GENERIC SAFETY ANALYSIS REPORT

CHAPTER 3

HAZARD AND ACCIDENT ANALYSIS (U)

September 1999

Westinghouse Savannah River Company
Aiken, SC 29808

SAFETY • RESPONSIBILITY • SECURITY
SRS
SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

DISCLAIMER

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ACRONYMS AND ABBREVIATIONS

ARF	Airborne Release Fraction
BIO	Basis for Interim Operation
CEDE	Committed Effective Dose Equivalent
DBA	Design Basis Accident
DID	Defense-in-Depth
DOE	Department of Energy
DR	Damage Ratio
EG	Evaluation Guideline
ERPG	Emergency Response Planning Guide
FDB	Fault Tree Data Bank
HA	Hazard Analysis
HC	Hazard Classification
HE	Hazard Evaluation
ICR	Incremental Cancer Risk
LHS	Latin Hypercube Sampling
LPF	Leak Path Factor
MACCS	MELCOR Accident Consequence Code System
MAR	Material at Risk
MOI	Maximally Exposed Offsite Individual
MORT	Management Oversight Risk Tree
NI	New Information
OSHA	Occupational Safety and Health Administration
PHA	Preliminary Hazard Analysis
PHR	Process Hazard Review
PRA	Probabilistic Risk Assessment
rem	roentgen equivalent man
RF	Respirable Fraction
S/RID	Standard/Requirement Identification Document
SAR	Safety Analysis Report
SIRIM	Site Item Reportability Management
SRS	Savannah River Site
SSC	Structure, System, and Component
TSR	Technical Safety Requirement
WSRC	Westinghouse Savannah River Company

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3.0 HAZARD AND ACCIDENT ANALYSIS

3.1 INTRODUCTION

This chapter identifies and assesses the potential hazards associated with Savannah River Site (SRS) facility operations, analyzes accidents, and then determines the sets of controls required to protect the general public. This chapter is organized in accordance with Department of Energy (DOE) Order 5480.23 and DOE-STD-3009-94, which implements DOE Order 5480.23 (Ref. 1, 2). A Hazard Analysis (HA), performed for the facilities, is the basis for identification of Safety Significant Structures, Systems, and Components (SSCs) and administrative controls to protect the onsite worker. A Design Basis Accident (DBA) analysis, performed using the results of this HA, is the basis for identification of Safety Class SSCs and administrative controls to protect the offsite public and onsite facility workers.

3.1.1 HAZARD ANALYSIS

DOE Orders require that a HA be performed for SRS facilities to characterize the hazards associated with operation of the individual facilities. The HA performs the following functions:

- Provides the basis for hazard classification of the facility
- Identifies and assesses the hazards that are present within the facility
- Evaluates the potential for hazards to develop into accidents
- Identifies the lines of defense within the facility that form the basis for Defense-in-Depth (DID) against adverse consequences to the workers and public from accidents

SSCs that make up lines of defense are considered in the analysis as candidates for Safety Significant items. Safety Significant SSCs and items requiring Technical Safety Requirement (TSR) controls are identified.

Additionally, the HA postulates bounding accident scenarios resulting from these hazards, evaluates their frequencies of occurrence and consequences in a qualitative, conservative manner. These scenarios are binned into one of twelve risk categories according to the frequency of occurrence and the severity of consequence. Analyses that are more rigorous are performed for accidents with the potential to subject the public to unacceptable combinations of frequency and consequence.

3.1.2 ACCIDENT ANALYSIS

The accidents selected from the HA with potentially significant consequences to the offsite public are considered to be DBAs. The DBAs also include Natural Phenomena Hazards and external events.

For each DBA, a detailed scenario is defined, and the offsite radiological, or chemical, consequences are analyzed quantitatively, using conservative assumptions regarding failure of SSC boundaries that could prevent or mitigate the release, as well as conservative treatments of the quantity and nature of the released material. Offsite consequences are compared to Evaluation Guidelines (EGs). If the EGs are exceeded, mitigative, or preventive SSCs, or administrative controls are credited (and subsequently controlled) as necessary, to reduce the frequency and consequence levels such that the risk is below the EGs. These SSCs and administrative controls are identified as Safety Class items and are subject to TSR controls. If the EGs are not exceeded, then further analysis, taking credit for additional SSCs to reduce the frequency and offsite consequences further, is not performed.

The DBA, although quantitative in nature, is performed using a graded approach -- one that employs a level of detail adequate to demonstrate that EGs for consequences are met with a sufficient margin to depict safety in an appropriately conservative fashion. Thus, the magnitude of the analysis effort and the resultant level of detail in the presentation of accidents are dependent upon the particular DBA, and are generally directly proportional to the perceived contribution of that accident to facility risk.

3.1.3 CHAPTER ORGANIZATION

Section 3.2 identifies the design codes, standards, regulations, and orders that are used in the establishment of the safety basis.

Included in Section 3.3 are presentations of the HA methodology, including hazard classification and evaluation, DID, worker safety, environmental protection, and accident evaluation.

Section 3.4 presents the methodology for the accident analyses.

3.1.4 SUMMARY OF RESULTS

The facility-specific Safety Analysis Report (SAR) chapters include the results of hazard identification, Hazard Classification (HC), Hazard Evaluation (HE), and accident analysis. Items discussed include the following:

- Identification of hazardous energy/material sources present
- Radiological and chemical inventories
- Facility HC
- Risk evaluation of identified accident scenarios based on a qualitative or semi-quantitative assessment of consequences and frequency
- Summary of design and operational preventive and mitigative features
- Identification of planned design and operational safety improvements

- Summary of DID, including identification of Safety Class or Safety Significant SSCs and other items needing TSR coverage
- Summary of significant worker safety features, including identification of Safety Significant SSCs and any relevant TSR administrative controls
- Identification of a limited set of unique and representative accidents to be analyzed further
- Accident analysis of DBAs
- Accident analysis of Beyond DBAs
- Comparison of consequences to EGs

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3.2 REQUIREMENTS

Standard/Requirement Identification Documents (S/RIDs) state the codes, standards, and regulations governing the hazard and accident analysis elements of SRS (Ref. 3). Programmatic compliance assessment has been performed against the S/RIDs and documented as specified in the Westinghouse Savannah River Company (WSRC) Procedure Manual 8B (Ref. 4). The Standards Management/Compliance Section maintains records of the programmatic compliance assessments.

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3.3 HAZARD ANALYSIS

This section describes the performance of a HA. The HA is the initial analytical effort and systematically presents an analysis of potential process-related, natural phenomena, and external hazards that can affect the public, the workers, and the environment due to singular, or multiple failures. This analysis considers the potential for both equipment failure and human error.

The HA provides a thorough, qualitative evaluation of the spectrum of risks to the public and the workers due to accidents involving identified hazards. It consists of three basic analytical activities: HC, Hazard Identification, and HE. DOE-STD-3009-94 requires the following:

- HA comprehensively identify potential events, event initiators, and dominant scenarios
- Estimate their frequencies and consequences
- Identify prevention and mitigation features
- Present the results in a risk matrix (Ref. 2)

Gross estimates of consequences and frequencies are performed in the HA such that attention is focused on those scenarios that are of greatest concern (i.e., highest risk).

3.3.1 HAZARD ANALYSIS METHODOLOGY

This section presents the methodology used to identify and characterize hazards and to perform a systematic evaluation of basic accidents. In addition, the methodology used to determine the facility HC is also presented.

3.3.1.1 Hazard Classification

DOE Order 5480.23 defines three HCs (Ref. 1). A facility is designated as HC 1 if the analysis shows the potential for significant offsite consequences. The analysis for an HC 2 facility shows the potential for significant onsite consequences. An HC 3 shows the potential for significant localized consequences only. A facility that does not exceed the HC 3 threshold criteria but still possesses some amount of radioactive material is considered a Radiological Facility.

Examples of engineered features or administrative controls are those specific facility features (not including site location), such as building confinement, elevated exhaust ventilation stacks, equipment, systems, actions, or operating conditions that are established to control risk. This approach allows credit to be taken for existing location and the physics of the dispersion of hazardous material releases, but not for containment, confinement, shielding, protection systems, administrative controls, or human activities.

- Hazard Classification 1
  - The facility has the potential for significant offsite consequences based on total curie content, potential material forms, and maximum energy for dispersion available. Only Category A Reactors or facilities designated by the Program Secretarial Officer are designated as HC 1 in accordance with the directions of DOE-STD-1027-92 (Ref. 5).

- Hazard Classification 2
  - The quantity of any radionuclide exceeds the HC 2 threshold quantity provided in Table A.1 of DOE-STD-1027-92 (Ref. 5). This quantity is based on 1 roentgen equivalent man (rem) at a distance of 100 meters (onsite) as determined in 10 CFR 30 and as modified by DOE (Ref. 5,6).
  - The minimum critical mass limit for any fissile material, as specified in ANSI/ANS-8.1, is exceeded (Ref. 5,7).
  - The total quantity of mixed fission products, where the individual radionuclides have been determined, is greater than 1000 curies (Ref. 5).
  - Where there are combinations of radioactive materials, the sum of the ratios of the quantity of each radionuclide to the HC 2 threshold quantity exceeds one (Ref. 5).

- Hazard Classification 3
  - The quantity of any radionuclide exceeds the HC 3 threshold quantity provided in Table A.1 of DOE-STD-1027-92 (Ref. 5). This quantity is based on 10 rem at 30 meters, based on a 24-hour exposure, except for tritium, which is reduced from the calculated value to 16,000 curies based on a recommendation from the Tritium Focus Group.
  - Where there are combinations of radioactive materials, the sum of the ratios of the quantity of each radionuclide to the HC 3 threshold quantity exceeds one (Ref. 5).

3.3.1.2 Hazard Identification

Hazard Identification is a comprehensive, systematic process by which all known hazards (hazardous materials and energy) associated with the facility are identified, recorded, and screened by a team of individuals representing the stakeholder organizations.

Hazards are primarily identified by developing comprehensive lists of all potential hazardous energy/material sources for the specific facility. Information for identifying the hazardous energy/material sources is obtained from applicable safety documentation. These sources typically include SARs, Bases for Interim Operation (BIOs), Preliminary Hazard Analyses (PHAs), Process Hazard Reviews (PHRs), design drawings and reviews, facility walkdowns, facility-operating history reviews, and consultations with facility personnel.

Following the identification of hazardous energy/material sources, each facility is systematically evaluated to determine the sources applicable to that facility. Each potential hazardous

energy/material source is evaluated for each facility. This process provides general information for the identification of hazardous energy and hazardous material sources and the facilities in which these sources exist.

HAZARD IDENTIFICATION IS DIVIDED INTO THREE STEPS: (1) DIVISION OF THE FACILITY INTO "SEGMENTS," (2) FACILITY WALKDOWNS, AND (3) SCREENING FOR COMMON HAZARDS.

3.3.1.2.1 DIVISION OF THE FACILITY

The facility is divided into "segments" to facilitate hazard identification and evaluation. These segments may be individual unit operations, individual or grouped facility systems, specific function(s), and/or physical boundaries inside the facility. A single, general hazard facility-segment that includes common cause events, which could involve more than one facility-section (e.g., facility fire and earthquake), is also identified.

3.3.1.2.2 FACILITY WALKDOWNS

Facility walkdowns include both physical walkdowns and information (i.e., paper) walkdowns to identify hazardous materials and energy sources for each facility section. Physical walkdowns permit the HA team to familiarize themselves, first-hand, with actual facility systems, processes, practices, equipment, and inventory.

The paper walkdown includes a review of the facility description, inventory, existing safety documentation (e.g., PHRs, SARs, BIOs, Operational Safety Requirements, Technical Standards, Project Design Documents, Fire Hazard Analysis), the SRS Fault Tree Data Banks, and/or consultations with facility system and/or process experts.

A hazards checklist (see Table 3.3-1) records generic facility hazards and is a useful aid in conducting Hazard Identification. This checklist is based on the DOE Management Oversight Risk Tree (MORT) methodology (Ref. 8). Hazard Identification Tables, used to document the results of Hazard Identification, are a useful "checklist" when performing physical, or paper walkdowns. Hazard Identification Tables are filled out for each facility section.

3.3.1.2.3 SCREENING OF COMMON HAZARDS

The Hazard Identification process provides the information required to perform both radiological and chemical HEs. As part of this methodology, inventory, location, segmentation and existing applicable controls are documented and analyzed. This process also identifies standard industrial hazards and routinely accepted hazards. However, in accordance with DOE-STD-3009-94, HEs of industrial hazards and routinely accepted hazards are not included (Ref. 2). Standard industrial hazards and routinely accepted hazards are identified only to the degree that they are initiators and contributors to events that result in radiological and chemical hazards. The

following characteristics are used to determine hazards that are standard industrial hazards and routinely accepted hazards:

- The hazard is routinely encountered first-hand by the general public in the home, home workshop, or public areas.
- Public consensus standards exist to control the hazard.
- No evidence exists that there are public or employee concerns about the hazard beyond normal prudence.
- The hazard is subject to Occupational Safety and Health Act (OSHA) regulations.

Protection against industrial and routinely accepted hazards is provided by practicing basic safety in the workplace. SRS has committed to practices and requirements defined in the S/RID [3]. Such hazards are formally and systematically treated by the following programmatic elements:

- WSRC Procedure Manual 8Q defines basic sitewide safety policies and minimum requirements (Ref. 9). This procedure manual is augmented by detailed rules and procedures developed by departments and facilities for activities within their areas of responsibility and require compliance with DOE Orders and OSHA regulations, at a minimum, for industrial safety.
- Industrial safety involves the detection, mitigation, management, and prevention of workplace hazards to protect against accidental death, injury, property damage, or interruption of production. The operating philosophy at SRS is that the safety and health of employees is the first and utmost priority. Policies are implemented at the facility level through site-level or facility-specific procedures.
- During the design phase of SRS facilities, or modifications to existing facilities, various design reviews contribute to industrial Hazard Identification. Application of the MORT process and information from past operating reports also contributes to the hazards identification process (Ref. 8).
- During facility operation, several programs ensure the timely identification of industrial hazards. These programs include OSHA compliance reviews, routine safety audits, periodic safety inspections, incident investigations (formal reviews and assessments of any unsafe situation or incident), annual safety program reviews, monthly safety meetings, safety suggestion programs, and the SRS Quality Assurance program.

3.3.1.3 Hazard Evaluation

The HE constitutes the primary focal point of the HA. The HE considers the detailed information that allows the development of specific events and scenarios associated with a hazardous release and the estimation of their frequency and consequences. The HE is performed in accordance with the requirements of the S/RID [13], which includes DOE-STD-3009-94, Standard 1027 and Standard 5502, as well as WSRC Procedure Manual 11Q (Ref. 2, X, X, 10).

The scope of the HE includes the following:

- All aspects of facility process and modes of operation including startup, shutdown, production, and routine maintenance that are considered part of operations
- Natural phenomena (e.g., earthquakes, tornadoes, and straight-line winds), external events (e.g., aircraft and vehicular impact), and nuclear criticality (where applicable)
- The entire spectrum of possible events for a given hazard in terms of both frequency and consequence levels
- Hazards addressed by other programs and regulations (e.g., Process Safety Management, OSHA, Resource Conservation and Recovery Act, Department of Transportation, and Environmental Protection Agency) only if loss of control of the hazard will result in a release

The scope of the HE does not include the following:

- Willful acts, such as sabotage
- Hazards routinely accepted by the public (These are screened during the standard industrial hazard screening process prior to conducting the HE.)

Using the results of Hazard Identification as a basis, detailed information is considered regarding hazardous material and energy sources in the context of facility section and whole facility operations to begin developing specific release events. Event categorization, identification of event cause(s), assignment of event frequency and consequence level, identification of mitigative and preventive features, screening for common hazardous events, and risk binning are tasks conducted during the HE.

Information related to the HE is collected and organized in HE Tables. The tables are a useful guide for performing a HE, and provide an effective format for documenting HE results. The team produces a separate HE Table for each facility section. Information in these tables is organized into the following categories:

- Event Number
- Event Category
- Postulated Event Description
- Causes
- Preventive Features
- Initiating Event Frequency Level
- Method of Detection
- Mitigative Features
- Unmitigated Consequence Level

- Risk Bin Number

Additional details and pertinent methodology information regarding each of the HE Table categories are provided in the following sections.

In the HE process, certain inherent assumptions are made that apply to each facility. First, facility segmentation, as defined in the hazard categorization, is credited. The concept of facility segmentation is applied where facility features preclude bringing material together, or causing harmful interaction from common severe phenomena.

In addition to facility segmentation, the HE also assumes that the radionuclide and chemical inventories used to determine the facility hazard category are the bounding values for each facility. Guidance in the 11Q Manual [10] directs exclusion of sealed sources under certain conditions. The de-minimus concept is applied where permitted under the guidance stated in standard 5502 [X]. Finally, the HE assumes that facility access is controlled such that the population exposed to a given potential event is limited.

3.3.1.3.1 EVENT NUMBER

Events are numbered to provide each with a sequential reference.

3.3.1.3.2 EVENT CATEGORY

Events are categorized according to the nature of the postulated release mechanism. A standard list of event categories, based on those given in Appendix E of DOE/TIC-11603, is used (Ref. 11). They are as follows:

- E-1 Fire
- E-2 Explosion
- E-3 Loss of Containment/Confinement
- E-4 Direct Radiological/Chemical Exposure
- E-5 Nuclear Criticality
- E-6 External Hazards
- E-7 Natural Phenomena

Events are categorized according to the event description rather than the event cause. For example, a facility fire might be a postulated event that is caused by an earthquake, or some other natural phenomenon. This event would fall under Category E-1 (Fire) rather than E-7 (Natural Phenomena). Table 3.3-2 gives some additional information regarding event categories and associated hazardous material and energy sources.

3.3.1.3.3 POSTULATED EVENT DESCRIPTION

A brief description of a postulated event is given in this column of the HE Tables. The event description clearly defines the nature of the event. It includes the following types of events

- Its location; hazard source
- Affected system(s) or equipment
- Any interaction with other facility section(s), system(s), equipment, and/or hazards
- Any pertinent operating characteristics

Using the Hazard Identification Tables as a basis, the event scenarios are developed for each facility section where a potential exists for a release of hazardous energy and/or material. The scenarios cover the entire spectrum of possible events for a given hazard, from small consequence events, for which procedures or equipment is acknowledged to provide adequate protection, to reasonable worst-case conditions. Unlike "worst-case," "reasonable worst-case" does not necessarily consider every parameter in its most unfavorable state.

3.3.1.3.4 CAUSES

A cause specifically states the failure, error, operational, and/or environmental condition that initiated the release event of the postulated event. Causes are synonymous with initiating events and, therefore, need to be clearly identified to support frequency evaluation. The Hazard Identification Tables are used as a guide in developing specific causes for release events.

3.3.1.3.5 PREVENTIVE FEATURES

A preventive feature is any feature that could be expected to act to prevent the release of hazardous material to an unwanted location, thus reducing the frequency of the associated release event. The selection of preventative features is made without regard to any possible pedigree of the feature such as procurement level or current functional classification. Preventive features might include engineered features (e.g., SSCs), administrative controls (e.g., procedures, policies, and programs), natural phenomena (e.g., ambient conditions, buoyancy, and gravity), or inherent features (e.g., physical or chemical properties, location, and elevation) operating individually or in combination. Preventive features are those that are assumed operable and available prior to an event, and are not required to be operable during the event or post-event.

Preventive features constitute a significant portion of DID evaluation, and they provide essential input to the Functional Classification task. Therefore, the identification effort captures essentially all of the possible features that could be counted on to prevent the release of the hazardous material to an unwanted location.

3.3.1.3.6 INITIATING EVENT FREQUENCY LEVEL

Event frequency evaluation is a qualitative, or semi-quantitative, process that involves assigning a frequency level to each initiating event in the HE Tables. Frequency levels and descriptions are summarized in the Frequency Evaluation Levels (see Table 3.3-3), which are based on DOE-STD-3009-94 (Ref. 2).

