the facility, and summarize the main aspects of conduct of operations implementation at the facility.

Chapter eleven describes: 1) the bases for the conduct of operations program; and 2) the fire protection program. Expected products of this chapter, as applicable based on the graded approach, include the following:

- Identification of the aspects of conduct of operations directly applicable to the facility
- Integrated summary of the main features of the facility conduct of operations program
- Description of the facility fire protection program

Chapter 12, Procedures and Training

Chapter twelve describes the processes by which the technical content of the procedures and training programs are developed, verified, and validated. These processes will ensure that the facility is operated and maintained by personnel who are well qualified and competent to carry out their job responsibilities using procedures and training elements that have been well developed and are kept current by the use of feedback and continuous improvement.

A programmatic commitment to ongoing procedures and training programs is considered to be a necessary part of safety assurance. Expected products of this chapter, as applicable based on the graded approach, include the following:

- Summary of the overall facility procedures and training programs
- Description of the processes by which the form and content of procedures and training materials are developed, verified and validated for normal, abnormal, and emergency operations; surveillance testing and maintenance
- Summary of the processes for maintaining written procedures, training materials, and training records
- Summary of the processes for modifying procedures and training materials

- Summary of the methods used to feed back operations experience, new analyses, other DSA changes, etc., to the procedures and training programs
- Description of the mechanisms to identify and correct technical or human factor deficiencies

Chapter 13, Human Factors

Chapter thirteen focuses on human factors engineering, its importance to facility safety, and the design of the facility to optimize human performance. Human factors consist of:

- Human factors engineering that focuses on designing facilities, systems, equipment, and tools so they are sensitive to the capabilities, limitations, and needs of humans
- Human reliability analysis that quantifies the contribution of human error to the facility risk

Chapter thirteen demonstrates that human factors are considered in facility operations where humans are relied on for preventive actions and for operator mitigative actions during abnormal and emergency operations. In this respect, the human-machine interface is an integral part of facility safety and, thus, requires special treatment in the DSA. The emphasis is on human-machine interfaces required for ensuring the safety function of safety SSCs that are important to safety and on the provisions made for optimizing the design of those human-machine interfaces to enhance reliable human performance.

A complete discussion of human factors without application of the graded approach includes the following:

- Description of the human factors process for systematically inquiring into the importance of human factors in facility safety
- Description of human-machine interfaces with SS SSCs and SC SSCs that are important to safety
- Description of the systematic inquiry into the optimization human-machine interfaces with SS SSCs and SC SSCs to enhance human performance

Chapter 14, Quality Assurance

Chapter fourteen describes the provisions for a quality assurance program. Expected products of this chapter include the following:

- A description of the quality assurance program and organization
- A description of document control and records management
- A description of the quality assurance process ensuring that performed safety related work meets requirements

Chapter 15, Emergency Preparedness Program

Chapter fifteen summarizes the emergency preparedness functions and response at the facility. Expected products of this chapter, as applicable based on the graded approach, include the following:

- Identification of the scope of the facility emergency preparedness plan
- Description of the philosophy, objectives, organization, and emergency response of facility emergency preparedness

Chapter 16, Provisions for Decontamination and Decommissioning

Chapter sixteen describes provisions that facilitate future D&D of a facility. Design of significant modifications to an existing facility must consider provisions for D&D. Chapter sixteen also contains guidance on the description of the conceptual D&D plan for existing facilities. Expected products of this chapter include the following:

- A description of design features incorporated in major modifications of an existing facility to facilitate future D&D of the facility
- A description of operational considerations to facilitate future D&D
- A description of a conceptual D&D plan

Chapter 17, Management, Organization, and Institutional Safety Provisions

Chapter seventeen presents information on management, technical, and other organizations that support safe operation. Chapter seventeen also enumerates the requirements used to develop the safety management programs, includes descriptions of the responsibilities of and relationships between the non-operating organizations having a safety function and their interfaces with the line operating organization, and presents sufficient information on the safety management policies and programs to demonstrate that the facility operations are embedded in a safety conscious environment. Expected products of this chapter include the following:

- A description of the overall structure of the organizations and personnel with responsibilities for facility safety and interfaces between those organizations
- A description of the programs that promote safety consciousness and morale, including safety culture, performance assessment, configuration and document control, occurrence reporting, and staffing and qualification.