A determination of the appropriate frequency level for a particular event is made based on the event's root cause(s) and may be either qualitative or quantitative. Sources of frequency information include generic initiator database, existing safety documentation, engineering calculations, failure rate data, and facility expert opinion. Uncertainties in frequency levels are accommodated by erring in the conservative direction from best-estimate values. The initiating event frequency level is recorded in the HE Tables in accordance with the Table 3.3-3 lettering scheme.

When evaluating event frequency, credit may be taken for the inventory controls limiting the amount of hazardous material in the facility, or facility section. Credit may be taken for the ability of workers to react to obvious hazardous conditions and evacuate; this, of course, invokes the assumptions that the workers are physically able to evacuate, and that an evacuation route is available immediately following the hazardous condition. Crediting active SSCs, or Administrative Controls for preventive properties is discouraged. Any preventive feature, or assumption, credited during consequence determination must be listed and highlighted in the Hazard Identification Tables for Functional Classification purposes.

3.3.1.3.7 METHOD OF DETECTION

Methods of detection include features that are designed to detect initiating events or subsequent event progression. These include alarms, monitors, indicators, and the operator's ability to recognize the events by visual observation, or sound.

3.3.1.3.8 MITIGATIVE FEATURES

Mitigative features are any features that are expected to act to reduce the consequences associated with the release of hazardous material to an unwanted location for a particular event. The identification of such features is made without regard to any possible pedigree of the feature such as procurement level or current classification. Mitigative features are those which are assumed to be operable during an event or post-event. Therefore, mitigative features must be capable of withstanding the environment of the event. These might include engineered features (e.g., SSCs), administrative controls (e.g., procedures, policies, and programs), natural phenomena (e.g., ambient conditions, buoyancy, and gravity), or inherent features (e.g., physical or chemical properties, location, and elevation) operating individually, or in combination.

Mitigative features constitute a significant portion of DID evaluation and provide essential input to the Functional Classification task. Therefore, it is important that the identification effort capture essentially all of the possible features that could be counted on to reduce the consequences of a release of the hazardous material to an unwanted location.

3.3.1.3.9 UNMITIGATED CONSEQUENCE LEVEL

Event consequences are documented by specifying the impact on the receptors (described below) and the guideline used to assess the impact. Any potential impact of consequences on other systems is also documented in this column of the HE Tables.

For HA purposes, consequences are defined as the unmitigated dose or exposure to specified receptors. Consequences are a function of the type and characteristics of the hazard, the quantity released, the release mechanism, relative location of the release, and any relevant transport characteristics. Consequences are determined from (1) simple source term calculations, (2) existing safety documentation, and/or (3) qualitative assessment. Engineering judgement, expertise, and knowledge of facility hazards is used to select one or more of the above methods, as appropriate, for consequence determination. Unlike frequency levels, which cover two orders of magnitude, consequence levels sometimes span less than one order of magnitude. Thus, a more refined effort may be required to determine the appropriate consequence level for a given event and receptor. Much like frequency EGs, uncertainties in consequences are accommodated by erring in the conservative direction, especially for those events with consequences at the high end of a given level.

Consequences are evaluated at various receptor locations to assess health and environmental effects of the postulated release. Tables 3.3-4 and 3.3-5 give the radiological and chemical consequence levels for the specified receptor locations. Receptors are offsite, onsite 1, and onsite 2. Offsite receptors are the public, or everyone outside the site boundary. The onsite 1 receptor is the facility worker considered to be at the scene of the accident. The onsite 2 receptor is the worker at the worst case location outside the facility boundary. The HA is concerned with the Maximally Exposed Offsite Individual (MOI).

The HE Tables should provide the impact of the event on the three receptors for each of the postulated release events. This information is documented in the column labeled "Impact on Receptors" under the "Unmitigated Consequence Level" heading of the HE Tables.

To lend completeness to the HA task, the HA team considers any potential impact on other systems and lists these in the HE Tables in the "Impact on Other System" columns under the "Unmitigated Consequence Level" heading.

3.3.1.3.10 RISK BIN NUMBER

Using event frequency and consequence levels, events are "binned" in frequency-consequence space to assess relative risk. The objective of risk binning is to focus attention on those events that pose the greatest risk to the public and the onsite receptors. Higher risk events might be candidates for additional analysis and/or Functional Classification evaluation.

Tables 3.3-6, -7, and -8 are risk-binning matrices for the three receptor locations considered in HA (i.e., offsite, onsite 1, and onsite 2). In each of these tables, bins are defined by a rectangular matrix in frequency-consequence space. Each bin is uniquely numbered, but numbering is for

identification purposes only (i.e., risk severity is not proportional to the magnitude of the bin label).

Table 3.3-6 is the risk-binning matrix for offsite receptors. The darker shaded bins (i.e., 1, 2, 3, 4, 5, and 7) represent risks that exceed the offsite radiological and/or chemical EGs for Functional Classification. Events falling into these bins typically require further evaluation by a Functional Classification engineer as candidates for Safety Class functions. In DOE-STD-3009-94 terminology, these events are considered "unique," or "situations of major concern," with sufficiently high risk that individual examination is needed by accident analysis (Ref. 2).

The three lighter shaded bins in Table 3.3-6 (i.e., 6, 8, and 9) fall below the Functional Classification EGs, yet per Reference 2, these events are considered "situations of concern" that yield a subset of "representative" events needing further examination. Representative events bound a number of similar events of lesser risk (i.e., the worst fire for a number of similar fires). At least one event from each of the event types (i.e., fires, and explosions) is considered representative; however, representative events are examined only to the extent that they are not bounded by unique events.

Table 3.3-7 is the risk-binning matrix for the onsite 1 receptor(s), located anywhere inside the facility with the hazardous release or hazardous condition. The darker shaded bins (i.e., 1, 2, 4, 5, and 7) represent risks that exceed the onsite radiological and/or chemical EGs for Functional Classification. Events falling into these bins typically require further evaluation by a Functional Classification engineer as candidates for Safety Significant functions.

Table 3.3-8 is the risk-binning matrix for the onsite 2 receptor(s), located outside the facility boundary. The darker shaded bins (i.e., 1, 2, 4, 5, and 7) represent risks that exceed the onsite radiological and/or chemical EGs for Functional Classification. Events falling into these bins typically require further evaluation by a Functional Classification engineer as candidates for Safety Significant functions.

The HE Tables provides, for each of the postulated release events, a bin number representing risk at each receptor location.

3.3.2 HAZARD ANALYSIS RESULTS

As discussed in Section 3.3.1, the HA consists of three basic analytical activities: HC, Hazard Identification, and HE. This section provides an in-depth discussion of the results from the performance of these activities.

3.3.2.1 Hazard Classification

The criteria for determining the radiological Hazard Category of facilities are provided in DOE-STD-1027-92 (Ref. 5).

The analysis conducted to determine the facility Hazard Category is performed without credit taken for engineered features or administrative controls. The radiological evaluation is accomplished by comparing the inventory of each radionuclide to the Threshold Quantities provided in Table A.1 of DOE-STD-1027-92 (Ref. 5). The criteria for determining the Hazard Categories of nuclear facilities are then completed in accordance with the methodology in Section 3.3.1.

3.3.2.2 Hazard Identification

As stated in Section 3.3.1.1, hazards are systematically identified by listing hazardous materials, energy sources, and their locations in tables to ensure completeness. Screening was performed to eliminate material/energy types and quantities that are considered "common hazards." Hazard Identification was divided into three steps: (1) division of the facility into "sections," (2) facility walkdowns, and (3) screening for common hazards. This process included the following primary elements:

- Applicable safety documentation was reviewed to identify the hazardous energy/material sources associated with the specific facility. These documents included existing SARs, Hazard Categorization, the existing BIOs, and supporting PHA.
- Selected procedures were reviewed.
- The facility Fault Tree Data Bank (FDB) was reviewed.
- Discussions with facility personnel were held, and information was incorporated into this evaluation.
- Facility "sections" were identified.
- Facility walkdowns were performed to verify information from existing documentation and to identify additional information on recent facility modifications.
- Screening of standard industrial hazards was performed.

The results of performing these activities are documented in the HE Tables. These results identify the hazardous energy/material sources applicable to the specific facility. During the Hazard Identification process, potential hazardous energy/material sources for the facility were identified, and each facility was evaluated to determine if these energy sources and/or material sources actually existed.

3.3.2.2.1 REVIEW OF PAST RELEVANT OPERATING HISTORY

The following sources are utilized for data collection on the past operational history of SRS facilities:

- Fault Tree Data Bank: The FDB documents safety-related incidents (e.g., off-normal, unusual, routine, and emergency) that have occurred throughout the operating history

of the SRS. Although this database is not totally inclusive, it is regarded as the best source of information as far as historical occurrences are concerned.

- Site Item Reportability and Issue Management (SIRIM) database: The SIRIM program was established in January of 1991 to provide a system for reporting abnormal occurrences at SRS. The significance determination of abnormal occurrences is based on criteria provided in the SIRIM Handbook (Ref. 12).
- Savannah River Site New Information (NI) database: An NI report is written when a potential safety issue, which may or may not be of significance, requires further review. A NI report is distributed periodically and describes the outstanding NIs.
- Existing facility safety documentation: (e.g., SARs and the existing BIOs) summarize the accident initiators and abnormal occurrences that have resulted from facility operations.
- Discussions with cognizant facility engineers and operators

Data acquired during the review of historical occurrences are factored into the evaluations presented in the HE Tables for the facilities. Analysts use the knowledge gleaned from the review of historical occurrences for the qualitative appraisal of the frequency and consequence of events.

3.3.2.3 Hazard Evaluation

As discussed in Section 3.3.1, each event analyzed as a part of the HA process received a risk ranking. The specific accidents to be considered for further analysis, because of the HE, are described in the accident selection section for each facility segment. The following topics are discussed for each facility segment:

- Planned Design and Operational Safety Improvements
- DID
- Worker Safety
- Environmental Protection
- Accident Selection

The discussion of the Planned Design and Operational Safety Improvements for each facility segment summarizes the basis for committing to the improvement and, if needed, any interim controls proposed until implementation is complete.

DID, as an approach to facility safety, builds layers of defense against the release of hazardous materials so that no single layer comprises an entire system of protection. To compensate for potential facility failures, DID is based on several layers of protection with successive barriers to prevent the release of hazardous materials to the environment. Site level procedures and the E7 manual contain the methodology used at SRS for the implementation of DID. This approach includes measures to protect the public, workers, and the environment from harm in case any of

these barriers are not fully effective. Defining DID as it exists at a given facility is crucial for determining a safety basis. However, no requirement to demonstrate a particular number of layers of defense is imposed.

The methodology for identifying the DID features for the various facility segments involves a hazard screening process. The hazard screening process is similar to that used in the HE. However, the identification of DID features also involves an evaluation of the potential for adverse effects on the facility worker.

The purpose of the hazard screening process is to identify the potential events that result in a risk rank of 1, 2, 4, 5, or 7 for the two onsite receptors. Potential events falling within these bins are evaluated for appropriate DID features. Potential events falling in bins 3, 6 or 8 through 12 are eliminated from further consideration, and the identification of DID is not required for these events. Also, as discussed in Section 3.3.1.1, standard industrial hazards are eliminated from further consideration, and the identification of the DID feature is not required for these events.

The Worker Safety and Environmental Protection Sections summarize the major features protecting workers and the environment, respectively. General prioritization of the features and a brief discussion is required.

Accident analysis entails the formal quantification of a subset of accidents that define the safety envelope for the facility. These accidents are to represent "a complete set of bounding conditions" according to DOE Order 5480.23 (Ref. 1).

It is important to realize that the risk rank for each event is based upon the consequences of the postulated release.

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3.4 ACCIDENT ANALYSIS

This section documents the analysis of those accidents identified in the accident selection portion of Section 3.3 and the additional accidents identified by the facility for analysis. The accidents, frequencies, and doses to the MOI are summarized in the facility SAR.

3.4.1 METHODOLOGY

For each accident, the accident frequency and a bounding source term is calculated. The bounding source term is converted to an offsite dose using conversion factors from MELCOR Accident Consequence Code System (MACCS). For fire scenarios, the MACCS conversion factors are calculated without consideration for plume rise but assuming various release times depending on the scenario. The estimated dose for the MOI is compared to EGs for the appropriate frequency bin. If the bounding dose to the MOI is below the EGs, no further assessment of the accident scenario occurs.

In DOE Order 5480.23 and DOE-STD-3009-94 SARs, there is a significant difference in the treatment of offsite and onsite receptors. A bounding, deterministic analysis for the offsite receptor only credits Safety Class SSCs for preventing and/or mitigating the dose to the offsite receptor. An additional set of Safety Significant SSCs is available for preventing/mitigating the dose to the onsite receptors. Thus, the scenario by which the onsite receptor receives a dose and the scenario (and associated source term) by which the offsite receptor receives a dose may vary considerably depending on the specific Safety Significant preventers/mitigators credited in the analysis. The current DOE guidance manages the variation in the onsite and offsite accident scenarios by analyzing offsite impacts in Section 3.4 to determine Safety Class SSCs and basing the selection of Safety Significant SSCs on the qualitative HA in Section 3.3 and specific, as needed, analyses documented in Chapter 4 of the SAR during Functional Classification (Ref. 5).

3.4.1.1 Source Term Analysis

The source term is the amount of hazardous material (radioactive or toxicological) released to the air. Radioactive material is reported in grams or curies while non-radioactive toxic material is reported in grams or pounds. The initial source term is the amount of hazardous material driven airborne at the accident source (Ref. 13).

The airborne pathway is of primary interest for nonreactor nuclear facilities. DOE-STD-1027-92 quotes observations of the Nuclear Regulatory Commission 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" (Ref. 5). The airborne source term is typically estimated by the following five-component linear equation:

$$\text{Source Term} = \text{MAR} \times \text{DR} \times \text{ARF} \times \text{RF} \times \text{LPF} \qquad \text{(Eq. 3.4-1)}$$

where:

MAR	=	Material-at-Risk (Curies, grams, pounds)
DR	=	Damage Ratio
ARF	=	Airborne Release Fraction (or Airborne Release Rate for continuous release)
RF	=	Respirable Fraction
LPF	=	Leak Path Factor

3.4.1.1.1 MATERIAL-AT-RISK

The MAR is the amount of hazardous material 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 hazardous material 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 release may involve only the contents of a single confinement vessel. Conversely, a seismic event may involve all of the material in a building (Ref. 13).

3.4.1.1.2 DAMAGE RATIO

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, analysts may exclude this material from the MAR.

The DR is estimated based upon engineering analysis of the response of structural materials and materials of construction for confinement 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 conservatism due to simplification of phenomena to obtain a usable model, but the purpose of the approximation is to obtain, to the degree possible, a realistic understanding of potential effects (Ref. 13).

3.4.1.1.3 AIRBORNE RELEASE FRACTION

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 a physical stress 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 upon 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 1 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 (Ref. 13).

3.4.1.1.4 RESPIRABLE FRACTION

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 Aerodynamic Equivalent Diameter and less (Ref. 13).

RFs for particles made airborne under accident-induced stresses are dependent upon a variety of factors, such as the bulk density (i.e., how well the powder at rest compacts), the presence of moisture, how effectively the type and level of stress deagglomerates the powder or subdivides the solid/liquid, the efficiency with which the stress suspends the powder/fragments of solid over varying size ranges, and the degree of immediate proximity of surfaces on which airborne particles may impact/settle (Ref. 13).

3.4.1.1.5 LEAK PATH FACTOR

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 leakpaths are involved, their cumulative effect is often expressed as one value based upon (1) established relationships between size of the particulate material, airborne transport mechanisms, and losses by deposition mechanisms, or (2) specified filtration efficiencies (Ref. 13).

3.4.1.2 Consequence Analyses

Consequences of releases of radionuclides under postulated accident conditions are evaluated with Version 1.5.11.1 of the MACCS code (Ref. 14).

MACCS was developed by Sandia National Laboratories for the Nuclear Regulatory Commission for use in commercial reactor Probabilistic Risk Assessment (PRA) studies (Ref. 14). More recently, MACCS has been used in various DOE facility PRAs and baseline safety documents. Included are PRAs for K and N Reactors, compliance with SEN-35-91 safety goals for facility Building 233-H, red oil consequence analyses, and proposed fusion sitting evaluations (Ref. 14).

MACCS predicts dispersion of radionuclides by the use of multiple, straight-line Gaussian plumes. Although each plume treats the released material as a neutrally buoyant gas, the direction, duration, sensible heat, and initial radionuclide concentration may be varied from plume to plume. Crosswind dispersion is treated by a multi-step function and both wet and dry

deposition features can be modeled as independent processes. Meteorological variability is treated in MACCS with a stratified random sampling algorithm. MACCS uses the Latin Hypercube Sampling (LHS) mode of 1 year of site-specific meteorological data to analyze under the random sampling option. Based on the LHS distribution and application of user-specified dose and/or health effects models, Complementary Cumulative Distribution Functions are calculated for various measures of consequence. The average, median, 95th, and 99.5th percentile doses are provided in the output (Ref. 14).

For regulatory applications, MACCS is used to calculate the 50-year Committed Effective Dose Equivalent (CEDE) to specified stationary receptors from the plume passage phase of a hypothetical release[1]. This dose estimate is reported in the SAR, BIO, and HA applications. The CEDE is calculated for onsite and offsite receptors using standard uptake assumptions and dose conversion database values. Sensitivity studies may also be performed with MACCS to show the relative benefits of evacuation, sheltering, interdiction, and the effects of various shielding assumptions (Ref. 14).

For SARs, the code is executed through the second of three modules. Air and ground concentrations, plume size, and timing information for all plume segments as a function of downwind distance are calculated in the first module. The second module calculates the consequences due to exposure to radiation during the emergency phase (first 7 days) of the event. Only the direct exposure pathways are considered (cloudshine, groundshine, and inhalation). A third module, for the consequences due to exposure to radiation subsequent to the emergency phase of the postulated accident and for computing decontamination and other economic impacts incurred because of the accident, is not applied. The third module is typically applied to probabilistic safety and environmental impact analyses, where both direct and indirect exposure pathways are considered (groundshine, inhalation, and contaminated water and food ingestion) (Ref. 14).

3.4.1.3 Evaluation Criteria

Certain aspects of evaluation criteria used in the overall hazard and accident analysis process are presented in Section 3.3.1 and in Section 3.3.2.3, which addresses worker safety and accident selection.

The HA used the hazard binning matrix shown in Table 3.3-6.

According to the qualitative nature of the process, the severity and frequency of each hazard was estimated to within two orders of magnitude, based on experience and knowledge of the team performing the evaluation. Severity of consequences was also estimated qualitatively, based on inventories.

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[1] In most non-reactor, non-criticality releases, the Committed Effective Dose Equivalent is equivalent to the Total Effective Dose Equivalent.

The HA used the following frequency ranges for the evaluation of events, as shown in Table 3.3-6. For the evaluation of offsite radiological and chemical consequences, the HA used the following quantitative guidelines to assign qualitative consequence ranges:

Consequence	Radiological (at site boundary)	Chemical (at site boundary)
(H) High	> 25 rem	> ERPG-2
(M) Moderate	> 5 rem and ≤ 25 rem	> ERPG-1 and ≤ ERPG-2
(L) Low	> 0.5 rem and ≤ 5 rem	> PEL-TWA and ≤ ERPG-1
(N) Negligible	≤ 0.5 rem	≤ PEL-TWA

The accident analysis for each DBA below includes a comparison of predicted bounding offsite consequences of hazardous material releases from postulated accidents to specific criteria (termed EGs). It is necessary to examine the postulated accident sequences for candidate Safety Class functions. Each sequence should credit a set of equipment such that the risk is below the offsite EGs. For Safety Class functions, the consequence of concern is the exposure of the offsite individual to hazardous materials. Each analyzed accident or event requires the determination of a discrete initiating event frequency and a discrete consequence. The methodology used for Functional Classification, ensuring that the Safety Class function is maintained, allows credit to be taken for items that are designated Safety Class, which are those SSCs or administrative controls necessary to prevent or mitigate the release of hazardous material to the offsite public that would exceed the EGs (Ref. 15). Each accident was reviewed against these guidelines (see Figure 3.4-1) to identify the need for any Safety Class items.

As shown in Figure 3.4-1, the EGs use different frequency and severity criteria. The guidelines are based on the following frequency ranges:

Description	Frequency Range (events/year)
Expected	f > 1E-1
Anticipated	1E-1 ≥ f > 1E-2
Unlikely	1E-2 ≥ f > 1E-4
Extremely Unlikely	1E-4 ≥ f > 1E-6
Beyond Extremely Unlikely	1E-6 ≥ f

The consequence EGs for the MOI are as follows:

DBAs shall not cause a dose to the MOI exceeding the following:

- 0.5 rem (radiological) and Permissible Exposure Limit-Time Weighted Average (hazardous material), or less than 2E-6 increase in the Incremental Cancer Risk (ICR) (hazardous material) for accidents with estimated frequencies ≤ 1.0E-01 events/year but > 1.0E-02 events/year

- 5 rem (radiological) and Emergency Response Planning Guide (ERPG)-1 (hazardous material), or less than 2E-4 ICR (hazardous material) for accidents with estimated frequencies < 1.0E-02 events/year but ≥ 1.0E-04 events/year
- 25 rem (radiological), ERPG-2 (hazardous material), or less than 2E-2 ICR (hazardous material) for accidents with estimated frequencies < 1.0E-04 events/year but ≥ 1.0E-06 events/year

The radiological dose is based on the 50-year CEDE.

3.4.2 DESIGN BASIS ACCIDENTS

As discussed in Section 3.4.1, the qualitative frequencies of accidents selected in Section 3.3 are further analyzed to verify that the selected accidents were in the correct frequency bin. The facility SARs provide a discussion of the accident analysis performed for the selected accidents that were found credible or Beyond DBA.

3.5 REFERENCES

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11. Nonreactor Nuclear Facilities: Standards and Criteria Guide. DOE/TIC-11603, Rev. 1, Science Applications International Corp., La Jolla, CA, September 1986.

12. Site Item Reportability and Issue Management Handbook (U). WSRC-RP-93-75, Rev. 1, Westinghouse Savannah River Company, Aiken, SC, July 1993.

13. Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities. DOE-HDBK-3010-94, U.S. Department of Energy, Washington, DC, October 1994.

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15. Functional Classification Methodology Manual (U). WSRC-TM-93-9, Rev. 3, Westinghouse Savannah River Company, Aiken, SC, March 1995.

Table 3.3-1 Hazards Checklist

Group	Hazard Energy Source	Hazard Energy Source
Electrical	Battery banks Cable runs Diesel generators Electrical equipment Hot plates Heaters High voltage Locomotive, Electrical Motors	Pumps Power tools Switchgear Service outlets, fittings Transformers Transmission lines Underground wiring Wiring
Thermal	Bunsen burner/Hot plates Cryogenic liquids Electrical equipment Furnaces Heaters Steam lines Welding torch	Boilers Lasers Electrical wiring Welding surfaces Engine exhaust Exothermic reaction
Kinetic – Linear and Rotational (Friction)	Belts Bearings Fans Gears Motors Presses Grinders Crane Loads (in motion) Power tools	Vehicles Rail cars Fork lifts Carts Dollies Centrifuges Drills Saws Shears
Pyrophoric Material	Plutonium and Uranium metal	plutonium
Spontaneous Combustion	Nitric acid and organics Grease Diesel fuel	Paint solvents Cleaning/Decon solvents Gasoline
Open Flame	Bunsen burners	Welding/cutting flames
Flammables	Flammable gases Flammable liquids Natural Gas Spay paint Gasoline	Compressed flammable gases Propane Paint solvent Cleaning/decon solvents
Combustibles	Combustible materials Plastics	Paper/wood products Petroleum based products
Chemical Reactions	Uncontrolled chemical reactions	
Potential (pressure)	Gas bottles Gas receivers Pressure vessels Steam headers and lines Coiled springs	Boilers Heated surge tanks Autoclaves Furnaces Stressed members

Table 3.3-1 Hazards Checklist (continued)

Group	Hazard Energy Source	Hazard Energy Source
Potential (height/mass)	Stairs Lifts Cranes Elevated doors Loading docks Hoists Elevators	Trucks Jacks Scaffolds and Ladders Pits Elevated work Surfaces Mezzanines
Explosive/Pyrophoric Material	Explosive gases Explosive chemicals Hydrogen Dynamite Sodium Hydrogen (batteries) Primer cord Propane	Dusts Nitrates Peroxides Caps Plutonium/Uranium Potassium Electric squibs Superoxides
Radiological Material Hazardous Material	Radiological Material Alkali Metals Asphyxiants Biologicals Carcinogens Corrosives Acetone Fluorides Lead Oxidizers Asphyxiation Drowning Other toxics	Cryogenic liquids Ammonia and compounds Beryllium and compounds Chlorine and compounds Trichlorethylene Decontamination solutions Dusts and particles Sandblasting particles Metal plating Herbicides Insecticides Bacteria Viruses
Ionizing Radiation Sources	Fissile material Radiography equipment Radioactive material Radioactive sources	Electron beams X-ray machines Critical masses Contamination
Nuclear Criticality	Fissile Material	Fissionable Material
Non-facility Events	Explosion Fire Other	Power Outage Aircraft crash Transportation accident
Vehicles in Motion	Airplane Helicopter Train	Forklifts Truck/Car Heavy construction equipment
Crane	Crane	Crane loads
Natural Phenomena	Straight wind Tornado Earthquake Flood	Lightning Rain/hail Snow, freezing weather

Table 3.3-2 Event Categories and Relationship to Hazard Energy and Material Sources

Event Category	Event Category Description	Hazard Energy and Material Groups
E-1	Fire	Electrical Thermal Friction Pyrophoric material Spontaneous combustion Open flame Flammables Combustibles Chemical Reactions
E-2	Explosion	Potential (pressure) Explosive materials Chemical Reactions Kinetic
E-3	Loss of Containment or Confinement	Radiological Material Hazardous Material
E-4	Direct Radiological/Chemical Exposure	Ionizing radiation sources Non-ionizing radiation sources
E-5	Nuclear Criticality	Fissile Material
E-6	External Hazards	Non-facility Events Vehicles in Motion Crane
E-7	Natural Phenomena	Natural Phenomena

Table 3.3-3 Frequency Evaluation Levels

Event Frequency Code	Description	Estimated Annual Frequency of Occurrence ($year^{-1}$)
Anticipated (A)	Accidents that may occur several times during the life cycle of the facility (accidents that commonly occur).	$f \geq 10^{-2}$
Unlikely (U)	Accidents that are not anticipated to occur during the life cycle of the facility. Natural phenomena of this probability class include the following: Uniform Building Code-level earthquake, 100-year flood, maximum wind gust, etc.	$10^{-2} > f \geq 10^{-4}$
Extremely Unlikely (EU)	Accidents that will probably not occur during the life cycle of the facility. This class includes the design basis accidents.	$10^{-4} > f \geq 10^{-6}$
Beyond Extremely Unlikely (BEU)	All other accidents.	$f < 10^{-6}$

Table 3.3-4 Radiological Consequence Evaluation Levels for Hazard Receptors

Consequence Level (Abbreviation) ↓	Receptor (considered location): Offsite	Onsite 1 (Inside Facility)	Onsite 2 (Outside Facility)
High (H)	≥ 25.0 rem	Radiological material quantity exceeds Hazard Category 3 threshold (per DOE-STD-1027), or prompt worker fatality, acute injury that is immediately life threatening, or permanently disabling	≥ 100.0 rem, or prompt worker fatality, acute injury that is immediately life threatening, or permanently disabling
Moderate (M)	5.0 ≤ C < 25.0 rem	Radiological material quantity exceeds Hazard Category 3 threshold (per DOE-STD-1027), or serious injury, no immediate loss of life, no permanent disabilities, hospitalization required	25.0 ≤ C < 100.0 rem, or serious injury, no immediate loss of life, no permanent disabilities, hospitalization required
Low (L)	0.5 ≤ C <5.0 rem	Radiological material quantity less than Hazard Category 3 threshold (per DOE-STD-1027), or Minor injuries, no hospitalization	5.0 ≤ C < 25.0 rem, or minor injuries, no hospitalization
Negligible (N)	< 0.5 rem	Minor injuries, no hospitalization	< 5.0 rem, or minor injuries, no hospitalization

Table 3.3-5 Chemical Consequence Evaluation Levels for Hazard Receptors

Consequence Level (Abbreviation) ↓	Receptor (considered location): Offsite	Onsite 1 (Inside Facility)	Onsite 2 (Outside Facility)
High (H)	≥ ERPG-2	Uniform distribution of total release exceeds ERPG-3, or prompt worker fatality, acute injury that is immediately life threatening ,or permanently disabling	≥ ERPG-3, or prompt worker fatality, acute injury that is immediately life threatening ,or permanently disabling
Moderate (M)	ERPG-1 ≤ C < ERPG-2	Uniform distribution of total release exceeds ERPG-3, or serious injury, no immediate loss of life, no permanent disabilities, hospitalization required	ERPG-2 ≤ C < ERPG-3, or serious injury, no immediate loss of life, no permanent disabilities, hospitalization required
Low (L)	PEL-TWA ≤C < ERPG-1	Uniform distribution of total release ERPG-1 to ERPG-3, or minor injuries, no hospitalization	ERPG-1 ≤ C < ERPG-2, or minor injuries, no hospitalization
Negligible (N)	< PEL-TWA	Uniform distribution of total release < ERPG-1, or minor injuries, no hospitalization	< ERPG-1, or minor injuries, no hospitalization

Table 3.3-6 Risk Binning Matrix in Frequency-Consequence Space - Offsite Receptors

Frequency → / Consequence ↓	**Beyond Extremely Unlikely** $< 10^{-6}$ /yr	**Extremely Unlikely** $10^{-6} \leq f < 10^{-4}$ /yr	**Unlikely** $10^{-4} \leq f < 10^{-2}$ /yr	**Anticipated** $\geq 10^{-2}$ /yr
High Radiological ≥ 25 rem Chemical ≥ ERPG-2	**10**	**7**	**4**	**1**
Moderate Radiological 5 ≤ C < 25 rem Chemical ERPG-1 ≤ C < ERPG-2		**8**	**5**	**2**
Low Radiological 0.5 ≤ C < 5 rem Chemical PEL-TWA ≤ C < ERPG-1		**9**	**6**	**3**
Negligible Radiological < 0.5 rem Chemical < PEL-TWA	**11**	**12**		

Key:

- **Risk Bins 1, 2, 3, 4, 5, 7**
  * exceed Functional Classification evaluation guidelines for offsite receptors
  * "unique" events, individual examination needed

- **Risk Bins 6, 8, 9**
  * "representative" events, examined to the extent that they are not bounded by unique events, at least one bounding event from each event category (fires, explosions, etc.)

Table 3.3-7 Risk Binning Matrix in Frequency-Consequence Space - Onsite 1 Receptor (at the scene of the accident inside facility)

Frequency → Consequence ↓	**Beyond Extremely Unlikely** $< 10^{-6}$/yr	**Extremely Unlikely** $10^{-6} \leq f < 10^{-4}$/yr	**Unlikely** $10^{-4} \leq f < 10^{-2}$/yr	**Anticipated** $\geq 10^{-2}$/yr
High (multiple workers) Radiological radiological material quantity exceeds Hazard Category 3 threshold (per DOE-STD-1027) Chemical: uniform distribution of total release exceeds ERPG-3 Any Hazard: prompt worker fatality, acute injury that is immediately life threatening, or permanently disabling	**10**	7	4	1
Moderate (single worker) Radiological: radiological material quantity exceeds Hazard Category 3 threshold (per DOE-STD-1027) Chemical: uniform distribution of total release exceeds ERPG-3 Any Hazard: prompt worker fatality, acute injury that is immediately life threatening, or permanently disabling		8	5	2
Low Radiological: radiological material quantity less than Hazard Category 3 threshold (per DOE-STD-1027) Chemical: uniform distribution of total release ERPG-1 ≤ C < ERPG-3 Any Hazard: serious injury, no immediate loss of life, no permanent disabilities, hospitalization required		9	6	3
Negligible Radiological: NA Chemical: uniform distribution of total release < ERPG-1 Any Hazard: minor injuries, no hospitalization	**11**	**12**		

Key:

♦ **Risk Bins 1, 2, 4, 5, 7**

* exceed Functional Classification EGs for onsite receptors

Table 3.3-8 Risk Binning Matrix in Frequency-Consequence Space - Onsite 2 Receptor (outside facility)

Frequency → Consequence ↓	**Beyond Extremely Unlikely** $< 10^{-6}$ /yr	**Extremely Unlikely** $10^{-6} \leq f < 10^{-4}$ /yr	**Unlikely** $10^{-4} \leq f < 10^{-2}$ /yr	**Anticipated** $\geq 10^{-2}$ /yr
High Radiological: ≥ 100 rem Chemical: ≥ ERPG-3 Any Hazard: prompt worker fatality, acute injury that is immediately life threatening or permanently disabling		7	4	1
Moderate Radiological: 25 ≤ C < 100 rem Chemical: ERPG-2 ≤ C < ERPG-3 Any Hazard: prompt worker fatality, acute injury that is immediately life threatening or permanently disabling	10	8	5	2
Low Radiological: 5 ≤ C < 25 rem Chemical: ERPG-1 ≤ C < ERPG-2 Any Hazard: serious injury, no immediate loss of life, no permanent disabilities, hospitalization required		9	6	3
Negligible Radiological: < 5 rem Chemical: < ERPG-1 Any Hazard: minor injuries, no hospitalization	11	12		

Key:

♦ **Risk Bins 1, 2, 4, 5, 7**

* exceed Functional Classification EGs for onsite receptors

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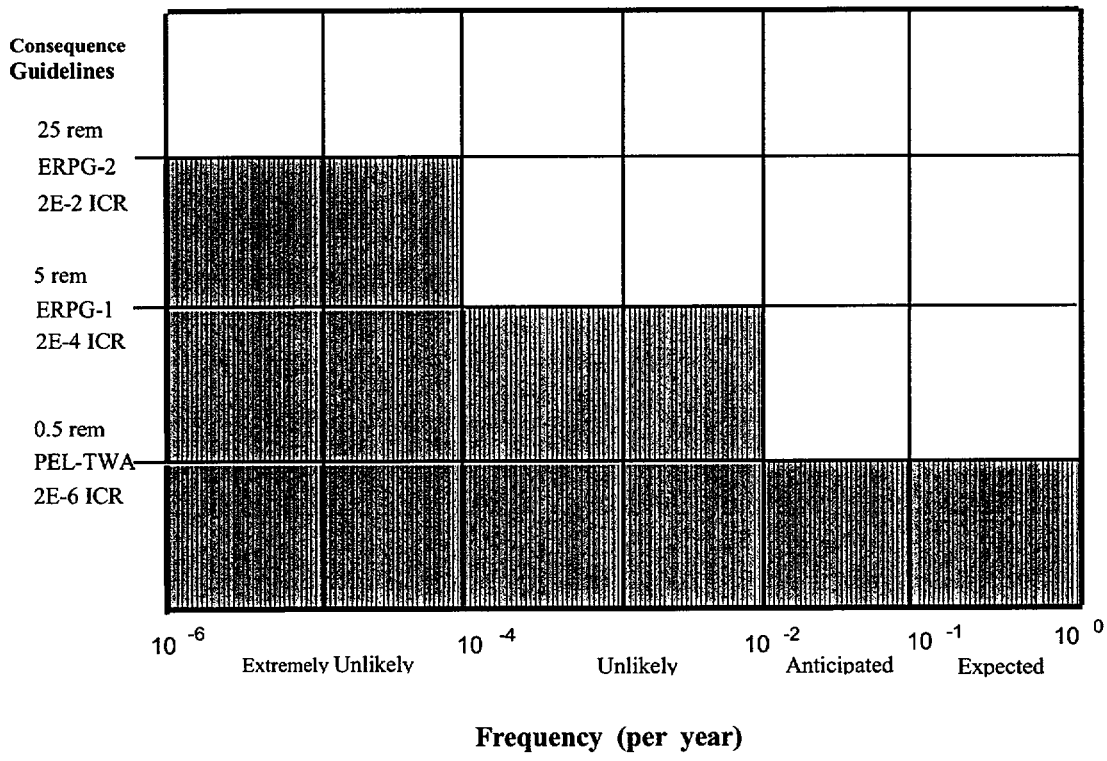

Figure 3.4-1 WSRC Offsite Evaluation Guidelines (From E7 Procedure 2.25, Rev. 3)

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Rev. 4 DOE Review Copy

GENERIC SAFETY ANALYSIS REPORT

CHAPTER 6

PREVENTION OF INADVERTENT CRITICALITY

September 1999

Westinghouse Savannah River Company
Aiken, SC 29808

SAFETY • RESPONSIBILITY • SECURITY
SRS
SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

DISCLAIMER

CONTENTS

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

ACRONYMS AND ABBREVIATIONS

AB	Authorization Basis
ANS	American Nuclear Society
ANSI	American National Standards Institute
CSC	Criticality Safety Committee
CSF	Criticality Safety Function
DCA	Double Contingency Analysis
DOE	Department of Energy
ESH&QA	Environment, Safety, Health, and Quality Assurance
FEB	Facility Evaluation Board
GSAR	Generic Safety Analysis Report
HLWM	High-Level Waste Management
MP	Management Policy
NCSASR	Nuclear Criticality Safety Analysis Summary Report
NCSE	Nuclear Criticality Safety Evaluation
NCSRC	Nuclear Criticality Safety Review Committee
NCSS	Nuclear Criticality Safety Supplement
NIM	Nuclear Incident Monitor
NMSS	Nuclear Materials Stabilization and Storage
NSDS	Nuclear Safety Data Sheet
ORR	Operational Readiness Review
OSR	Operational Safety Requirements
PE&CD	Projects Engineering and Construction Division
RBOF	Receiving Basin for Offsite Fuels
S&HO	Safety and Health Operations
S/RID	Standards/Requirements Identification Document
SAR	Safety Analysis Report
SFS	Spent Fuel Storage
SFSD	Spent Fuel Storage Division
SRS	Savannah River Site
SRTC	Savannah River Technology Center
SW	Solid Waste
TSR	Technical Safety Requirement
WSMS	Westinghouse Safety Management Solutions, LLC
WSRC	Westinghouse Savannah River Company

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6.0 PREVENTION OF INADVERTENT CRITICALITY

6.1 INTRODUCTION

Westinghouse Savannah River Company (WSRC) Manual WSRC-1-01, Management Policy (MP) 4.5 establishes the inadvertent criticality prevention program for the Savannah River Site (SRS) (Ref. 1). MP 4.5 states that this nuclear criticality safety program is established, maintained, and applied to any process, structure, system, or component that requires the control of one, or more, parameters for criticality safety purposes. The features of the inadvertent criticality prevention program are described in WSRC-SCD-3 (Ref. 2). The nuclear criticality safety program described in WSRC-SCD-3 satisfies the Department of Energy (DOE) requirements as referenced in Standards/Requirements Identification Documents (S/RIDs) (Ref. 3).

6.1.1 APPLICABILITY

The purpose of this chapter of the Generic Safety Analysis Report (GSAR) for DOE nuclear facilities and operations at SRS is to provide information that satisfies DOE requirements for Safety Analysis Reports (SARs) as referenced in S/RIDs. This chapter describes the essential features of the inadvertent criticality prevention program as it relates to facility safety at SRS and serves as the basis for the development of detailed facility-specific SAR chapters dealing with criticality safety issues. However, this chapter is not to be used as the vehicle for review and approval of the nuclear criticality safety program.