Note: You do not have to do example 2 on the following page, but it is a good time to check your skill and knowledge of the information covered. You may do example 2 or go to section 3.

EXAMPLE 2

1. What is included in chapter 1 of a DSA?

2. In what chapter of a DSA would you expect to see a summary of the use of radiation work permits?

3. What is the purpose of a DSA as it applies to facilities transitioning into D&D?

Note: When you are finished, compare your answers to those contained in the example 2 self-check. When you are satisfied with your answers, go on to section 3.

EXAMPLE 2 SELF-CHECK

1. What is included in chapter 1 of a DSA?
   - Description of the location of the site, location of the facility within the site, the site's proximity to the public and to other facilities, and identification of the point where the evaluation guideline is applied
   - Specification of population sheltering, population location and density, and other aspects of the surrounding area of the site that relate to assessment of the protection of the health and safety of the public
   - Determination of the historical basis for site characteristics in meteorology, hydrology, geology, seismology, volcanology, and other natural phenomena events to the extent needed for hazard and accident analyses
   - Identification of design basis natural events
   - Identification of sources of external accidents, such as nearby airports, railroads, or utilities such as natural gas lines
   - Identification of nearby facilities impacting, or impacted by, the facility under evaluation
   - Validation of site characteristic assumptions common to safety analysis that were used in prior environmental analyses and impact statements, or of the need to revise and update such assumptions used in facility environmental impact statements

2. In what chapter of a DSA would you expect to see a summary of the use of radiation work permits?
   Chapter 7, Radiation Protection

3. What is the purpose of a DSA as it applies to facilities transitioning into D&D?
   For facilities transitioning into D&D, the safety basis of the D&D operations is documented in a DSA. The DSA provides the necessary analysis and supporting information to describe the facilities as they undergo shutdown, deactivation, decontamination, and decommissioning or dismantlement.

SECTION 3, EVALUATION GUIDELINE

The EG specifies a numerical radiological dose value to be used in identifying SC SSCs. Calculation methods and assumptions needed to provide general consistency in dose estimation are also described, with relevant background and interpretation discussions included as appropriate.

The methodology provided in DOE-STD-3009-94 focuses on characterizing facility safety with or without well-documented design information. The EG construct is intended for use with existing facilities.

Evaluation Guideline

The evaluation guideline is 25 rem total effective dose equivalent (TEDE). The dose estimates to be compared to it are those received by a hypothetical maximally exposed offsite individual (MOI) at the site boundary for an exposure duration of two hours. The nominal exposure duration of two hours may be extended to eight hours for those release scenarios that are especially slow to develop. Dose calculations for comparison against the EG are based on the concept of an unmitigated release to determine if the potential level of hazard in the specific facility warrants SC SSC.

The value of 25 rem TEDE is not to be used as a hard pass/fail level. Unmitigated releases should be compared against the EG to determine if they challenge the EG, rather than exceed it. This is because consequence calculations are highly assumption-driven and uncertain.

There is no predetermined frequency cutoff value, such as 1E-6 per year, for excluding low frequency operational accidents. For operational accidents, there is no explicit need for a frequency component to the unmitigated release calculations since the determination of need is driven by the bounding consequence potential. Natural events are defined in terms of the frequency of the initiating event, while external events are defined with a cutoff frequency of 10^{-6} per year, conservatively calculated, or 10^{-7} per year, realistically calculated.

Unmitigated release is meant to consider material quantity, form, location, dispersability, and interaction with available energy sources, but not to consider safety features that would prevent or mitigate a release. Final dose estimations representing the anticipated behavior of the facility under accident conditions should be based on the mitigated DBAs, wherein full or partial functionality of SC SSCs is assumed. In cases where the designated SC SSCs are not capable of performing their required safety function without significant upgrade, other compensatory measures such as material-at-risk (MAR) limits may be implemented in the facility and incorporated into the DSA.

Comparison of the unmitigated consequences for a limited subset of potential accidents to the EG is performed to determine if the need for designation of SC SSCs exists. If the EG values are approached by the unmitigated consequences of a release scenario, a need for SC SSC designation is indicated. Safety-class SSCs are only one of many layers of hardware- and administrative-based controls that are incorporated into an NNSA operation for the protection of the public, worker, and environment consistent with the precepts of the defense-in-depth philosophy. The SC designation helps to focus a higher level of attention and requirements on this select subset of all controls intended for the protection of the public.