6.1.2 OBJECTIVES

This chapter describes the inadvertent criticality prevention provisions for SRS nuclear facilities and their operations. This chapter summarizes the SRS inadvertent criticality prevention program as established by WSRC-1-01, MP 4.5 and as implemented through WSRC-SCD-3 (Ref. 1, 2). The products of this chapter, as delineated in DOE-STD-3009-94, are as follows (Ref. 4):

- A summary of the overall SRS inadvertent criticality prevention policy and program
- A description of the general basis and analytical approach used for deriving operational criticality limits
- A summary of the design and administrative controls used by the inadvertent criticality prevention program

6.1.3 SCOPE

The following four hazard categories are defined for classifying facilities that contain inventories of radioactive materials and are listed in order of decreasing hazard severity (Ref. 5):

- Hazard Category 1
- Hazard Category 2
- Hazard Category 3
- Radiological

All facilities classified as at least Hazard Category 3 are required to have a documented SAR.

Hazard Category 1 and 2 facilities can contain inventories of fissionable materials sufficient to present an inadvertent criticality hazard, whereas Hazard Category 3 facilities, by definition, do not. Therefore, this chapter is only applicable to Hazard Category 1 and 2 facilities that contain sufficient fissionable materials to present a criticality hazard.

6.2 REQUIREMENTS

S/RIDs state the codes, standards, and regulations governing the inadvertent criticality prevention policies and program elements of the SRS (Ref. 3). Programmatic compliance assessment has been performed against the S/RIDs and documented as specified in the WSRC Procedure Manual8B (Ref. 6). The Standards Management/Compliance Section maintains records of the programmatic compliance assessments.

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6.3 DESCRIPTION OF POTENTIAL CRITICALITY HAZARDS

6.3.1 DESCRIPTION OF STRUCTURES, SYSTEMS, AND COMPONENTS POTENTIALLY CONTAINING FISSIONABLE MATERIALS AND THEIR OPERATIONS

SRS contains numerous facilities that contain, or process, fissionable material, including the following:

- Separations facilities (200-Areas)
- Waste handling and disposal facilities (E Area)
- Experimental and analytical facilities (Savannah River Technology Center [SRTC] and the Analytical Laboratories)
- Fuel storage facilities (Receiving Basin for Offsite Fuel [RBOF], L-Basin, K-Basin, K-Assembly Area)

In addition, several facilities have been shutdown and substantially de-inventoried of fissionable material, but still contain fissionable material in ductwork, or on internal glovebox surfaces (e.g., 200-Area facilities, 247-F).

The processes, structures, systems, and components potentially containing fissionable materials and their operations are described in the facility-specific SARs.

6.3.2 FISSIONABLE MATERIAL INVENTORIES, ENRICHMENTS, FORMS, AND LOCATIONS

Fissionable materials of primary concern at SRS are U-235 and Pu-239; however, Pu-238 and Np-237 are also present in substantial quantities, and other isotopes are present in small quantities, depending on the involved processes. The forms of fissionable materials at SRS include metallic forms of pure elements, alloys with aluminum fabricated fuel forms, hydrogenous solutions, and oxide forms. Uranium enrichment varies from depleted (0.2 wt. % U-235 in uranium) to highly enriched (93 wt. % U-235 in uranium). Plutonium isotopic composition varies from high Pu-239 content (from irradiated reactor fuel) to high Pu-238 content (from heat source materials) for National Aeronautics and Space Administration missions.

Information on inventories of fissionable materials, including enrichment, form, and location, in the processes, structures, systems, or components of the facility is provided in the facility-specific SARs.

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6.4 CRITICALITY CONTROLS

Criticality control is achieved with process designs and operations that ensure that criticality safety parameters are maintained within safe ranges. The three basic means of control are passive engineered control, active engineered control, and administrative control. Passive engineered control is preferred over active engineered control, and active engineered control is preferred over administrative control. WSRC-SCD-3 provides details about each means of control (Ref. 2).

6.4.1 ENGINEERING CONTROLS

The two means of engineered controls are defined as follows:

- Passive engineered controls are fixed design features, or devices, that rely on natural forces, or properties of material to control, or prevent, a nuclear criticality. These controls do not require human intervention, or the activation or movement of a device by electrical, mechanical, hydraulic, or other means.
- Active engineered controls are devices that require activation, or movement, by electrical, mechanical, hydraulic, or other means to control, or prevent a nuclear criticality. These devices sense parameters important to criticality safety and automatically initiate action to secure the system to a safe condition without human intervention.

The facility-specific SARs provide information on engineered controls for criticality accident prevention. The facility-specific SARs discuss the criticality safety design limits on engineered controls (passive or active), the bases that support their application, and any design criteria used to ensure subcritical configurations under all normal, abnormal, and accident conditions. This facility-specific discussion addresses the application, as appropriate, of engineered means, as applied to the several criticality safety control parameters. Examples of passive engineered controls include fixed geometry, or fixed spacing. An example of an active engineered control is a neutron detector, which monitors solution concentrations and is hardwired to an interlock to stop flow if the concentration of fissile material exceeds a preset value.

6.4.2 ADMINISTRATIVE CONTROLS

Administrative controls require personnel interactions with the process to verify that the process safety parameters remain within specified limits. Administrative controls are generally steps in procedures (including surveillance programs). Such controls include the following:

- Sampling and reviewing sample results
- Recording and reviewing material transfers
- Calculating totals of material transfers

- Verifying material types and quantities introduced to a system

The facility-specific SARs provide information on administrative controls for criticality accident prevention. The facility-specific SARs discuss, as applicable, administrative controls on nuclear material safety limits such as administrative concentration control, and procedures for handling, storing, and transporting fissionable materials. Administrative controls may involve action, caution, or verification steps in a procedure, or steps in a surveillance program that rely on the judgment, training, and responsibility of personnel for implementation.

Section 6.5 discusses administrative controls for the review and approval of changes to process, or system, configurations.

6.4.3 CRITICALITY SAFETY PARAMETERS

6.4.3.1 Geometry

Geometry control involves the use of dimension and shape restrictions on equipment to provide "geometrically safe," or "geometrically favorable," containers, vessels, drains, sumps, etc., for fissionable materials, or restrictions on fluid flow preventing fissionable solutions from assuming an unsafe geometry. All dimensions and nuclear properties upon which reliance is placed shall be verified prior to beginning operation and continuing control shall be exercised over such properties and dimensions.

The facility-specific SARs discuss, as appropriate, the following items that involve the application of geometry controls and the factors used in the design of these controls:

- Reliance on limits in equipment dimensions including manufacturing tolerances
- Limitation on types of equipment, process flow rates, process volumes, and other process variables
- Reliance on maintenance, surveillances, or inspections for detection of corrosion, distortions, erosion, or changes in equipment and instrumentation tolerances
- Prevention of carryover, or inadvertent transfer of, fissionable materials from geometrically favorable vessels to vessels, or systems, that are not geometrically favorable
- Prevention of the accumulation of fissionable materials into heating, or cooling, jackets of a system resulting from leakage
- Limitation on, or prevention of, the accumulation of fissionable materials in floor sumps and drains
- Reliance on equipment design to maintain safe dimensions during operation to ensure a safe geometrical configuration

6.4.3.2 Spacing (Interaction) Control

Spacing or interaction control involves the use of distance, arrangement, and shielding (neutronic isolation) restrictions between, and among units, vessels, containers, equipment, and accumulations of fissionable materials to minimize the potential for neutron interaction of fissionable materials.

The facility-specific SARs discuss, as appropriate, the following items that involve the application of spacing, or interaction controls and the factors used in the design of these controls:

- Limitation on spacing between units, vessels, containers, and equipment containing fissionable materials that are stored or moved such that the entire array (or batch) will remain subcritical under all credible conditions that might affect the array (or batch)
- Reliance on the integrity of storage racks to maintain spacing control during and following design basis accidents

6.4.3.3 Neutron Absorber (Poison) Control

Neutron absorber (poison) control involves the use of solid or soluble neutron absorbers in vessels, sumps, etc., to reduce the neutron interaction of fissionable material should it accumulate in such areas.

The facility-specific SARs discuss, as appropriate, the following items that involve the application of neutron absorber controls and the factors used in the design of these controls:

- Reliance on neutron-absorbing materials, such as cadmium, boron, and gadolinium, in equipment and processes to prevent a nuclear criticality based on reliable and suitable data
- Reliance on borosilicate-glass Raschig rings for packed vessels to prevent a nuclear criticality in accordance with American National Standards Institute (ANSI)/American Nuclear Society (ANS)-8.5-1986 (Ref. 7)
- Limitation on the minimum soluble neutron (poison) absorber concentration allowing a sufficient margin of safety to exist should all other variables be at their worst-case, (credible) or Limiting Condition for Operation values
- Reliance on two independent sample analyses, or methods, of determining the concentration of a soluble neutron poison to confirm that the poison concentration limit is satisfied
- Reliance on reliable and timely monitoring of the presence of soluble neutron poison to provide automatic, or operator-initiated protective action
- Reliance on samples of corrosion coupons to represent the actual presence of fixed neutron poison in a system

6.4.3.4 Concentration (Density) Control

Fissionable material concentration, or density control, involves the use of restrictions on items such as the following:

- Permitted concentrations of fissionable materials dissolved, or dispersed, in another medium
- Density of fissionable material powder, metal chips, and machine turnings
- Allowable chemical compounds or the physical state for fissionable materials at particular process stages, workstations, and storage areas
- Allowed fissionable mass per unit area (e.g., a floor, or the bottom, of a glovebox) to prevent a nuclear criticality

The facility-specific SARs discuss, as appropriate, the following items that involve the application of fissionable material concentration, or density controls and the factors used in the design of these controls:

- Assurance that vessels relying solely on fissionable material concentration control for criticality safety do not contain concentrations of fissionable isotopes in uniform aqueous solutions that exceed the single-parameter limit concentration for aqueous solutions reflected by an infinite thickness of water
- Reliance on two independent sample analyses or methods of determining the concentration of fissionable material to confirm that the concentration limit is satisfied
- Reliance on periodic monitoring of tanks containing fissionable solutions for evaporation and the addition of a makeup solution as necessary to prevent unsafe concentrations

6.4.3.5 Moderation and Reflection Control

Moderation and reflection control involves the use of restrictions on items such as the following:

- The allowed range of hydrogenous material density relative to fissionable material density in moderator/fissionable material mixtures (i.e., H/X ratio) or on the total amount of moderating materials allowed
- The quantity, composition, and configuration of hydrogenous or other neutron-reflecting materials, in proximity to fissionable material, to prevent a nuclear criticality

The facility-specific SARs discuss, as appropriate, the following items that involve the application of moderation and reflection controls and the factors used in the design of these controls:

- Assurance that, for operations dependent upon control of neutron moderation for criticality safety, the prescribed extent of moderation remains unchanged, or that, if a credible change occurs, the reactivity of the system remains below acceptable subcritical limits
- Consideration of interstitial moderation whenever such moderation is credible
- Consideration of the presence, or introduction of, moderators, or reflectors, more effective than water
- Consideration of firefighting materials (particularly water, or other moderators) and limitations or guidelines for their use
- Limitation on the moderation and reflection based on the worst credible reflection possible during normal operating, or credible accident conditions
- Reflection controls based on limiting personnel access to a system should not be relied on for criticality safety without evaluation on a case by case basis
- Limitation on the addition of fissionable and moderating material to otherwise geometrically favorable enclosures, or areas, by material transport and transfer systems

6.4.3.6 Mass and Volume Control

Fissionable material mass and volume control involves the use of restrictions on items such as the following:

- Quantity of fissionable material permitted in an individual unit, or in an area, room, or rooms (i.e., a mass control zone)
- Total number of fissionable material units
- Fissionable material volume, container volume, or vessel volume (may be specific to fissionable material composition) to prevent a nuclear criticality

The facility-specific SARs discuss, as appropriate, the following items that involve the application of fissionable material mass and volume controls and the factors used in the design of these controls:

- Consideration of the possibility of overbatching when operations depend upon fissionable material mass controls or when the contained volume does not automatically limit the contents of a vessel, or unit, to a safe fissionable material mass, or less
- Consideration of minimum safety margins in areas where overbatching is credible

- Consideration of overbatching for the storage of fissionable material
- Consideration of the change of fissionable material concentration to a more reactive state in a solution due to polymerization, precipitation, or dilution
- Limitation on the fissionable material volume based on the minimum volume required to sustain a neutron chain reaction for a given geometric shape under the worst credible process conditions that may exist within the system, including consideration of system interactions with other process systems and the environment

6.4.3.7 Enrichment Control

Fissionable material enrichment control to prevent a nuclear criticality involves the use of restrictions on the maximum fraction of fissile isotopes (usually expressed as weight percent) for a fissionable element, such as uranium, or plutonium, that has both fissionable and fissile isotopes.

The facility-specific SARs discuss, as appropriate, the application of fissionable material enrichment controls and the consideration of the most reactive isotopic composition that is credible in the determination of criticality safety limits.

6.4.4 APPLICATION OF DOUBLE-CONTINGENCY PRINCIPLE

The double-contingency principle is stated as follows (Ref. 8):

> "Process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. Protection shall be provided by either (i) the control of two independent process parameters (which is the preferred approach, when practical, to prevent common mode failure), or (ii) a system of multiple controls on a single process parameter. The number of controls required upon a single controlled process parameter shall be based upon control reliability and any features that mitigate the consequences of control failure. In all cases, no single credible event or failure shall result in the potential for a criticality accident, except as referenced in the paragraph that follows.
>
> An exception to the application of double contingency, where single-contingency operations are permissible, is presented in paragraph 5.1 of ANSI/ANSI-8.10-1983, R88. This exception applies to operations with shielding and confinement (e.g., hot cells or other shielded facilities).
>
> Double contingency shall be demonstrated by documented evaluations."

The facility-specific SARs discuss how the double-contingency principle is met and reference detailed Nuclear Criticality Safety Evaluations (NCSEs) as necessary. As appropriate, the

discussion includes, but is not limited to, the following items abridged from WSRC-SCD-3 (Ref. 2):

- A double-contingency analysis shall be included in, or referenced by, the nuclear criticality safety evaluation which establishes limits/controls for new or modified processes, pieces of equipment, storage, and transport involving fissionable material consistent with DOE-STD-3007-93 (Ref. 9). This analysis may be qualitative in approach.
- If double-contingency defenses are not independent, areas of overlap are identified (e.g., common power supplies, common methods of calibration, and common components), and steps are taken to remove common mode failure dependencies to the extent practical.
- Passive and active engineered controls and administrative controls associated with a double-contingency defense are identified, as well as the Technical Safety Requirements (TSRs) related to the controls.

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6.5 CRITICALITY PROTECTION PROGRAM

Nuclear criticality safety is administered at SRS by assigning responsibilities for key nuclear criticality safety requirements and activities to appropriate WSRC organizational units.

6.5.1 CRITICALITY SAFETY ORGANIZATION

This section discusses the WSRC organizational units having responsibilities in the implementation of the site nuclear criticality safety program specified in WSRC-SCD-3 (Ref. 2).

6.5.1.1 Westinghouse Savannah River Company President Authority and Responsibility

The WSRC President establishes the company-level policy for implementing the criticality safety requirements of DOE and, in doing so, informs all WSRC employees involved in operations with fissionable materials of the criticality safety requirements.

The WSRC President accepts overall responsibility for the criticality safety of operations but delegates the authority and assigns the responsibility for day-to-day criticality safety of operations to lower-level management.

The WSRC President establishes a system of WSRC committees whose purpose is as follows:

- Provide guidance and consultation to facilities
- Work with facilities on criticality safety questions and issues
- Assist in identifying potential improvements in equipment and operations
- Promote the site nuclear criticality safety program

6.5.1.2 Westinghouse Savannah River Company Division General Manager and/or Chief Engineer Authority and Responsibility

The WSRC organizations whose division general managers and/or chief engineers have authority and responsibility, including training, with respect to criticality safety.

The division general managers and/or chief engineers of the WSRC organizations listed above have authority and responsibility over the following activities abridged from WSRC-SCD-3 (Ref. 2):

- Accept and implement criticality safety responsibilities as delegated by the WSRC President.
- Accept the responsibility for criticality safety for facility operations under their control.

- Ensure that applicable criticality safety standards and DOE Orders are applied in the design, modification, and operation of facilities under their control.
- Ensure that the WSRC policy for nuclear criticality safety, as stated in WSRC-1-01, MP 4.5 and as implemented in WSRC-SCD-3, is applied at the division level (Ref. 1, 2).
- Ensure that an interface or matrix arrangement exists with the Westinghouse Safety Management Solutions LLC (WSMS) affiliate, which works under control of the WSRC Site Chief Engineer, to provide necessary criticality safety expertise to facility activities and operations.
- Ensure that appropriate criticality training is developed and provided to the staff under their control.
- Delegate criticality safety responsibilities to lower-level managers within the division as necessary, but maintain overall accountability for such responsibilities at the general manager level.
- Maintain a division/area-level Criticality Safety Committee (CSC), as appropriate, and document the authority and responsibility of the committee in division/area manuals and/or committee charters.

The Environmental, Safety, Health, and Quality Assurance (ESH&QA) Division General Manager has the following responsibilities abridged from WSRC-SCD-3 (Ref. 2):

- Accept, and implement, criticality safety responsibilities as delegated by the WSRC President.
- Provide personnel with criticality safety experience to serve on area-level CSCs and to assist the Nuclear Criticality Safety Review Committee (NCSRC), as requested.
- Provide Safety and Health Operations (S&HO) and Safety and Health Programs Department functions for the calibration, use, and control of sources used to perform Nuclear Incident Monitor (NIM) field response checks, and for the calibration, use, and adequacy of dosimetry and field survey instrumentation used in conjunction with NIM evacuations.
- Include criticality safety among the functional areas to be addressed by Facility Evaluation Board (FEB) Reviews and Operational Readiness Reviews (ORRs), as appropriate.

The PE&CD General Manager and/or Chief Engineer has the following additional responsibilities abridged from WSRC-SCD-3 (Ref. 2):

- Comply with the nuclear criticality safety design requirements specified in applicable WSRC documents
- Administer the sitewide NIM program

- Ensure that the WSMS affiliate, working under control of the Site Chief Engineer, provides probabilistic risk assessments
- Provide personnel with criticality safety experience to serve on area-level CSCs and to assist the NCSRC, as requested
- Ensure that the WSMS affiliate, working under control of the Site Chief Engineer, provides criticality safety engineering services across the site (Section 6.5.1.6 addresses responsibilities of the WSMS affiliate)

The High-Level Waste Management (HLWM) General Manager and/or Chief Engineer has the following additional responsibilities abridged from WSRC-SCD-3 (Ref. 2):

- Review facility and process changes to determine if criticality safety controls are required, or if less than a significant quantity of fissionable material, is maintained
- Maintain an interface with the WSMS affiliate to obtain criticality safety engineering support, as necessary

These HLWM responsibilities apply to the Defense Waste Processing Facility.

6.5.1.3 Facility Managers

Facility managers, whose facilities warrant criticality safety consideration and controls, have the following responsibilities abridged from WSRC-SCD-3 (Ref. 2):

- Accept responsibility for the day-to-day criticality safety of their facility
- Accept responsibility and administer a program for NIM system management within the facility
- Accept responsibility for preparation, use, and maintenance of, as well as adherence to, facility procedures and drawings related to, or affecting, criticality safety
- Maintain a program of staff training in both general and facility-specific aspects of criticality safety, as appropriate
- Maintain a program of NIM familiarization and evacuation instruction for non-WSRC visitors and WSRC personnel needing access, but not normally assigned, to the facility
- Provide any additional criticality training deemed necessary for non-WSRC visitors and WSRC personnel needing access, but not normally assigned, to the facility
- Provide appropriately trained staff to determine when procedures, drawings, and design documents require criticality safety review, ensuring that such procedures, drawings, and design documents are forwarded to the WSMS affiliate for review

- Ensure that criticality safety is included in facility self-assessments and that related deficiencies, action items, corrective actions, and other findings are properly addressed
- Accept responsibility for compliance with applicable DOE Orders, WSRC requirements, and facility-specific authorization basis documentation
- Approve the response to action items noted by the CSC
- Ensure that criticality safety aspects of facility design, construction, and operation are covered by a documented safety analysis
- Ensure that the facility is covered by a documented double-contingency analysis as specified by the nuclear criticality safety program
- Accept responsibility for placing their facilities in a safe condition when warranted by actual, or indicated, unsafe criticality conditions
- Ensure that passive engineered, active engineered, and/or administrative criticality safety means of control are in place and functioning properly
- Accept responsibility for the development of a facility fire safety plan that recognizes, to the extent necessary, both fire safety and criticality safety considerations as specified by applicable WSRC documents and the nuclear criticality safety program
- Establish and maintain criticality safety postings for the facility and labeling of fissionable materials as specified by applicable WSRC documents and the nuclear criticality safety program
- Delegate criticality safety responsibilities to lower-level facility supervision, as necessary, still retaining overall responsibility for criticality safety in the facility
- Ensure that NIM evacuation routes are properly marked, provide for timely facility evacuation, ensure that facility changes do not unnecessarily impede, or otherwise lengthen, evacuation time, and that, to the extent possible, routes do not require personnel to approach potential sites of a criticality accident
- Provide monitoring and surveillance as necessary to detect and prevent accumulations of fissionable material in process equipment, storage areas, piping, and ventilation systems, taking corrective action with input from the Criticality Safety Function (CSF) when accumulations of fissionable material are detected
- Provide other criticality safety features and administrative controls as necessary to provide for the criticality safety of the facility, including facility access control

6.5.1.4 Facility Operators

Facility operators have the following responsibilities abridged from WSRC-SCD-3 (Ref. 2):

- Be trained in criticality safety commensurate with their duties, typically consisting of general and facility-specific training

- Accept responsibility for criticality safety of their own actions and the operating systems under their control
- Be familiar with, and follow, criticality safety procedures as written, and adhere to all nuclear criticality safety steps in operating procedures related to their assignments
- Stop performing a procedure, and notify supervision, if the procedure does not correspond to the operating system upon which it is being performed, or if a condition develops that is not addressed by procedures

6.5.1.5 First-Line Supervisors

First-line supervisors have the following responsibilities abridged from WSRC-SCD-3 (Ref. 2):

- Accept responsibility for the criticality safety of operations under their control
- Be knowledgeable in those aspects of criticality safety relevant to operations under their control
- Ensure that criticality safety training is administered to, and that the appropriate training records are maintained for, personnel under their control, and require that these personnel have an understanding of procedures and safety considerations necessary to perform their functions without undue risk
- Develop or participate in the development of written procedures applicable to the operations under their control, and be responsible for the maintenance and update of those procedures
- Verify compliance with criticality safety specifications for new, or modified, equipment prior to its use
- Require conformance with good safety practices including unambiguous identification of fissile materials and good housekeeping
- Obtain training and assistance, as necessary, from the CSF

6.5.1.6 Criticality Safety Personnel of the Westinghouse Safety Management Solutions Affiliate

Criticality Safety personnel have the following responsibilities abridged from WSRC-SCD-3 (Ref. 2):

- Provide for independent reviews (i.e., technical, or design reviews in accordance with WSRC Procedure Manual E7 [Ref. 10]) of NCSEs and nuclear safety engineering calculations prepared within the CSF
- Review all NCSEs prepared by other groups
- Review and approve all formal documents that establish criticality safety limits

- Certify, validate, and provide to criticality safety engineers controlled neutronics codes (e.g., k_{eff} and kinetics calculation) and data sets for use on their own computer systems, and/or maintain a set of such codes on centralized site computer systems, including established bias and bias uncertainties (or data by which bias and bias uncertainties can be constructed)
- Certify, validate, and provide to criticality safety engineers neutron/gamma transport (i.e., shielding) codes and associated data sets
- Qualify criticality safety engineers and others, as requested, in the use of neutronics codes and neutron/gamma transport codes
- Perform NIM location analyses, and document the results as required by the nuclear criticality safety program
- Maintain familiarity with current developments in nuclear criticality safety standards, guides, and codes, consulting with knowledgeable individuals, as necessary, to obtain required technical information
- Initiate formal criticality safety documentation of recommended limits
- Provide assistance in the development, review, and concurrence of procedures and procedural changes affecting criticality safety.
- Review and concur with design drawings and drawing changes affecting criticality safety
- Provide technical assistance in the design of equipment and processes affecting criticality safety
- Review and concur with equipment and process changes affecting criticality safety
- Provide first-level independent evaluation of criticality safety control effectiveness
- Recommend to facility management that any operation that does not have the required level of criticality safety be brought to a safe condition
- Assist other divisions in the development of nuclear criticality safety programs, including manuals and procedures
- Comply with the site requirements for safety documentation as specified in WSRC Procedure Manual 11Q and as applicable to criticality safety documentation (Ref. 11)
- Maintain familiarity with all operations within organizations for which criticality safety support is provided and for which criticality safety controls are required
- Lead and/or participate in the division/area CSCs for the purpose of assessing criticality safety practices, providing in-process reviews and technical support, as necessary

- Review and approve, as well as provide requested technical information for general and facility-specific criticality safety training courses
- Review and concur with the fire safety plan for each facility in which moderation control is a concern
- Examine reports of procedural violations and other deficiencies for potential improvements of safety practices and procedural requirements, reporting potential improvements to management

6.5.1.7 Interfaces and Interrelationships with Other Organizations

The interfaces and interrelationships of the organizations specified above are defined in WSRC-1-01, MP 4.5 and WSRC-SCD-3 (Ref. 1, 2).

6.5.1.8 Staff Qualifications

Staff qualifications are specified in the facility-specific SARs, including consideration of educational levels, related relevant experience, job requirements, and other pertinent special skills that may be necessary.

6.5.1.9 Staff Levels

Minimum staffing requirements are specified in the facility-specific SARs, including consideration of the number of shifts for normal operation, types of job skills required for certain operations, and manning levels for emergency situations.

6.5.1.10 Criticality Safety Committees

CSCs monitor the site nuclear criticality safety program. This system of committees is described in WSRC-1-01, MP 4.5 and WSRC-SCD-3 (Ref. 1, 2). These WSRC documents establish committees for monitoring site criticality safety as follows:

- The NCSRC is sponsored by the Site Chief Engineer. The NCSRC is responsible for fostering and monitoring criticality safety across the site. Revisions to the site criticality safety program are approved by the NCSRC. The charter of this committee is established in WSRC-1-01, Charter 6.10 (Ref. 12).
- The CSCs, also designated as criticality review committees, report to the NCSRC. The CSCs are maintained in the 100 and 200 Areas and in the SRTC to monitor fissionable material handling, storage, processing, and disposal activities from a criticality safety perspective. The CSCs support criticality safety self-assessments; provide technical support for, and perform in-process reviews of, such activities on a periodic basis; and, as appropriate, report their findings to facility management and the NCSRC.

- The NIM Committee reports to the NCSRC. The NIM Committee reviews NIM design changes (in conjunction with the Chief Engineer, who is the design authority for the NIM unit), troubleshooting, maintenance, and coordinates configuration control of NIM location analysis.

The facility-specific SARs may provide more detail about the makeup and activities of the CSC responsible for the specific facility.

6.5.2 INCORPORATING CRITICALITY SAFETY IN PROCEDURES

Operations in a facility (handling or storing fissionable materials) are performed in accordance with approved procedures. Such procedures identify the appropriate criticality safety considerations. All expected activities are covered by procedures. Activities involving conditions not covered by a procedure are stopped (in a safe state) until procedures are written and approved to cover the unexpected conditions.

The following sections indicate general requirements abridged from WSRC-SCD-3 (Ref. 2).

6.5.2.1 Review of Operations

Operations, (including storage areas) in which criticality safety is a consideration, are governed by written procedures. The facility-specific SAR provides a summary description of the procedures applicable to the specific facility.

For facilities in which criticality safety is of concern, written procedures cover startup, operations, and any modifications that may affect criticality safety.

Procedures should clearly specify all parameters and limits related to criticality safety that is being controlled. All criticality safety related limits contained in operating procedures are based on NCSEs.

Operations are reviewed, at least annually, to ensure that procedures are being followed and that process conditions have not been altered.

No single inadvertent departure from a procedure should cause a criticality accident.

Procedures are supplemented by posted criticality safety limits, or other operator aids, (e.g., checklists, and flowsheets) as necessary.

Deviations from operating procedures and unforeseen alterations in process conditions that affect criticality safety are documented, reported to management, and investigated promptly. Actions are taken to prevent recurrence.

The facility-specific SARs specify the degree of operational review required for a specific facility.

6.5.2.2 Transferring, Storing, and Processing Fissionable Materials

The transferring of fissionable materials satisfies the following requirements abridged from WSRC-SCD-3 (Ref. 2):

- Onsite and offsite transfers of fissionable materials comply with requirements specified in applicable DOE and WSRC documents and are performed in accordance with approved procedures.
- Operating procedures, including traffic controls, are implemented, as appropriate, during onsite transfers of fissionable materials to minimize accident probabilities.
- The site emergency plan includes coverage of potential accidents during onsite transfers of fissionable materials, notification of emergency response, and prior to specific onsite transfers organizations, when appropriate.
- Onsite and offsite transfers of fissionable materials are made using approved packaging, documentation (including NCSEs, if necessary), and procedures as specified in WSRC-SCD-3 (Ref. 2).
- The movement of fissionable materials in significant quantities (as specified in WSRC-SCD-3) within the boundary of a facility, or any other movement that is not an onsite transfer, complies with the requirements of WSRC-SCD-3 and includes an NCSE and the use of approved procedures (Ref. 2).
- Approved operating procedures applicable to an onsite transfer of fissionable materials are readily available within the loading, unloading, or storage areas for the materials.
- Fissionable materials in significant quantities (as specified in WSRC-SCD-3 [Ref. 2]) and contained in approved packaging are defined as being stored, unless they are in the active custody of a material handler or other such person, are being handled, or are being processed.
- Offsite shipments involving fissionable materials in significant quantities (as specified in WSRC-SCD-3 [Ref. 2]), received or originating at WSRC, are contained in packaging approved by DOE, the Nuclear Regulatory Commission, or the Department of Transportation, as appropriate, for transport of fissionable materials.
- Offsite receipts or shipments of fissionable materials are handled in accordance with the package shipping requirements specified in approved documentation available in the area where the package is loaded or unloaded.

The storing and processing of fissionable materials satisfy, as appropriate, the following requirements abridged from WSRC-SCD-3 (Ref. 2):

- Nonessential combustible materials are not stored, if avoidable, in a fissionable material storage area.

- A fire protection system is installed, in fissionable material storage areas, where the presence of significant quantities of combustibles cannot be avoided.
- Process operations, storage of nonnuclear materials, or equipment, that is not directly required for fissionable material storage operations, and all other functions not directly a part of normal fissionable material storage operations are excluded from the storage area, if possible.
- Documented inspections, in situ tests, and preventive maintenance are performed periodically on fissionable material storage areas to ensure that the safety systems and components necessary for criticality control are maintained in a state of readiness.
- Criticality safety limits are conspicuously posted at the entrance of, as well as inside, fissionable material storage areas, as applicable.
- Signs or other devices are used, as appropriate, at strategic locations in or near fissionable material storage areas to provide instructions regarding the following:
  - Interpretations of, and required responses to, alarms
  - Evacuation routes
  - Firefighting
- A firefighting plan, including appropriate training drills and exercises, is developed for fissionable material storage areas and is incorporated into the overall facility and site emergency plans.
- Auxiliary firefighting equipment, self-contained breathing apparatuses, and protective clothing are provided in, or near, fissionable material storage areas, as necessary and appropriate, to facilitate manual fire suppression.
- Excess fissionable material should not be construed as being "in process" to circumvent fissionable material storage area requirements.
- Fissionable materials may be stored in shipping containers for the purpose of enhancing safety, but not for the purpose of circumventing fissionable material storage area requirements.
- Fissionable materials are stored in racks (or equivalent fixtures) capable of securely locating stored materials to prevent displacement, to ensure spacing control, and to meet design requirements for criticality safety under normal operational and credible accident conditions. Floor storage within a storage facility is only permitted where the original containers and their restraints inherently provide control of location or other safety requirements, (equivalent to the safety provided by storage racks).
- Fissionable materials that are determined to be pyrophoric are put into a safe form that is nonpyrophoric prior to storage, or are stored in approved containers that do not permit spontaneous ignition, or dispersal.
- Dispersible fissionable materials are stored in approved storage containers.

- Fissionable material storage containers are marked, or coded, to indicate the type, or category, of the material, the amount and enrichment of the material, and the radiation level at the outside surface of the container.
- Containers are securely closed and positioned so as to prevent significant displacement and to maintain criticality prevention requirements.
- Plutonium- or U-233-bearing, or contaminated, material is packaged in a closed metal storage container, when practical; however, combustibles within closed storage containers are minimized.
- Plutonium storage facilities and containers are monitored and checked periodically to ensure continued integrity of containment.
- Plutonium containers in which gas buildup can occur are designed either to prevent leakage of gas over the maximum storage period, or to be vented, (without the spread of contamination) as necessary, to prevent container bulging, or an accumulation of explosive gases.
- Fissionable material storage container designs are based upon defined and documented criteria and are periodically inspected against the criteria with the inspection frequency dependent upon container quality and type.
- Provisions are made in plutonium storage facilities to ensure necessary and adequate heat removal for plutonium storage containers as established by facility safety documentation.
- The effects of interstitial moderation are evaluated using a validated computational technique for units of an array where ANSI/ANS-8.7-1975 is used to establish mass limits and where interstitial moderation is credible (Ref. 13).
- Any alternative approach applied to a specific system, where the methodology of ANSI/ANS-8.7-1975 is unnecessarily conservative, is performed using a validated computational technique (Ref. 13).
- Access to fissionable material storage areas is controlled.
- Containers of fissile materials in areas with sprinkler systems are designed to prevent an accumulation of water.
- Consideration is given to the possibility of a criticality accident occurring from an accumulation of runoff water, when sprinkler systems are installed in fissile material storage areas.
- Paragraphs 5 and 6 of ANSI/ANS-8.7-1975 apply when ANSI/ANS-8.7-1975 is used to establish mass limits for units in an array (Ref. 13).
- Design criteria for fissionable material storage containers and storage and processing facilities satisfy applicable DOE and WSRC requirements.

The preceding requirements are not applicable to packages of materials prepared for shipment that are limited in accordance with DOE Order 460.1A, or radioactive waste storage or disposal facilities (Ref. 14).

6.5.2.3 Criticality Safety Posting and Labeling

Positive identification of fissionable materials, particularly fissile materials, is essential to criticality safety. Adequate labeling of fissionable materials and clear posting of work and storage areas where fissionable materials are present are provided to avoid the accumulation of unsafe quantities of such materials.

Posting refers to the placement of signs indicating the presence of fissionable materials, summarizing key criticality safety requirements and limits, designating work and storage areas, or providing instructions, or warnings, to personnel.

Posting considerations include the following items abridged from WSRC-SCD-3 (Ref. 2):

- Facility operations managers are responsible for ensuring that the WSRC-SCD-3 posting requirements are met in their facility (Ref. 2).
- The presence of fissionable materials is posted at the entrance to the actual work and storage areas (e.g., benches, hoods, gloveboxes, cabinets, rooms, zones, and modules, as applicable) where fissionable materials are handled, processed, or stored.
- The identification symbol used to identify the presence of all fissionable materials is as specified in ANSI Standard N12.1-1989 and referred to as the "fissile material symbol" (Ref. 15).
- Criticality safety limits are posted in conspicuous places near fissionable material storage areas, as appropriate, accompanied by additional information as specified by WSRC-SCD-3 (Ref. 2).
- Key limits and controls that are controllable, or observable, by an operator are posted at each work station in a process area, as necessary, to supplement operating procedures; however, care should be exercised to avoid posting so many limits that confusion develops.
- Each process facility develops facility-specific criteria to be used as a basis for determining the key limits and controls to be posted versus those controls that only appear in operating procedures.
- Limits are posted at the entrance of individual sub-areas (e.g., labs and groups of labs), as applicable, for laboratories or other areas using administrative mass control limits.
- Criticality safety precautions, or prohibitions, related to firefighting, such as precautions, or prohibitions, on the use of water, are posted at the entrance to areas containing fissile materials, as appropriate, and are coordinated with the area fire companies.
- Posting may also be required for other purposes, such as accountability or security, and may appear on a single sign as long as separate meanings are not compromised.

- Fissionable material postings are not required for natural, or depleted, uranium.
- Fissionable material postings are not required at high-level waste storage/treatment tanks, or low-level areas, such as disposal trenches and vaults.
- Fissionable material postings are not required for equipment that is monitored and controlled remotely rather than on location; however, control room postings should be used as necessary to complement procedures and aid operator performance.
- When fissionable materials are permitted on both sides of a wall, or other visual obstruction, and a special mass, or spacing limit, applies to both areas because of potential interaction, the limit, or limits, are posted in both areas.
- Nuclear incident evacuation routes and instructions are posted to clearly indicate the evacuation routes from a facility, or area.
- Postings should be easy to read and located in such a manner that they may be easily seen.
- Each division/area/facility may add additional posting requirements, according to their own particular needs, as long as the postings comply with applicable DOE and WSRC requirements.
- If an incorrect and unsafe posting is identified and the posting involves a criticality safety limit, fissionable material handling and processing operations are brought to an orderly stop in the affected area, and supervision is notified.
- Facilities are responsible for developing postings in accordance with the requirements and guidance in WSRC-SCD-3 and for obtaining assistance (as needed) and concurrence from the CSF and review and approval by the cognizant CSC (Ref. 2).
- A posting should be of durable construction to survive in the environment in which it is located and under the conditions it may be expected to encounter.

Labeling refers to the placement of clear and positive identifying marks on specific units, or batches of fissionable materials (e.g., cans, packages, containers, boxes, reactor fuel assemblies, and targets) to prevent them from being mistaken for other materials and to clearly show the type and amount of fissionable materials present. Labeling considerations include the following items abridged from WSRC-SCD-3 (Ref. 2):

- A fissionable material label is used to identify non-process storage, transport, and burial containers of fissionable materials containing over 1 gram, whether present as the elemental material, in compounds, or in mixtures.
- Containers of fissionable materials received from offsite locations that meet the requirements of DOE Order 460.1A have a label affixed, unless an unusually high radiation dose would result from the labeling process (Ref. 14). If an unusually high radiation dose is expected from the labeling process then an alternate means

of positively identifying the containers and of clearly identifying the type and amount of fissionable materials inside the containers is used.

- Labels should clearly show all information necessary to ensure adequate identification of fissionable materials.
- Labels may be developed by each division, area or facility to suit their own needs.
- Label requirements for unirradiated reactor fuel elements and targets consist of a unique serial number etched/machined onto each reactor fuel element and target. Accompanying paperwork/cards specific to each element, or target, reflect the appropriate serial number, providing the type of information, as applicable.
- Label requirements for irradiated reactor fuel elements and targets are similar to those for unirradiated reactor fuel elements and targets.
- Fissionable material containers that no longer contain fissionable material (i.e., <1 gram) are labeled as empty, and/or the old fissionable material label is removed.

Information related to specific facility posting and labeling is addressed in the facility-specific SARs, as appropriate.

6.5.2.4 Evacuation and Treatment Considerations for Criticality Accidents/Nuclear Incident Monitor Alarms

The evacuation and treatment considerations for criticality accidents/NIM alarms are as follows:

- Emergency procedures for criticality accidents are prepared for each facility in which criticality safety controls are instituted and/or criticality alarm systems are installed. Such emergency procedures are approved by division/area management and the WSMS affiliate and are consistent with WSRC-SCD-7 (Ref. 16).
- Personnel assembly stations (i.e., rally points), located outside the areas to be evacuated in the event of a criticality accident, are designated for each facility, as applicable, by the management of such facilities in conjunction with the CSF and division/area management.
- Provisions are made for the evacuation of transient personnel, as applicable, from the vicinity of a facility in which a criticality accident may occur.
- Arrangements are made in advance for the care and treatment of personnel injured and exposed as a result of a criticality accident.
- Procedures are developed by facility and division/area S&HO for the immediate identification of individuals exposed in a criticality accident and include personnel dosimetry.
- Instrumentation and procedures are provided by facility and division/area S&HO for determining the radiation level at the rally point and in the evacuated area following a criticality accident.