If the need for SC designation is determined, all preventive and mitigative features associated with the sequence of failures that result in a given release scenario and any features whose functionality is assumed as part of the scenario definition itself are candidates for SC SSC designation. The process of designating one or more safety SSCs as SC is judgment-based and depends on many factors, such as effectiveness, a general preference of preventive over mitigative and passive over active features, relative reliability, and cost considerations.

Dose Comparison Calculations

General discussion is provided for source term calculation and dose estimation and prescriptive guidance for the latter. The intent is that calculations should be based on reasonably conservative estimates of the various input parameters.

The dose estimate is that dose received during a two-hour exposure to plume, considering inhalation, direct shine, and ground shine. Other slow developing release pathways such as ingestion of contaminated food, water supply contamination, or resuspension are not included. However, quick release accidents involving other pathways, such as a major tank rupture that could release large amounts of radioactivity in liquid form to water pathways, should be considered. In this case, real potential uptake locations should be the evaluation points.

The airborne pathway is of primary interest for nonreactor nuclear facilities. This position is supported by NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licenses*, which states 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. For some types of facilities such as waste storage, the surface water and groundwater pathways may be more important, but accident releases usually develop more slowly than airborne releases. More time would also be available for implementing preventive and mitigative measures. Therefore, the emphasis on SC SSCs, in terms of immediate availability and operation, is not generally necessary for safety SSCs associated with these pathways.

Unmitigated Release Calculation

The unmitigated release calculation represents a theoretical limit to scenario consequences assuming that all safety features have failed, so that the physical release potential of a given process or operation is conservatively estimated. The unmitigated release should characterize both the energies driving the release, and the release fractions according to the physical realities of the accident phenomena at a given facility or process. As a result, some assumptions must be made to define a meaningful scenario. To capture these assumptions and their resulting potential impact on safety SSC designation and/or TSR protection, the unmitigated calculation should

- take no credit for active safety features, such as ventilation filtration systems in the case of a spill;
- take credit for passive safety features that are assessed to survive accident conditions where that capability is necessary to define a physically meaningful scenario;
- take no credit for passive safety features producing a leak path reduction in source term, such as building filtration; and
- assume the availability of passive safety features that are not affected by the accident scenario.

Design Basis Accident Calculation
Once a set of SC SSCs has been identified, accident consequences can be estimated in a DBA calculation that represents the accident scenario progression where SC SSCs successfully perform their intended safety function.

For each scenario in the DSA, sufficient documentation of the unmitigated and mitigated accident scenarios should be made so that the thought process of determining the SC SSCs is well understood. In all cases, the level of protection provided by the identified SC SSCs should be evident. However, this does not require explicit reporting of unmitigated consequences in the DSA, if it is evident that the unmitigated release consequences are large.

Source Term Calculation

The radioactive airborne source term is typically estimated as the product of five factors:

- Material at risk (MAR)
- Damage ratio (DR)
- Airborne release fraction (ARF)
- Respirable fraction (RF)
- Leak path factor (LPF)

MAR

The MAR values used in hazard and accident analyses must be consistent with the values noted in hazard identification and should represent documented maximums for a given process or activity. Such documentation may be present in TSRs or lower-tier documents referenced in TSRs, as necessary. While DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, excludes material in qualified containers from consideration for the purposes of hazard classification, the existence of such material should be acknowledged in a DSA. Such material should later be excluded from the source term for the applicable accident scenarios if the containers can perform their functions under the accident environments.

DR

The DR is that fraction of material affected by the accident generating conditions. DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, notes that some degree of ambiguity can result from overlapping definitions of MAR and DR in various applications. One consistent definition should be used throughout a given DSA.

ARFs and RFs

Bounding estimates for radionuclide ARFs and RFs for a wide variety of MAR and release phenomena are systematically presented in DOE-HDBK-3010-94. In those cases where there may be significant direct shine contribution to dose, that contribution should be evaluated without the use of the respirable fraction.

LPF

The LPF is the fraction of material passing through some confinement deposition or filtration mechanism. Several LPFs may be associated with a specific accident. For the purposes of the unmitigated release calculation, the LPF should be set to unity.

Dose Estimation

The relevant factors for dose estimation are receptor location, meteorological dispersion, and dose conversion values. Specific guidance for each is provided below.