The responses to criticality accidents/NIM alarms for a facility are summarized in the facility-specific SARs.

6.5.2.5 Firefighting and Criticality Safety

Water, the firefighting agent used most often, is an efficient moderator and reflector of neutrons. In the absence of moderating materials such as water, relatively large masses of dry fissile materials, such as powders or metals, may be handled safely. If the presence of water is likely, then some operations with dry fissile materials may have to be constrained, modified, or eliminated.

The following general considerations are provided for firefighting in areas where fissile material may be handled:

- Divisions, areas, and facilities establish guidelines for permitting aqueous, or carbon-based fire suppression agents to suppress fires within, or adjacent to, moderation controlled areas.
- Automatic water sprinkler coverage is provided, as applicable, throughout a facility, except in areas where the potential for nuclear criticality, or other hazards, specifically preclude its use.
- When the use of water sprinkler coverage is precluded because of the potential for nuclear criticality or other hazards, non-aqueous extinguishing systems, sand, or halogenated organics are used.
- Automatic fire protection systems are designed such that they are not a credible cause of an inadvertent criticality accident.
- The presence of non-fissionable combustible/flammable materials is minimized in areas in which the potential for a criticality accident exists consistent with the nature of the facility.
- Any facility that has restrictions on the use of water for firefighting includes such restrictions as part of the routine criticality safety training for that facility.

The following conditions apply to fire preplans:

- A fire preplan is prepared for each facility in which criticality safety is of concern and specifically includes criticality safety considerations.
- The fire preplan is prepared by the management/engineering staff of each facility with assistance by the CSF, site fire safety engineers, and area fire department personnel, as necessary.

- Typical categories or types of areas that are considered in developing fire preplans for facilities containing fissile materials are defined in WSRC-SCD-3 and include the following (Ref. 2):
  - An area with no possibility of accidental criticality if water is used in any manner to fight fires, either because little, or no, fissile material is present, or because the material cannot be arranged into a critical configuration even in the presence of water, or other moderating substances.
  - An area in which the use of water to fight fires may violate a criticality safety limit, but double-contingency control will be maintained and criticality cannot occur.
  - An area in which fissile material is present in amounts and configurations that could become critical by the combination of fire, consequential conditions, the addition of water, and the concurrent physical rearrangement of the fissile material.
  - An area in which fissile materials are handled in quantities or in configurations, such that criticality safety requires the exclusion of water for firefighting purposes. Only dry chemicals, gases, or high expansion foam may be used to fight a fire.
  - An area in which the pyrophoric nature of the fissile material requires the use of firefighting agents other than, or in addition to, water.
- Fire preplans are kept on file by the facility and the area fire department.
- Area fire departments periodically review the fire preplans for all facilities under their cognizance in which firefighting restrictions have been instituted for criticality safety purposes.
- Facility changes (e.g., equipment, process types or conditions, types and quantities of fissile material) are reviewed to determine if the fire preplan requires revision. The WSMS affiliate and area fire department personnel review any changes to the pre-fire plan and the WSMS affiliate approves such changes.

The criticality aspects of firefighting plans and procedures are summarized in the appropriate chapter of the facility-specific SARs.

6.5.2.6 Procedure Review, Approval, and Control

The procedure review, approval, and control considerations are as follows:

- All procedures that involve operational changes related to systems currently having criticality safety controls, or that may require criticality safety controls because of new or planned operational changes, are reviewed and approved by the WSMS affiliate prior to system operation.

- All process flowsheet procedure changes that involve systems currently having criticality safety controls, or that may require criticality safety controls because of such changes, are reviewed and approved by the WSMS affiliate prior to system operation.
- All procedural changes that may impact criticality safety are reviewed and approved by the WSMS affiliate prior to use.
- Engineering drawings identify equipment and engineered systems important to criticality safety, particularly if such equipment, or systems, are used as a double-contingency defense. Engineering drawings that show equipment and systems important to criticality safety are reviewed by the WSMS affiliate. Changes to drawings involving equipment important to criticality safety are reviewed by the WSMS affiliate.
- Supervisors periodically review active procedures. The requirement to periodically review active procedures is itself a procedure.

Details of the procedure generation, review, and approval process for a specific facility are provided in the facility-specific SARs.

6.5.3 CRITICALITY SAFETY TRAINING

WSRC-1-01, MP 1.18 states that it is WSRC's policy to provide its employees with the training and development opportunities necessary for them to complete their work assignments in a safe, effective, and quality manner (Ref. 17). Regarding criticality safety training, the training policy supports WSRC-1-01, MP 4.5 in that all reasonable efforts are taken to reduce, or eliminate, the potential for, and consequences of, a criticality accident (Ref. 1).

This section describes criticality safety-related WSRC requirements for the selection, criticality safety training, examination, qualification, retraining, reexamination, and requalification of individuals who do the following:

- Work with, handle, or store greater than exempt quantities of fissionable materials, or work with equipment (including construction and maintenance) in which greater than exempt quantities of fissionable materials are processed
- Manage facilities, or provide engineering support, to facilities in which greater than exempt quantities of fissionable materials are stored, handled, or processed
- Perform special nuclear material accountability functions
- Perform nuclear materials packaging activities
- Perform reviews of criticality safety documentation

6.5.3.1 Establishment of Savannah River Site Criticality Training Requirements

The Nuclear Operating Divisions have each developed criticality safety training programs consistent with S/RIDs for their personnel (Ref. 18). This criticality safety training provides instruction for the following:

- Facility operators
- Supervisors
- Fissionable material handlers
- Accountable material handlers
- Maintenance mechanics
- Engineers providing technical or process support
- Fissionable material custodians
- Construction
- Facility management
- Laboratory technicians and operators
- Personnel who perform FEB, or ORR reviews
- Other personnel whose job assignment requires them to be in an area where controls are instituted to ensure criticality safety

Some divisions have common criticality safety training programs and/or training personnel. Specific training requirements for facility personnel are provided in the facility-specific SARs.

The criticality safety training programs consider the nature of a person's assignment. Those persons whose responsibilities are managerial are provided training that includes information on how to manage criticality safety concerns. Those persons whose responsibilities are technical, or operational, are provided training that includes information related to the development of technical and operational competence. The training of supervisors is of greater depth than that of operators, or fissionable material handlers.

Engineers who do not work directly in or for the facility may also require some form of criticality training. Design engineers (PE&CD) responsible for the design of facilities, processes, or equipment involving fissionable material receive annual criticality safety training.

Each division documents the criticality safety qualifications of each job category (i.e., education, experience, and training) of staff personnel. Qualification of such persons is valid for a maximum of 2 years (unless revoked for cause), at which time personnel are re-qualified in accordance with their facility-specific criticality safety training requirements.

6.5.3.2 General Facility Criticality Training Requirements

The general facility criticality training requirements include the following:

- Facility managers, level three and below, complete general criticality safety training. Engineers and scientists directly assigned to a facility and responsible for the operation, or maintenance of, processes, laboratories, or equipment involving fissionable materials in the facility complete both general and facility-specific criticality safety training. This training is conducted annually and is documented.
- The following areas are addressed in the facility criticality safety training programs and are discussed in the facility-specific SARs (additional guidance for facility criticality safety training programs is provided in WSRC-SCD-3 [Ref. 2]):
  - Operational criticality safety training
  - Reexamination
  - Requalification
  - Qualification extension
  - Training program review
  - Reentry team member training
  - Training program records and retention
- Training for responding to NIM alarms is conducted for new employees, visitors, transferees, and employees in areas that have NIMs installed and where changes have recently been made in NIM systems that affect the alarms. Personnel are made aware of NIM alarm sounds and characteristics and are briefed on actions to take if alarms sound.

6.5.4 DETERMINATION OF OPERATIONAL NUCLEAR CRITICALITY LIMITS

6.5.4.1 Nuclear Criticality Safety Evaluation Methodology and Bases

An NCSE is an independent, documented analysis that establishes the technical basis for nuclear criticality safety and provides the basis for, and recommends, subcritical operating limits, criticality safety controls, and engineered criticality safety features. These NCSEs identify the minimum subcritical margin when establishing such limits. NCSEs may also identify specific procedural and hardware limits necessary to implement limits or to document criticality evaluations that do not produce specific limits.

The general requirements for NCSEs are described in WSRC-SCD-3 and are summarized as follows (Ref. 2):

- The WSMS affiliate, or others such as HLWM staff, prepares and documents NCSEs for facilities, processes, and operations in which they are cognizant.
- The WSMS affiliate provides independent review, confirmation, and approval of, the methods and results contained in all NCSEs.
- After approval of an NCSE by the WSMS affiliate, the NCSE is issued to the requesting organization, or, if the evaluation was generated by an organization other than the WSMS affiliate, that organization may issue the NCSE to appropriate parties on its own.
- The subject of the NCSE (e.g., the equipment, and process) is not operated with fissionable materials until the NCSE has been independently reviewed, verified, and approved and until operating limits (e.g., TSRs), operating procedures, and control devices are developed and verified/approved, as applicable.
- All fissionable materials that are, or may be, present are considered in the evaluation. If changes to the mass of fissionable materials are credible, they are considered, including the maximum quantities likely to be present.
- The evaluation considers all materials with neutron-moderating properties (e.g., water, plastics, organic solutions, or carbon) that are normally present, or that may be introduced during credible abnormal conditions (e.g., water from fire sprinklers).
- The degree of neutron reflection is considered for all systems containing fissionable materials based on actual reflectors present or expected under normal or credible abnormal conditions.
- Solid, or soluble, neutron absorbers (poisons) may be included in the evaluation as long as their presence is confirmed and ensured.
- The evaluation considers the geometry of packages, vessels, and equipment containing fissionable material or the geometry of the fissionable material itself under normal and credible abnormal conditions (e.g., vessel overpressure that may change the original geometry).
- The concentration, or density, of fissionable material under normal and credible abnormal conditions (e.g., precipitation, or phase separation) is considered in the evaluation.
- The evaluation considers neutron interaction among fissionable material systems, or individual units unless such systems, or units, are demonstrated to be sufficiently separated, or neutronically isolated from each other.
- The evaluation may assume actual enrichment of the fissionable isotope provided that confirmation exists of the enrichment and that changes to the enrichment are not credible.
- An NCSE determines and explicitly identifies the controlled parameters and their associated limits upon which criticality safety depends. An NCSE also defines the assumptions for uncontrolled parameters.

- The format and content of the NCSEs usually conform to the guidelines delineated in WSRC-SCD-3 (Ref. 2).

The requirements for the analysis methods and safety margins are as follows:

- Any analytical method used to determine criticality safety limits is validated (whenever possible) by comparison with known experimental critical, or subcritical systems.
- Various computer codes are available for criticality safety analysis ranging from one-dimensional codes, which are suitable for simple problems, to three-dimensional codes, which can satisfactorily describe complex geometries.
- Checks are performed of all computer codes used for criticality safety calculations to confirm that the mathematical operations are performed as intended.
- Any change to a computer code is followed by a reconfirmation that the mathematical operations are performed as intended.
- Computer codes for criticality safety calculations comply with Quality Assurance Procedures of WSRC Procedure Manual 1Q (prior to May 1995), Safety Engineering Department procedures, or Manual E7 (Ref. 10, 19, 20).
- Any method used in an NCSE to determine criticality safety limits has its calculational bias established by correlating the results of criticality experiments with results obtained for these systems by the method being validated.
- If the calculational method area of applicability is extended beyond the range of experimental conditions over which the bias is established, then trends are established in the bias that may be expected to apply to the extended area of applicability. When the extension of the area of applicability is large, other methods are used to supplement the applied method to provide a better estimate of bias in the extended area.
- The analysis demonstrates that there is an acceptable margin of sub-criticality for all normal and credible abnormal conditions.
- Initial scoping calculations may be made without the need for independent review.

The following criteria are used in identifying the need for an NCSE:

- Operating organizations are responsible for identifying the need for a new or revised NCSE. However, the WSMS affiliate may also decide that an NCSE is needed to address a particular situation. Similarly, groups such as the CSCs, NCSRC, or design groups of PE&CD may also request NCSEs.
- A new NCSE is required whenever a new fissionable material handling, processing, transfer, shipping, or storage operation is planned with greater than exempt quantities of fissionable materials. A new, or revised, NCSE is required when an existing operation involving the handling, processing, transfer, shipping,

or storage of fissionable materials is changed beyond the scope of existing NCSEs and established limits.

- Changes that may require a new, or modified, NCSE are reviewed by the appropriate facility manager and by site management, as appropriate.

The facility-specific SARs provide summaries of NCSEs used to establish the technical basis for nuclear criticality safety of the facility and provide the basis for, and recommendations of, subcritical operating limits, criticality safety controls, and engineered criticality safety features. The summaries of the NCSEs include discussions of the following:

- Analytical methods, codes, and analysis techniques used to derive operational nuclear criticality limits and the error contingency criteria or margin of error (i.e., uncertainty)
- Use of contingency analyses
- Basic justification of the appropriateness of the analytical approach used

6.5.4.2 Nuclear Criticality Safety Supplements

Nuclear Criticality Safety Supplements (NCSSs) are criticality safety supplements to Technical Standards for various facilities. WSRC Manual SCD-3 provides the requirements for NCSSs (Ref. 2). Technical Standards are in the process of being converted to TSRs in accordance with DOE Order 5480.22 (Ref. 21).

The facility-specific SARs provide summaries of NCSSs used to establish the technical basis for nuclear criticality safety of the facility and provide the basis for, and recommendations of, subcritical operating limits, criticality safety controls, and engineered criticality safety features. The summaries of the NCSSs include discussions of the following:

- Analytical methods, codes, and analysis techniques used to derive operational nuclear criticality limits and the error contingency criteria or margin of error (i.e., uncertainty)
- Use of contingency analyses
- Basic justification of the appropriateness of the analytical approach used

6.5.4.3 Nuclear Safety Data Sheets

Nuclear Safety Data Sheets (NSDSs) are unique to Spent Fuel Storage Division (SFSD) Facilities, and serve as the base document for approval of fuel specific criticality safety limits. An NSDS is generated for each fuel type handled or stored in the SFSD basins. The Nuclear Criticality Safety and Fuel Receipt Manual – Spent Fuel Basin Facilities and WSRC-SCD-3 provide the requirements for NSDSs (Ref. 2, 22). NSDSs are derived from NCSEs that analyze all aspects of handling and storage of the fuel type in the basins. The NSDS document provides a mechanism for approval of the criticality safety limits and controls to be applied for the fuel type.

The specific limits are transcribed from the applicable NSDS directly into operational procedures used in the facility.

The RBOF-specific SAR provides examples of NSDSs used to establish the subcritical operating limits and criticality safety controls for RBOF and discusses preparation of the NSDS. The types of criticality safety limits and their corresponding controls are summarized in the facility-specific SAR. Individual NSDSs are not summarized in the facility-specific SAR.

6.5.4.4 Nuclear Criticality Safety Analysis Summary Report

A Nuclear Criticality Safety Analysis Summary Report (NCSASR) summarizes applicable NCSEs, criticality accident scenario analyses (e.g., fault trees, and event trees), and/or detailed double-contingency analyses applicable to TSRs or their revisions. The NCSASRs may form the basis for SARs/TSRs. NCSASR requirements are discussed in WSRC-SCD-3 (Ref. 2). Portions of an NCSASR may be directly incorporated into various sections of a SAR/TSR, as necessary.

6.5.4.5 Safety Analysis Report Nuclear Criticality Safety Information

The basis for criticality safety is included in the facility-specific SARs. In addition to the requirements of the ANS standards, detailed nuclear criticality safety analyses are performed for specific operations, storage arrangements, and the handling and transportation of fissionable materials. Additional requirements for nuclear criticality safety analysis are delineated in WSRC-SCD-3 (Ref. 2).

6.5.4.6 Interrelationship Between Operational Nuclear Criticality Limits and Technical Safety Requirements

Operational nuclear criticality limits and TSRs are developed based on NCSEs for the facility.

The margins of safety associated with operational nuclear criticality limits are greater than or equal to the margins of safety associated with the TSRs.

6.5.5 CRITICALITY ASSESSMENTS, INSPECTIONS, APPRAISALS, AND REVIEWS

This section provides an overview of the criticality assessments, inspections, appraisals, and reviews (including record keeping) discussed in WSRC-SCD-3 as well as in other sections of this chapter (Ref. 2). Section 6.5.1 discusses related responsibilities of the criticality safety organization, including general guidelines and criteria for conducting criticality assessments, inspections, appraisals, and reviews. Specifically, Section 6.5.1.2 discusses related ESH&QA responsibilities, while Section 6.5.1.10 discusses the involvement of the criticality safety committees.

The facility-specific SARs provide additional details, as necessary. Chapter 17 of this GSAR provides additional information concerning safety reviews and performance assessments, including those in the area of criticality safety.

6.5.5.1 Assessments and Inspections

Each facility that requires criticality safety controls periodically performs criticality safety self-assessments. These criticality safety self-assessments are conducted at least annually. The criticality safety self-assessments are defined as reviews of operations important to criticality safety or of progress on issues and findings from previous self-assessments, DOE appraisals, or reviews, and facility walkthroughs.

Division/area CSCs may assist facilities under their cognizance in performing criticality safety self-assessments.

6.5.5.2 Appraisals

The ESH&QA FEBs perform independent oversight of criticality safety self-assessments performed by facilities.

The DOE Savannah River Operations Office and DOE Headquarters conduct periodic criticality safety appraisals of selected facilities at SRS.

6.5.5.3 Reviews

The NCSRC and area CSCs conduct criticality safety reviews, as necessary, of facility restarts, new, or modified, control systems/equipment, limit violations, etc.

Reports are prepared presenting the results of a criticality safety review and are distributed to the management of the facility under review.

6.5.5.4 Record Keeping

Each division, area, or facility, as applicable, implements a formal documented system for the control and retention of criticality safety documents and records. As indicated in WSRC-SCD-3, the document and record management system established by the WSRC Sitewide Records Inventory and Disposition Schedule, in conjunction with the WSRC quality assurance program, satisfies this requirement (Ref. 2). The criticality safety document and record management system is based upon and includes the following detailed items:

- Those NCSEs, NSDSs, NCSSs, NCSASRs, Technical Standards, Technical Specifications, Test Authorizations, operating procedures, and Operational Safety Requirements (OSRs)/TSRs that contain information important to criticality safety are considered criticality safety documents and records.

- The WSMS affiliate includes a system for uniquely numbering and archiving NCSEs.
- A system exists, at the facility level, for linking criticality safety requirements, limits, steps, and/or conclusions in Technical Standards, Technical Specifications, Test Authorizations, operating procedures, OSRs/TSRs, NCSSs, and NCSASRs to the NCSEs that serve as the basis.
- A system exists for linking functional tests and new (or revised) as-built drawing dimensions back to the NCSEs that serve as the basis.
- A system exists for tracking the status of the following:
  - Assessment deficiencies
  - Corrective actions
  - FEB/ORR findings
  - DOE appraisals
  - Other assessments, inspections, appraisals, and reviews related to criticality safety
- A system exists for tracking approvals of criticality safety documents that ensures that all necessary approvals are obtained prior to authorizing the use of processes, operations, procedures, or equipment whose safe performance is dependent (in whole or in part) upon such criticality safety documents.