Dose Calculation Location

For the purposes of comparison to the evaluation guideline, the comparison point is the location of a theoretical MOI standing at the site boundary. This location can also be beyond the NNSA site boundary if a buoyant or elevated plume is not at ground level at the NNSA site boundary. In such cases, the calculation location is taken at the point of maximum exposure, typically where the plume reaches the ground level. It is NNSA practice that onsite workers and the public are protected under the emergency response plans and capabilities of its sites. This protection, along with implementation of defense in depth and worker safety philosophy, SS and SC SSC designations, and NNSA's safety management programs, address onsite safety. However, an annual assessment of any changes in the site boundary and potential effects on safety SSC classification should be performed in association with the required annual update of the DSA for a facility. Privatization and site turnover initiatives may affect these determinations.

Atmospheric Dispersion

The 95th 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 regulatory position 3 of NRC Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, for evaluating consequences along the exclusion area boundary. The determination of distance to the site boundary should be made according to the procedure outlined in position 1.2 of Regulatory Guide 1.145. NRC Regulatory Guide 1.23, *Meteorological Monitoring Programs for Nuclear Power Plants*, describes acceptable means of generating the meteorological data upon 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. 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, which 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 evaluation guideline.

Functional Classification Process

The use of the EG is only one element in a larger safety SSC functional classification process that contributes to adequate safety. Other contributors are disciplined conduct of operations, training, and safety management programs such as radiation protection and emergency response. The functional classification process must recognize competing interests for resources and the need to optimize the application of resources for safety in a facility. Some principles that should be incorporated into a functional classification process include the following:

- Protection of the public is contributed to by all facets of safety in design.
- Protection of the public is paramount in safety design, but protection of workers is no less important. However, the degree of protection for facility workers achievable by safety SSCs is limited. Major contributions to overall safety assurance to the worker are institutional factors such as conduct of operations, training, and the safety management programs.
- Some considerations in the prioritization of facility safety issues, include the following:
    - Hazardous material inventory should be minimized at all times.
    - Safety SSCs are preferred over ACs.
    - Passive SSCs are preferred over active SSCs.
    - Preventive controls are preferred over mitigative controls.
    - Controls closest to the hazard may provide protection to both workers and the public.
    - Facility safety SSCs are preferred over personal protective equipment.
    - Controls that are effective for multiple hazards can be resource effective.

Additional Considerations

Selection of the term “evaluation guideline” is deliberate because it is different from safety or risk acceptance criteria and siting criteria. Such acceptance criteria have traditionally been used in the design and siting stage of nuclear power reactors.

Acceptance criteria have been linked to accident scenarios that are prescribed in some manner. The results of quantitative probabilistic risk assessments (PRAs), principally those of nuclear power or production facilities, are sometimes compared to another type of risk acceptance criteria, referred to as “safety goals.” PRAs are fundamentally different analytical methods from deterministic safety analyses and produce a different type of product. For example, in PRAs the failure of a safety feature (hardware or human action) to perform an intended function is always postulated, irrespective of the safety classification of the feature. Therefore, in contrast to assumptions employed in deterministic safety analyses, in PRAs even SC SSCs are not treated differently from typical, industrial grade SSCs in release scenario characterization, with the exception of their estimated failure probabilities.

A conceptually different approach is needed for safety analyses of existing facilities, where an analysis of the safety of the facility, as is, is performed. The primary objective of the analytical process must then turn to the identification of needed controls, their potential inadequacies, and the corresponding corrective or compensatory measures. Furthermore, for existing NNSA facilities, DBAs are typically either non-existent or irrelevant, due to a variety of reasons, such as changes in the original mission or early design philosophies. Thus, the standard adopted the notion of derivative DBAs that for simplicity of notation were summarized as DBAs in the text. However, these DBAs are not the actual accident scenarios that formed some aspects of the basis for the facility design. For these existing facilities, safety assurance is provided through an aggressive approach based on a comprehensive analysis of all hazards leading to the release of radiological or toxicological material, and ensuring that the controls identified against each hazard are relevant, specific, and effective.

It is emphasized again that the value of 25 rem TEDE is not to be used as a hard pass/fail level. Unmitigated releases should be compared against the evaluation guideline to determine if they challenge the evaluation guideline, rather than exceed it. This is because consequence calculations are highly assumption-driven and uncertain. There are uncertainties in initiating event intensity, plant SSC and personnel response, accident phenomenology, DRs, ARFs and RFs, and so on. Other factors may play a part in the decision, and the evaluation guideline value guides the decision-making process towards a level of uniformity that could not exist without some form of quantitative benchmark.