6.5.6 CRITICALITY EMERGENCIES AND LIMIT VIOLATIONS

6.5.6.1 Criticality Infraction Reporting and Follow-up

Abnormal operating conditions are related to specific TSR operating limit violations. Division, area, or facility abnormal operating procedures govern actions to be taken, in the event of an unanticipated situation, to place the operation into as stable and safe a condition as possible, until a criticality safety engineer/specialist can conduct an evaluation. Such actions may involve stopping the movement of nearby fissionable material, isolating the particular part of the process, and excluding persons from the immediate area.

The WSMS affiliate approves all recovery procedures for recovery from a limit violation.

DOE is notified of criticality TSR operating violations in accordance with S/RIDs and site reporting requirements, as appropriate. Reporting also ensures that other operations may benefit from lessons learned. For some facilities, the Double-Contingency Analysis (DCA) is an AB level document with controls much like the TSR. These DCA control failures have similar reporting requirements to TSR violations for those facilities with AB level DCAs.

If a criticality control step in an operating procedure is violated, activities controlled by the procedure are discontinued immediately, unless it is unsafe to do so, and supervision is notified.

If a limit for a mass control zone (i.e., an area, room, or rooms in which the fissionable material inventory is administratively limited) is violated, operations are stopped immediately, unless it is unsafe to do so, and supervision is notified.

If an equipment failure affects criticality control or monitoring, process operations are placed in safe shutdown, and fissionable material movements through the facility are suspended, if necessary, depending on the nature of the equipment and the facility operation. Facility management is notified immediately.

If it is determined that fissionable material is being stored, or handled, in excess of exempt quantities without appropriate criticality safety controls, then such activities are suspended immediately and supervision is notified.

Emergency procedures clearly specify reporting responsibilities and duties of the Facility Emergency Coordinator and the Area Emergency Coordinator in the event of a criticality accident or a NIM alarm of unknown origin. Chapter 15 of this GSAR and the facility-specific SARs contain additional information concerning emergency procedures.

6.5.6.2 Recovery from a Criticality Infraction

Procedures are developed, as appropriate, to govern actions to be taken in the event of an unanticipated criticality infraction. The procedures' objectives are to ensure that the operation is placed into a stable/safe condition until a criticality safety engineer can conduct an evaluation.

Recovery from violations of criticality limits is accomplished in a manner to ensure that the remaining safety margin is acceptable, or is not further reduced, if already unacceptable.

If the situation permits, safe shutdown and recovery from a criticality limit violation are conducted under an approved procedure consistent with existing requirements to conduct all operations in accordance with written procedures.

Procedures for recovery from a criticality limit violation assign specific responsibilities for actions to be taken during the recovery, including the role of the Facility Emergency Coordinator, the Area Emergency Coordinator, the recovery team, and the Technical Support Center.

6.5.6.3 Classification, Investigation, and Reporting

Deviations from procedures and unforeseen alterations in process conditions that affect criticality safety are reported to management and investigated promptly.

Violations of criticality safety limits, criticality safety steps in procedures, criticality safety procedures, Technical Standards, or equipment failures are classified as either an "off-normal

occurrence," an "unusual occurrence," or an "event", and are handled according to WSRC Procedure Manual 9B (Ref. 23).

An unplanned critical excursion is classified as an "event" and handled according to WSRC-SCD-3 (Ref. 2).

6.5.6.4 Corrective Action

Following recovery from a criticality limit violation, permanent corrective actions are developed and implemented to reduce the probability of a violation recurrence (e.g., prepare better procedures, install more reliable equipment, improve training, or install additional passive engineered controls).

Following recovery from a criticality accident or from a near criticality accident (i.e., loss of control to the extent that no known reliable mechanisms to prevent a nuclear criticality are functional), corrective actions are developed and implemented to effectively preclude the recurrence of the event.

Lessons learned are incorporated into the facility's training program and nuclear safety analysis, as appropriate, for either of these events.

Chapter 11 of this GSAR provides additional information concerning corrective actions.

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6.6 CRITICALITY INSTRUMENTATION

At SRS, criticality alarm systems, as discussed in ANSI/ANS-8.3-1986, are known as NIMs (Ref. 8, 24). The overall NIM program is the responsibility of NCSRC Chairman. The primary purpose of NIM systems is to minimize, by means of quick detection and alarm, the acute dose received by personnel from a criticality (and potential criticality) accident in areas where the cumulative absorbed dose in free air may exceed 12 rad. The secondary purpose of NIM systems is to notify people to avoid the evacuated area and to notify appropriate response teams.

6.6.1 DESCRIPTION OF CRITICALITY DETECTION AND ALARM SYSTEMS

In general, NIM systems are provided wherever it is deemed that they will result in a reduction in total risk. Consideration is given to hazards that may result from false alarms. NIM instruments continually monitor each potential criticality accident site, except as provided in WSRC-SCD-3 (Ref. 2). Approved NIM instruments are either the Type 2 or the Type 3 instruments. In the areas requiring NIM coverage, two NIM arrangements are used: some areas use NIM pairs of independent NIMs, and some use a two out of three rating scheme. When used in pairs, each NIM instrument alarms individually if the radiation setpoint is reached, if the two out of three NIM instruments in the system must exceed their radiation setpoints to actuate the auxiliary alarm system. WSRC-SCD-3 describes the NIM systems and their requirements (Ref. 2).

NIM systems are installed and maintained operational for facilities in which the following apply:

- In those cases where the mass of fissionable material exceeds the limits established in paragraph 4.2.1 of ANSI/ANS-8.3-1986 and the probability of criticality is greater than 1E-06/yr (as documented in the DOE-approved facility-specific SARs, or other authorization basis document), a NIM system meeting ANSI/ANS-8.3-1986 is provided to cover occupied areas in which the expected dose exceeds 12 rad in free air, where a NIM system is defined to include a criticality accident detection device and a personnel evacuation alarm (Ref. 24).
- NIM coverage is considered for facilities in which the probability of occurrence of criticality has been reduced to less than 1E-06/yr because of passive engineered controls, active engineered controls, or administrative means of criticality control. Cognizant CSCs review such situations and may recommend that NIM coverage is necessary regardless of the estimated low probability of an inadvertent criticality accident based on the reliability of the imposed controls.

The facility-specific SARs describe the location of the NIM system in the facility and whether it uses NIM pairs or the new three-instrument NIM system, as appropriate.

6.6.2 EXEMPTIONS FROM NUCLEAR INCIDENT MONITOR SYSTEM COVERAGE

The basis for the decision not to install a NIM system or a criticality detection device is documented in the facility-specific SARs.

Justification not to install a NIM system is based on the following:

- NIMs are not required in occupied areas where the maximum foreseeable absorbed dose in free air due to a critical excursion will not exceed 12 rad (i.e., because of shielding). For the purpose of this evaluation, the maximum yield may be assumed to be no more than 2E+19 fissions. In these situations, a criticality detection device that does not have an immediate evacuation alarm is installed with an appropriate response time such that process-related mitigating action may be taken to terminate the event or evacuation. Delayed or manual alarms, or voice announcements, may be used in conjunction with limiting personnel doses.
- NIMs are not required if the probability of criticality has a frequency less than 1E-06/yr, except as provided in WSRC-SCD-3 (Ref. 2).
- NIMs are not required for spent fuel stored under water in spent fuel storage pools provided sufficient water shielding is maintained above the fuel to protect personnel.
- NIMs are not required for fissionable materials during shipment, or for material packaged in approved shipping containers awaiting transport, provided there is the following:
  - No credible criticality accident that could occur while the containers are on a loading dock, or in a staging area
  - No other operation involving fissionable materials that are not packaged as stated but remain on the loading dock, or staging area
  - Essentially zero neutron interaction between the shipping containers and other fissionable materials in adjoining areas
- NIMs are not required where a documented analysis, based on consideration of the physical form and isotopic distribution of fissionable materials, demonstrates that a criticality accident is not credible.
- NIMs are not required for burial grounds, if either the potential for a criticality accident is not credible, or the potential for a criticality accident is credible but the resulting dose from the maximum criticality event is less than 12 rad at the surface of the burial ground.

The preceding list of criteria also applies to the removal or permanent disabling of NIMs that are no longer required.

6.6.3 NUCLEAR INCIDENT MONITOR SYSTEM DESIGN REQUIREMENTS

The NIM Committee is responsible for SRS standardization of NIMs, specification of maintenance routines and schedules, and procedures by which design modifications for the instruments may be authorized. The NIM Committee also assigns responsibility for maintenance, installation, and troubleshooting to appropriate organizations. Other NIM

Committee responsibilities are contained in its charter and are described in WSRC-SCD-3 (Ref. 2).

Detailed NIM design requirements, including the selection of equipment functions and sensitivities, are described in WSRC-SCD-3 and the NIM Technical Manual (Ref. 2, 25). NIM testing criteria, location analysis criteria, and system operations are also addressed in the referenced documents as well as in the following sections.

6.6.4 NUCLEAR INCIDENT MONITOR TESTING

WSRC-SCD-3 specifies requirements, including frequencies of performance, in the following areas associated with NIM testing (Ref. 2):

- Initial testing and inspections of NIM systems
- Testing and inspections of NIM systems following significant modification to, or repair of, a NIM system
- Response checks of NIM systems
- Re-calibration of NIM systems
- Testing of NIM alarm systems
- Testing of NIM bell circuits
- Field audibility tests of NIM signals
- Checks of NIM alarm decibel levels
- Procedures for NIM testing
- Corrective actions when tests reveal inadequate NIM system performance
- Records of NIM system tests, re-calibrations, and corrective actions

6.6.5 NUCLEAR INCIDENT MONITOR LOCATION, EVACUATION BOUNDARY, AND SOURCE TERM ANALYSIS

NIM locations, evacuation boundaries, and source term analyses involve the following items abridged from WSRC-SCD-3 (Ref. 2):

- The NIM Committee is responsible for coordinating the configuration control of all facility NIM location documentation, including associated analyses, and for reporting NIM configuration control status to the NCSRC on an annual basis.
- The WSMS affiliate is responsible for estimating the yield of the maximum criticality accident of concern for facilities.
- Facility management/engineering, working together with the WSMS affiliate, determines the potential need for NIM systems and requests a NIM location analysis from the WSMS affiliate.

- The WSMS affiliate is responsible for performing the NIM location analysis using shielding codes or empirical/analytical equations, as appropriate, providing an independent check of results, documenting the results, and maintaining records of the analyses.
- The WSMS affiliate is responsible for performing NIM evacuation zone boundary calculations, as requested.

The facility-specific SARs provide information concerning actual NIM locations, if applicable. The preceding list of requirements and responsibilities also applies to the removal, or permanent disabling, of NIMs that are no longer required.

6.6.6 NUCLEAR INCIDENT MONITOR SYSTEM OPERATION

NIM system operation involves the following items abridged from WSRC-SCD-3 (Ref. 2):

- Instructions regarding the response to the NIM alarm signal (audible or visual) and the NIM evacuation routes are posted throughout the area covered by the NIM system.
- Emergency procedures are prepared by each division, area, or facility having a NIM system and clearly designate NIM evacuation routes that follow the quickest and most direct routes practicable, avoiding recognized areas of higher risk.
- NIM evacuation routes are established such that there is no confusion with other emergency postings such as radiological hazard, toxic gas alarms, or postings.
- The WSMS affiliate ensures that NIM evacuation routes are adequately posted.
- NIM Technical Standards establish NIM alarm settings that are both high and low enough to comply with the applicable requirements in WSRC-SCD-3 (Ref. 2).
- If one or more NIM units monitoring a potential accident location are found to be out of service as a result of a malfunction, all operations involving fissionable material at the location, including movement and processing, are suspended or brought to a controlled and orderly conclusion, until full NIM coverage is restored.
- If one, or more, NIM units monitoring a potential accident location are going to be unavailable because of authorized preventive maintenance, all operations involving fissionable material at the location, including movement and processing, are brought to a controlled and orderly conclusion prior to the maintenance and are suspended until NIM maintenance is completed and full NIM coverage is restored.
- If NIM units monitoring a potential accident location need to be temporarily bypassed, or otherwise taken out of service, to permit activities that may cause false alarms, all operations involving fissionable material at the location, including movement and processing, are brought to a controlled and orderly

conclusion prior to the activity and are suspended until the activity is completed and full NIM coverage is restored.

- In facilities where a criticality accident is credible, consideration is given to using a combination of shield design and facility layout to reduce radiation doses to adjacent work stations and exit routes that would result from a criticality accident.

6.6.7 NUCLEAR ACCIDENT DOSIMETRY

Fixed nuclear accident dosimetry units are installed in facilities having NIM systems.

Personnel nuclear accident dosimeters meeting the requirements of S/RIDs are worn by all personnel who enter an area having a NIM system (Ref. 8).

Chapter 7 of this GSAR discusses the role of accident dosimetry in determining the impacts of a criticality incident.

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GENERIC SAFETY ANALYSIS REPORT

CHAPTER 7

RADIOLOGICAL PROTECTION PROGRAM

September 1999

Westinghouse Savannah River Company
Aiken, SC 29808

SAFETY • RESPONSIBILITY • SECURITY
SRS
SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

DISCLAIMER

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TABLES

FIGURES

ACRONYMS AND ABBREVIATIONS

A/RAC	ALARA/Radiological Awareness Committee
ACL	Administrative Control Level
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ARA	Airborne Radioactivity Area
ARM	Area Radiation Monitor
BAT	Best Available Technology
CA	Contamination Area
CAM	Continuous Air Monitor
CFR	Code of Federal Regulations
cm	centimeter
CND	Criticality Neutron Dosimeter
CSWE	Central Services Works Engineering
DAC	Derived Air Concentration
DCG	Derived Concentration Guides
DOE	Department of Energy
DOELAP	DOE Laboratory Accreditation Program
ED	Emergency Director
EDP	External Dosimetry Program
EM	Environmental Monitoring
EMS	Environmental Monitoring Section
EPA	Environmental Protection Agency
ETS	Environmental Technology Section
FARMS	Facility Annual Review of Monitoring Systems
FEC	Facility Emergency Coordinator
FLS	First Line Supervisor
GERT	General Employee Radiological Training
HCA	High Contamination Area
HPT	Health Physics Technology
HRA	High Radiation Area
mrem	milliroentgen equivalent man
NESHAP	National Emission Standard for Hazardous Air Pollutants - Radiological
NIST	National Institute of Standards and Technology
OS&HT	Occupational Safety and Health Technology

ACRONYMS AND ABBREVIATIONS (continued)

RA	Radiation Areas
rad	radiation absorbed dose
RBA	Radiological Buffer Area
RC	Radiological Control
RCO	Radiological Control Operations
RCT	Radiological Control Technician
REIRS	Radiation Exposure Information and Reporting System
rem	Roentgen equivalent man
RIDS	Records Inventory and Disposition Schedule
RMA	Radioactive Material Area
RW	Radiological Worker
RWP	Radiological Work Permit
RWT	Radiological Worker Training
S&HO	Safety and Health Operations
S/RID	Standards/ Requirements Identification Document
SAC	Site ALARA Committee
SAR	Safety Analysis Report
SCL	Special Control Level
SCSC	Site Central Safety Committee
SRD	Self Reading Pocket Dosimeter
SRS	Savannah River Site
SRTC	Savannah River Technology Center
TLD	Thermoluminescent Dosimeter
TLND	Thermoluminescent Neutron Dosimeter
VHRA	Very High Radiation Area
WSRC	Westinghouse Savannah River Company

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7.0 RADIOLOGICAL PROTECTION PROGRAM

7.1 INTRODUCTION

This chapter provides information on the radiological protection program for the worker and public as it applies to the Savannah River Site (SRS) facilities. The scope of this chapter includes the following:

- Description of the overall radiological protection organization
- Description of the policies and programs for reducing radiation exposures to values that are As Low As Reasonably Achievable (ALARA)
- Description of radiation exposure control including administrative limits, radiological practices, dosimetry, and respiratory protection
- Identification of radiological monitoring program to protect workers, the public, and the environment
- Discussion of radiological protection instrumentation
- Description of the program for maintaining records of radiation sources, releases, and occupational exposures

Additional or specific facility requirements are contained in the Chapter 7 of the facility-specific Safety Analysis Reports (SARs). The application of a graded approach may identify areas of this report which are not relevant to or different from a specific facility. These differences will be noted in the facility-specific SAR.

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7.2 REQUIREMENTS

Standards/Requirements Identification Documents (S/RID) state the codes, standards, and regulations governing the radiological protection program elements of the SRS (Ref. 1). Programmatic compliance assessment has been performed against the S/RIDs and documented as specified in the Westinghouse Savannah River Company (WSRC) Procedure Manual 8B(Ref. 2). The Standards Management/Compliance Section maintains records of the programmatic compliance assessments.

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7.3 RADIOLOGICAL PROTECTION ORGANIZATION

The Safety and Health Operations (S&HO) Department is the SRS consulting authority for radiological protection of site personnel and the public. The S&HO Department's Health Physics and Technology Section is responsible for the external dosimetry, internal dosimetry, instrument calibration, regulatory compliance, training, procedures, and health physics technology aspects of the radiological protection program. The S&HO Department is matrixed to the operating line organizations and is responsible for implementing the radiological protection program for the protection of the environment, the public, and the workers on and around SRS. In addition to responsibility for the radiological protection programs, the S&HO department is responsible for the field implementation of occupational safety and industrial hygiene programs for the site. The typical organizational structure of the S&HO Department is shown in Figure 7.3-1 (Ref. 3).

Specific objectives of the S&HO Department includes the following:

- Protects the health of employees, the general public, and the environment.
- Provides direction and oversight for the site radiological protection program.
- Ensures compliance with relevant federal and state regulations, Department of Energy (DOE) Orders, and WSRC directives governing the site radiological protection program.
- Enables SRS to safely and effectively meet its mission.

The S&HO Department has been assigned certain responsibilities in order to achieve these objectives. These responsibilities are included in Section 7.3.1.

The management at SRS is fully committed to maintaining external and internal exposures to radiation from site processes to ALARA levels. To establish a rigorous and frequent review of the program and performance against challenging goals, a network of review and approval committees is used to ensure adequate oversight by senior management (Ref. 4). The Site Central Safety Committee (SCSC) reviews radiological safety performance against established goals. The Policy Review Committee approves changes in radiological safety policies. These two committees are chaired by the WSRC President.

The Site ALARA Committee (SAC) is a subcommittee of the SCSC and is chaired by a senior site executive appointed by the WSRC President. The SAC reviews the overall conduct of the radiological protection program to ensure continuous improvement by developing ALARA initiatives.

The SAC has established ALARA/Radiological Awareness Committee (A/RAC) as a multidisciplined forum to include appropriate representatives of the line organizations, support organizations and significant others. The A/RAC provides the SAC with information concerning the implementation of ALARA policies and practices at SRS. The subcommittees are chaired by the SAC senior staff member of the division, or appointee. The A/RAC provides a direct link to

the workforce with respect to radiological work being planned and performed. Because the "perceived risk" involved varies with the type, amount and form of radioactive materials being handled, the number and kind of ALARA personnel and organization structure for implementing the ALARA program within the subcommittees vary.

Steering or ad hoc committees may be established to support sitewide activities that require interdepartmental coordination. The sponsorship of these committees is retained by the SAC. The support of these committees is retained by the A/RAC (Ref. 4).

7.3.1 RADIOLOGICAL PROTECTION PROGRAM

A description of the components of the radiological protection program at SRS is contained in the following sections. Included is a description of the S&HO organization, program objectives, and experience/qualifications of key radiological control personnel.

7.3.1.1 Program Organization

The S&HO Department is the responsible consulting authority within SRS for radiation protection of site personnel and the public. It is responsible for radiation hazard monitoring, awareness, analysis, direction, and advice to other departments on health hazards incident to the handling, use of, and exposure to radioactive materials. The S&HO Department also provides support to maintain each employee's personal work environment at a safe level of exposure from radiation.