Note: You do not have to do example 3 on the following page, but it is a good time to check your skill and knowledge of the information covered. You may do example 3 or go directly to the practice.

EXAMPLE 3

1. What is the definition for the term “evaluation guideline?”

2. What is the purpose of a DBA calculation?

3. What are the relevant factors for dose estimation?

Note: When you are finished, compare your answers to those contained in the example 3 self-check. When you are satisfied with your answers, go on to the practice

EXAMPLE 3 SELF-CHECK

1. What is the definition for the term “evaluation guideline?”
   The radioactive material dose value that the safety analysis evaluates against. The evaluation guidelines are established to identify and evaluate SC SSCs.

2. What is the purpose of a DBA calculation?
   A DBA represents the accident scenario progression where SC SSCs successfully perform their intended safety function.

3. What are the relevant factors for dose estimation?
   The relevant factors for dose estimation are receptor location, meteorological dispersion, and dose conversion values.

PRACTICE

This practice is required if your proficiency is to be verified at the familiar level. The practice will prepare you for the criterion test. You will need to refer to the resources to answer the questions in the practice correctly. The practice and criterion test will also challenge additional analytical skills that you have acquired in other formal and on-the-job training.

1. What is the definition for the following terms: beyond design basis accident, decommissioning, decontamination, and design basis?

2. What is the purpose of DSAs?

3. What is the purpose of DOE-STD-3009-94?

4. What is the defense-in-depth approach to facility safety?

5. What is the purpose of a hazard analysis?

6. What are the expected products of chapter 2 of a DSA?

7. How is the graded approach applied to the derivation of TSRs?

8. What is typically included in chapter 10 of a DSA for the facility maintenance program?

9. What are the expected products of chapter 11 of a DSA?

10. What is the definition for the term "evaluation guideline?"

Note: The course manager will check your practice and verify your success at the familiar level. When you have successfully completed this practice, go to the general level module.

DOE-STD-3009-94
PREPARATION GUIDE FOR U.S. DEPARMENT OF ENERGY NONREACTOR NUCLEAR FACILITY DOCUMENTED SAFETY ANALYSES
GENERAL LEVEL

OBJECTIVES

Given the familiar level of this module, a scenario, and an analysis, you will be able to answer the following questions:

1. Is the contractor’s action plan correct? If not, state what should have been done.
2. Were the correct sections of the documented safety analysis (DSA) referenced? If not, state the correct sections.

Note: If you think that you can complete the practice at the end of this level without working through the instructional material and/or the examples, complete the practice now. The course manager will check your work. You will need to complete the practice in this level successfully before taking the criterion test.

RESOURCES

DOE Orders Self-Study Program, Familiar Level, DOE-STD-3009-94, *Preparation Guidance for U.S. Department Of Energy Nonreactor Nuclear Facility Documented Safety Analyses*, July 2011.

10 CFR 830, "Nuclear Safety Management, Subpart B, Safety Basis Requirements," January 1, 2011.

10 CFR 830.204, "Documented Safety Analysis." January 1, 2011.

10 CFR 830.205, "Technical Safety Requirements." January 1, 2011.

29 CFR 1910.119, "Process Safety Management of Highly Hazardous Chemicals." July 1, 2010.

DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*. March 2000.

DOE G 421.1-2, *Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830*. 10/24/01

DOE G 423.1.1, *Implementation Guide for Use in Developing Technical Safety Requirements.* 11/3/10.

DOE G 424.1-1, *Implementation Guide for Use in Addressing Unreviewed Safety Question Requirements*. 4/5/10.

DOE O 420.1B, *Facility Safety*. 12/22/05.

DOE O 422.1, *Conduct of Operations*. 6/29/10.

DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. September 1997.

DOE-STD-1104-96, *Review and Approval of Nuclear Facility Safety Basis Documents (Documented Safety Analyses and Technical Safety Requirements)*. May 2009.

DOE-STD-1186-2004, *Specific Administrative Controls*. August 2004.

DOE-STD-3009-94, chg 3, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facilities Documented Safety Analyses*. March 2006.

Nuclear Regulatory Commission, NRC Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. February 1983.

Nuclear Regulatory Commission, NRC Regulatory Guide 1.23, *Meteorological Programs in Support of Nuclear Power Plants*. September 1980.

NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licenses*. June 1985**.**

INTRODUCTION

The familiar level of this module included the safety documents that make up the safety basis for a nuclear facility. The general level applies the knowledge gained in the familiar level to an example scenario, which includes a situation, the actions taken to remedy the situation, and the requirements related to the situation. Students will be asked to review the contractor's actions and decide if they are correct. Students will also be asked to decide if the correct requirements were cited in each situation. Please refer to the resources to make your analysis and answer the questions. You are not required to complete the example. However, doing so will help prepare you for the criterion test.

Note: You do not have to do the example on the following page, but it is a good time to check your skill and knowledge of the information covered. You may do the example or go on to the practice.

EXAMPLE SCENARIO

A routine National Nuclear Security Administration (NNSA) survey of facility operations raised a concern regarding the level of detail present in fire inspection procedures used to satisfy technical safety requirements (TSR) surveillances. Follow-up by facility engineering personnel led to the discovery that the fire system surveillance procedure that was credited for satisfying TSR surveillances did not contain steps to perform the required inspections. The surveillances in question were the visual inspection of sprinkler heads and the visual inspection of sprinkler piping and fittings.

During a critique, personnel indicated that they had revised the template for the procedure that was credited for performance of the annual surveillance on the fire system. This revision removed the two sections that described the visual inspection of the sprinklers and the piping. These inspections were required by National Fire Protection Association (NFPA) code, but were not part of the TSRs at that time. These sections were removed in anticipation of a proposed change. This revision to the procedure template occurred sometime before September 2001. Subsequently, a revision of the procedure was sent to the facility for review in September 2001, but the revision sheet did not indicate that these two sections of the procedure had been removed during the earlier template revision.

A phase II critique was held. During the critique, it was learned that during the period that the template for the annual test was revised, there was no formal process for controlling such changes. The procedure writers typically sought to obtain consensus for the change, and then make the change once they reached some level of agreement.

Apart from the inspection, the tritium fire protection coordinator conducts a quarterly inspection of fire systems. Facility management conducted a review to determine if the TSR requirements were adequately satisfied by this procedure and concluded that they were, thus preventing an actual TSR violation. NNSA concurred with this conclusion and a white paper was written to provide documentation.

Coincidental to the procedure revision in September 2001, the tritium facilities implemented an update to the DSA in October of 2001. For the fire system limiting condition for operations, the following surveillance requirements were added:

- A five-year preventive maintenance on certain valves in the system
- Visual inspection of the sprinkler heads
- Visual inspection of the sprinkler piping and fittings

During the DSA implementation, a review was conducted to ensure that the newly applied TSR requirements were satisfied by the completion of a procedure. The annual inspection procedure that had just been completed was incorrectly credited as satisfying the TSR requirements under the assumption that it contained the required inspections.

Action plans taken:

- Review all fire system surveillances from September 2001 to June 2002 and ensure that they are complete.

- Revise the procedure to indicate that it fulfills the requirements of the fire system surveillance requirements for the visual inspection of the sprinklers, piping, and fittings.
- Develop a formal change control process for the revision, review, and approval of procedures.

Sections of the DSA where you find information related to the scenario:

- Section 10.4, In-Service Surveillance Program
- Section 12.3.1, Development of Procedures
- Section 12.3.2, Maintenance of Procedures
- Section 12.4.1, Development of Training

Take some time to review the example scenario and the actions the contractor took or didn't take to correct the situation. Then decide if the contractor's actions were complete and correct. Finally, determine if the requirements cited by the contractor apply to this situation. Write your answer below and then compare your answer to the one contained in the example self-check.

EXAMPLE SELF-CHECK

Your answer does not have to match the following exactly. You may have added more corrective actions or cited other requirements from the resources that apply. To be considered correct, you answer must include, at least the following.

The action plans taken were complete and correct.

The sections cited were correct. However, one additional section should have been included: Section 5.5.X.2, Surveillance Requirements.

Additionally, facilities that have DSAs with surveillance requirements that are performed by outside support work groups need to be diligent in ensuring that the documents used by the outside work groups perform the surveillance as required. A formal change control process, with facility involvement, for the implementing document is essential to ensure that the DSA requirements are being met.

Implementation of future DSAs will have the required rigor and diligence by the system engineer and management to ensure that lessons learned from this event are implemented. This will include a rigorous plan that has sufficient detail for the implementation, and rigorous checks and reviews will be performed to ensure that the facility and personnel have a complete understanding of the DSA.

PRACTICE

This practice is required if your proficiency is to be verified at the general level. If you are to be qualified as a facility representative, the practice will prepare you for the criterion test. You will need to refer to the CFR, guides, and standards listed in the resources to answer the questions in the practice correctly. The practice and the criterion test will also challenge additional analytical skills that you have acquired in other formal and on-the-job training for the facility representative position.

Please review the following scenario, and then answer these questions:

1. Is the contractor’s action plan correct? If not, state what should have been done.
2. Were the correct sections of DOE-STD-3009-94 cited? If not, state the correct sections.

SCENARIO

Personnel were working in a facility operating room preparing to obtain material characterization samples from fissile material items that were introduced into a glovebox earlier in the shift. The work was being performed per a standard operating procedure. During the preparations, the operators discovered that one of the fissile items in the glovebox was not authorized to be in the glovebox based on the item identifications previously recorded in the procedure. The operating procedure requires that before introducing fissile material into the glovebox, two operations personnel must verify that only authorized items will be introduced by completing a nuclear safety control step that requires that the item identification, recorded on the can, be compared to the authorized item identifications recorded in the procedure. Additionally, an engineering representative is required to perform the same verification per a nuclear safety control step in the procedure. When these verifications were performed, the operations and engineering personnel failed to recognize that the item that was subsequently introduced to the glovebox was not on the authorized item list in the procedure. The verification by operations personnel and an engineering representative of the item identifications on the fissile material cans versus the authorized list of item identifications previously recorded in the procedure are the two defenses for nuclear criticality safety double-contingency analysis initiating event. The unauthorized item in the glovebox exceeded the glovebox fissile mass limit. However, introduction of the material did not cause the criticality safety limit to be exceeded.

An investigation of the event revealed the following.

The first error occurred when the new pail labels reflected wrong item numbers based on item 7 and item 13 being repacked into the wrong pails outside of the repack hut. The common error precursor was complacency/overconfidence. Two operators and a radiological control inspector (RCI) were at the room for the purpose of sorting and repacking. The original pail removed from the vault was opened by operator 1. All radiological surveys were performed by the RCI. Items 7 and 13 were removed from the pail, and were placed outside of the repack hut for repackaging. Operator 2 stationed outside of the repack hut placed item 13 into pail B and item 7 into pail A.

Placement of the items should have been made by placing item 13 into pail A and item 7 into pail B. The A pail was transported to the vault and the B pail was transported to the dissolver maintenance room (DMR) for eventual introduction into the material characterization glovebox. The self-check of the work was less than adequate.

The second error occurred when markings on items were incorrectly verified against the master sequencing procedure before items were introduced into the material characterization glovebox. Two operators verified tamper-indicating device (TID) seals and pail identifiers, but did not verify item identifiers against the procedure. A nuclear safety specialist (NSS) verified pail and TID identifiers, but did not verify item identifiers, based on over-confidence that the right pails, items, and TID seals were to be bagged into the cabinet. The present procedure refers to “product container” identification instead of “item identification.

Normally, third-level operators sort and repack items to be transported to DMR. Material characterization operators then introduce material into the material characterization glovebox. When the material characterization operators (two) and the NSS perform verifications in DMR, it is usually the first time that they have seen the pails/items, which constitutes a fresh set of eyes verifying identification of pails/items.

For this event, one of the material characterization operators was the same individual that sorted and repacked the items. Personnel interviews revealed that the operator had a high confidence level that the right item was in the pail because the qualified operator had sorted and repacked the item. Based on the confidence level of the one operator, a less than adequate verification was performed by all of the operators in DMR.

Action plans taken:

- Engineering submitted a procedure change request to change all references where “product container” is listed to read “item” for clarification.
- The facility will issue an operating experience program lessons learned to facility operators covering the issues surrounding this event.

Sections of DOE-STD-3009-94 that apply to this scenario:

- Section 6.4, Criticality Controls
- Section 6.4.1, Engineering Controls

Take some time to review the scenario and determine if the contractor’s action plans were complete and correct, and determine if the correct sections from DOE-STD-3009-94 were cited.

Write your answers on the next page and then bring the completed practice to the course manager for review.

Note: The course manager will check your practice and verify your success at the general level. When you have successfully completed this practice, the course manager will give you the criterion test.