The S&HO Department manager reports to the Vice President and General Manager of the Environmental, Safety, Health, and Quality Assurance Division. Activities in the S&HO Department is divided among three primary functions:

(1) Radiological Control Operations (RCO) - Provides radiological control field support to SRS facilities and organizations

(2) Health Physics Technology (HPT) - Responsible for internal and external dosimetry, the bioassay laboratory, in-vivo counting facility, calibration of portable instruments, and other technical support

(3) Training - Responsible for development of S&HO training programs (Ref. 5)

To fulfill its first responsibility to provide radiological control field support to SRS facilities and organizations, the S&HO area managers are matrixed to the line organizations that they support. Reporting to these S&HO area managers is S&HO facility managers matrixed to the facility line management.

The responsibilities of the S&HO Department include (Ref. 6):

- Stopping work when aware of a situation which reasonably poses an immediate danger to life or health
- Performing review activities to ensure compliance
- Providing occupational safety technical support to line organizations, analyzing hazards and devising methods of protection to ensure safety is not compromised
- Administering Occupational Safety Programs
- Maintaining and interpreting WSRC standards applicable to these programs
- Incorporating new or revised standards as necessary in response to DOE Directives, statutory requirements, WSRC policies, or operating experience (Occupational Safety and Health Technology [OS&HT] provides this responsibility for occupational safety and industrial hygiene programs.)
- Providing occupational safety program technical support to line organizations, analyzing hazards and devising methods of protection to ensure safety is not compromised
- Establishing programs, procedures, and practices that give preference to engineered corrective designs and guarded hazards over personal protective equipment

Responsibilities and functions of radiological protection personnel are described in WSRC Procedure Manual 5Q (Ref. 7). SRS maintains a level of radiological protection staff to ensure that the responsibilities and functions of the S&HO Department is fulfilled. See the facility-specific SAR for specific radiological protection staffing levels and requirements.

7.3.1.2 Program Objectives

The main objectives and implementation plans of the radiological protection program include the following:

- Minimize personnel exposures and skin contamination.
- Prevent intakes of radionuclides.
- Minimize contamination of facilities.
- Minimize the release of radioactivity to the environment.
- Train exempt and nonexempt radiological protection personnel in radiological work.
- Provide a series of special functions such as participating in Emergency Response Organization activities, reviewing site safety documentation for radiological issues, management of S&HO procedures, and providing radiological technical support for design and operations.

- Provide Quality Assurance (QA)/control on the various procedures and equipment to ensure proper implementation of radiological protection programs and operations.

The radiological protection program meets the radiological protection S/RIDs requirements, including training, measurements of radiation in the field (internal and external dosimetry), radiological design considerations, access and tool control, personnel radiation monitoring, emergency preparedness, program evaluation, and (QA). The requirements for the radiological tool control program are contained in Chapter 3 of WSRC Procedure Manual 5Q (Ref. 7). See the facility-specific SAR for specific information on tool control.

7.3.1.3 Experience/Qualification Requirements

Personnel associated with the radiological protection program must have a combination of education, experience, and training in order to perform their duties. The S&HO Department managers and supervisors are involved directly in the training and qualification of Radiological Control Technicians (RCTs). Experience and qualifications for radiological protection personnel are summarized in the following sections.

RADIOLOGICAL PROTECTION MANAGEMENT AND SENIOR STAFF

The management and professional staffs of the radiological protection organizations have qualifications that include a bachelor's degree (or the equivalent) in science or engineering, technical qualifications pertinent to their assigned duties, and technical education and refresher courses; attendance at professional meetings is encouraged. Senior staff are encouraged to pursue certification by the American Board of Health Physics.

SUPERVISORS

Supervisors are selected from qualified RCTs and participate in continuing radiological training programs. Supervisors are re-qualified every two years through comprehensive oral examination boards.

RADIOLOGICAL CONTROL TECHNICIANS

RCT qualification consists of standardized core course training material, on-the-job training per the Qualification Standard, and passing both a final comprehensive written examination and final oral examination board. RCTs are physically fit to perform assigned functions and have a minimum of a high school education or the equivalent.

Additional information on experience, qualifications, and responsibilities for radiological protection personnel is given in Chapter 6 of WSRC Procedure Manual 5Q (Ref. 7). Radiological protection training for radiological protection personnel is discussed in Section 7.5.

7.4 AS LOW AS REASONABLY ACHIEVABLE POLICY AND PROGRAM

7.4.1 AS LOW AS REASONABLY ACHIEVABLE POLICY CONSIDERATIONS

At SRS, it is the policy of the integrated team contractors (collectively known as WSRC) to comply with all DOE design and operational requirements addressed in applicable DOE rules, regulations, directives, and guidelines (Ref. 2). WSRC also has a corporate policy that human health and safety are the first priority. These policies form the basis for the primary objective of the SRS radiation protection program. The mission of the radiation protection program is to ensure safe handling and management of radioactive materials associated with site operation and to minimize radiation exposure and associated risk to human health and the environment over the lifetime of the radionuclides. Therefore, the radiation protection controls applied at SRS ensure that all radiation exposures are maintained ALARA and do not exceed applicable limits. These controls include engineered systems, administrative and procedural controls, protective equipment, state-of-the-art nuclear instrumentation for new facilities, and upgrading instrumentation for existing facilities as it becomes necessary. The SRS ALARA policy and program are provided in WSRC-SCD-6 and WSRC Manual 1-01 (Ref. 4, 6).

The ALARA concept at SRS is integrated into all site activities involving radioactive materials. ALARA is the responsibility of all SRS employees. Commitment to ALARA is demonstrated by the daily and detailed attention management gives to the radiation aspects of site operation and by the high priority assigned for facility modification and procedural changes to reduce radiation exposure. Information on progress and problems in reducing exposure is provided in periodic summaries, reports, and frequent meetings that review the radiation protection program.

WSRC Procedure Manual 5Q establishes practices for the conduct of radiological control activities at SRS (Ref. 7). It is issued by the WSRC President to invoke the requirements of the DOE Radiological Control Manual and applicable DOE rules and orders, and to implement the site radiological protection policies. The requirements of WSRC Procedure Manual 5Q apply to all DOE contractor employees, such as WSRC, University of Georgia, U.S. Forest Service, Wackenhut Services, Inc., their visitors, and other individuals at SRS (Ref. 7).

7.4.2 ASSOCIATED RADIOLOGICAL GOALS

The establishment of goals, their periodic review, and comparison with actual data are methods for tracking the progress toward the ultimate purpose of the ALARA program: reducing exposures to ALARA. Goals at SRS are established at least annually by those responsible for performing the work at the division/department/facility level. Periodic review of these goals against performance ensures that ALARA is considered in all facets of work at the site.

7.4.2.1 Establishment of Radiation Exposure Goals

The development of goals requires the review of historic work with associated exposures and schedules for anticipated production and maintenance. The Radiological Work Permit (RWP) program and associated pre-job and post-job ALARA reviews provide a base of historical information. After the type and amount of work that will actually be performed are developed and an estimated exposure value established, the amount of savings through implementation of ALARA principles will be subtracted. The goals for actual exposures should be less than the estimates (Ref. 4).

The initial development of radiation exposure goals is the responsibility of the facility and work group managers. Goals based on the planned work for the upcoming calendar year within the facility and the projected participation of any individual work group(s) are estimated. The ALARA coordinator for the specific division/department coordinates the total exposure goals.

After the facility has developed its activity plan and estimated the total exposure, the support organization work groups involved (e.g., Construction and S&HO) establish their ALARA goals. Work groups that have responsibilities in one or more facilities must coordinate their goals based on these activity plans (Ref. 4).

7.4.2.2 Documentation of Goal Performance

The trending of radiation exposures is a method of quantifying the success of the ALARA program. This trending applies both at the site level and the facility level. A performance indicator program to verify ALARA initiatives and for measuring and trending the effectiveness of the radiological control program against predetermined goals has been established. Site radiological performance indicators are provided in WSRC Procedure Manual 5Q (Ref. 7).

The S&HO Department provides a periodic summary report to the WSRC President and other affected organizations. This report includes the radiological performance goals established above and indicators that provide a more detailed analysis of performance. The S&HO Department also provides radiation exposure information, such as dosimeter readings, to supervisors and managers on a frequent enough basis to permit priority management of exposure control (Ref. 7).

7.4.2.3 Review of Goals

Senior management shall review progress towards the goals at least quarterly. Radiation exposure goals are compared to the actual exposure to measure performance in the quarterly reports.

Radiological performance goals are not normally revised, unless strong supporting documentation is provided. This adjustment provision is designed mostly to accommodate jobs that were started and not completed, delayed jobs, and operations that were not anticipated when the original goal was established. Any revisions will be recommended and approved using the method described in Section 3 of the ALARA Manual (Ref. 4).

Goals for minimizing solid, liquid, and airborne wastes are developed, approved and maintained by other SRS committees or organizations. Solid waste goals are provided in cubic feet by waste types (e.g., low level, hazardous, mixed low level and transuranic waste). Liquid and airborne waste goals are provided in terms of projected exposure to the public, expressed in milliroentgen equivalent man (mrem).

The individual A/RACs are responsible for interfacing with these other committees or organizations to correlate goals for reduction of existing contaminated areas and radioactive wastes. The reduction of existing contaminated areas need to be balanced by the recognition that in the short term, this may generate additional radioactive waste but, in the long term, it will reduce overall radioactive waste (Ref. 4).

7.4.3 PERSONNEL PROTECTIVE PRACTICES AND TECHNIQUES

Significant reductions in radiation exposure can occur at the task level, provided good job planning techniques are used each time a job is performed. These techniques not only include conducting pre-job reviews, but also evaluating the documented successes and shortcomings of completed jobs before performing a similar task.

Proper training is important in achieving good performance in radiological protection. Training is designed to supplement an individual's education and experience and provide the skill development and proficiency necessary to perform a particular job assignment. This performance includes the requirement to maintain exposures to radiation resulting from the site's operations to ALARA. Training is provided to all site personnel commensurate with the work to be performed (Ref. 4).

General ALARA practices and techniques are described below. The following methods are incorporated into the preplanning of tasks and development of procedures (Ref. 4, 7, 8):

- The source of radiation in an area may be from liquid material in lines or traps. Before conducting work, the lines may be drained and flushed. If draining and flushing are not possible, the use of shielding should be considered. The shielding may be temporarily installed, or where the work is a maintenance function that must be conducted on a routine basis, the installation of permanent shielding is considered. When evaluating the installation of shielding, the estimated exposure for installation and eventual removal is considered. The overall cost in dollars and initial radiation exposure may outweigh the savings (Ref. 4).
- As much as practical, jobs are performed in areas with low dose rates. This includes items such as reading instruction manuals or maintenance procedures, adjusting tools, repairing valve internals, and prefabricating components (Ref. 7).
- Access routes, as well as working areas, are determined based on lowest radiation and contamination levels (Ref. 4).
- Protective clothing and respiratory equipment are selected by the radiological control organization on the basis of the minimum levels of protection required to

ensure the health and safety of employees and maintain exposure ALARA (Ref. 7).

- Personnel working in high radiation areas are assigned self-reading dosimeters to allow determination of accumulated exposure at any time during the job. Self-reading dosimeters are checked during jobs to ensure that authorized exposures are not exceeded. Electronic personal dosimeters are used in some cases (Ref. 8).
- On jobs where general area radiation levels are unusually high (e.g., 1,000 mrem/hour to the skin or 200 mrem/hour to the whole body), a timekeeper keeps track of the total exposure time (Ref. 8).
- The number of personnel involved in actual radiation work is kept to the minimum number possible while still allowing the task to be performed safely and efficiently. If two people can accomplish a task in less than half the time it would take one person to do the same job, then exposure can be reduced through the use of the two people (Ref. 4).
- Remote observation, through shielded windows where available, and remote communication techniques, either through pre-established hand signals or radios, are used as practical. When dose rates are especially high, use of closed-circuit television is considered (Ref. 4).
- When applicable, mockup training for non-routine jobs that have the potential for high radiological consequences is conducted to familiarize the worker with the required task (Ref. 7).

7.5 RADIOLOGICAL PROTECTION TRAINING

Education and training of employees are essential to the success of radiation exposure reduction efforts. The educational program includes information on general and specific radiological considerations: time, distance, and shielding in minimizing exposure; radiation health effects; and dose limits. The appropriate level of radiological training is provided to each worker in the facility as well as to radiological protection personnel. WSRC Procedure Manual 5Q contains the requirements for training and frequency of training (Ref. 7). Training records are maintained by the site for each worker.

Standardized core courses and training materials are used to achieve site-wide consistency. In establishing training programs, the standardized DOE core courses are presented along with the addition of site-specific information. Standardized core course training material developed and maintained by DOE consists of lesson plans, view graphs, student handbooks, qualification standards, question banks and wallet-sized training certificates. The standardized core course training materials were based on the Radiological Worker Training Handbook (Ref. 9).

Surveying a limited subset of former students in the workplace verifies the effectiveness of the radiological control training. This verification is in addition to performance evaluations routinely performed by training departments. This evaluation includes observation of practical applications, discussions of the course material, and may include written examinations (Ref. 7).

Figure 7.5-1 illustrates typical control areas and required training.

7.5.1 PERSONNEL TRAINING

7.5.1.1 General Employee Radiological Training

Personnel who may routinely enter a Controlled Area and encounter radiological barriers, postings, or radioactive materials must receive General Employee Radiological Training (GERT). This training must be successfully completed prior to potential occupational radiation exposure. GERT is required for all employees as part of their SRS General Employee Training and Consolidated Annual Training.

Visitors who enter a Controlled Area must receive a radiological safety orientation as prescribed by WSRC Procedure Manual 5Q (Ref. 7). The orientation for continuously escorted individuals or groups is commensurate with the areas to be visited. Records of radiological safety orientations are maintained by the site.

7.5.1.2 Radiological Worker Training

Radiological Worker Training (RWT) consists of Radiological Worker (RW) I and RW II. The minimum training requirements for entry into specific areas are specified in WSRC Procedure Manual 5Q (Ref. 7).

Workers may challenge the RW I or II standardized core knowledge requirements by passing a comprehensive examination. However, if one attempt is unsuccessful, the worker must complete the entire standardized core RW I or II training. Challenges do not apply to the site-specific portions.

RW I training is not a prerequisite for RW II training. RW I and RW II training are self-contained courses. RW II training includes all of the requirements of RW I training and expands on the topic of hands-on work with radioactive materials. RW II training prepares the worker to deal with higher levels of radiation and radioactive contamination.

RW I and II retraining must be completed every two years. In the alternate year, when retraining is not performed, refresher training must be completed. Additional details for RWT are discussed in WSRC Procedure Manual 5Q (Ref. 7).

RADIOLOGICAL WORKER I AND II

Workers whose job assignments require access to Radiological Buffer Areas (RBA), Radiation Areas (RAs), and Radioactive Material Areas (RMAs), which contain sealed radioactive sources or labeled and packaged radioactive material, must complete DOE standardized core RW I training and site-specific RW I training before they will be permitted to enter these areas without a qualified escort.

Workers whose job assignments involve entry to High and Very High Radiation Areas, Contamination Areas (CAs), High Contamination Areas (HCAs), Soil Contamination Areas (to perform work that disturbs soil), and Airborne Radioactivity Areas (ARAs) must complete RW II training. Workers who have potential contact with radioactive particles or use gloveboxes with high contamination levels must also complete RW II training (Ref. 7).

7.5.1.3 Specialized Radiological Worker Training

Specialized RWT must be completed for nonroutine operations or work in areas with changing radiological conditions. This training is in addition to RW II training and is required for personnel planning, preparing, and performing jobs that have the potential for high radiological consequences. Such jobs may involve special containment devices, the use of mockups, and ALARA considerations (Ref. 7).

7.5.1.4 Radiological Control Technician Training

WSRC Procedure Manual Q1-1 defines and describes the selection, initial training, qualification, continuing training, and re-qualification requirements of the RCT program (Ref. 3). The procedure also defines the organization, planning, and administration of the training program and sets forth the responsibility, authority, and methods for implementing the RCT training program. Specific requirements necessary to qualify for the RCT initial training program are discussed in WSRC Procedure Manual Q1-1 (Ref. 3).

The initial training program is implemented to ensure that RCTs are trained to the performance requirements of the job. This is achieved by using a systematic approach to training. The RCT initial training program is conducted at the frequency established by the S&HO Department Manager. Three functional positions are defined for RCTs:

- RCT trainee
- Assistant RCT
- RCT

The RCT initial training program contains two elements, Phase I and Phase II. Phase I consists of classroom and on-the-job training for required technical knowledge and skills. Phase II consists of additional classroom and on-the-job training to develop the more advanced knowledge and skills applicable to all areas, and the unique knowledge and skills required at a specific area.

Continuing training includes retraining to build on prior fundamental knowledge and skills and periodic continuing training on industry/facility events, identified deficiencies, and changes that impact S&HO Department programs, procedures, and equipment. The continuing training program is designed to maintain and enhance technical proficiency, broaden the student's understanding of his or her job, or provide insight into industry events. The RCT continuing training program consists of two elements: periodic continuing training and retraining culminating in re-qualification. All qualified RCTs are advised to complete a minimum of 40 hours of continuing training per year (Ref. 3).

Additional experience/qualification requirements for the RCT are provided in Section 7.3.1.

7.5.1.5 Subcontracted Radiological Control Technician Training

Subcontracted RCTs must have the same knowledge and qualifications required of RCTs performing the same duties. Subcontracted RCTs who work at the facility for extended time periods (more than 6 months) receive continuing training commensurate with their assigned duties. This includes successful completion of an oral examination. Detailed information on subcontracted RCT training is given in WSRC Procedure Manual 5Q (Ref. 7).

7.5.1.6 Supervisor and Manager Training

WSRC Procedure Manual Q1-1 defines and describes the selection, initial training, qualification, continuing training, and re-qualification requirements of the Radiological Control (RC) First Line Supervisor (FLS) training program (Ref. 3). The procedure also defines the organization, planning, and administration of the training program and sets forth the responsibility, authority, and methods for implementing the FLS training program.

The initial training program is implemented to ensure that FLSs are trained in accordance with the performance requirements of the job. This is achieved by using a systematic approach to training. The FLS initial training program is conducted at the frequency established by the S&HO facility manager. The FLS training program consists of classroom and on-the-job training for required technical knowledge and skills, and supervisory skills training in accordance with human resource development programs.

Continuing training includes retraining to build on prior fundamental knowledge and skills and periodic continuing training on industry/facility events, identified deficiencies, and changes that impact S&HO programs, procedures, and equipment. The technical continuing training program is designed to maintain and enhance technical proficiency, broaden the student's understanding of his job, or provide insight into industry events. The FLS continuing training program consists of two technical elements, periodic continuing training and retraining culminating in re-qualification. All qualified FLSs must attend technical and supervisory skills continuing training (Ref. 3).

Line managers who manage, supervise, or provide oversight of radiological control programs shall be trained in the principles of the WSRC Radiological Control Manual (Ref. 7). Such training should be based on DOE standardized core course training materials supplemented by site-specific procedures and be completed by new personnel prior to formally assuming line supervision and management responsibilities. Incumbents should participate in continuing training. The continuing training should emphasize self-assessment and external evaluations including performance indicators, root causes, and lessons learned based on operational experience.

7.6 RADIATION EXPOSURE CONTROL

Radiation exposure in nuclear facilities involves stochastic and nonstochastic risks. Radiation-induced health effects, which do not have threshold doses, are referred to as stochastic effects. Nonstochastic effects can only be manifested if a threshold dose is exceeded. Exposure limits are established to minimize the potential stochastic effects (e.g., cancer) and prevent nonstochastic effects (e.g., skin erythema). External radiation exposure control includes limiting monthly, yearly, and lifetime whole-body exposures; organ exposures; skin and extremity exposures; exposures to the unborn; and exposures during emergencies. The site-level document for controlling radiation exposure is WSRC Procedure Manual 5Q (Ref. 7). Lower-tier implementing procedures are included in the 5Q derivative manuals such as WSRC Procedure Manual 5Q1.1 and WSRC Procedure Manual 5Q1.2 (Ref. 8, 10).

External radiation exposure control is accomplished by establishing administrative dose control levels well below DOE regulatory dose limits, monitoring personnel for external radiation exposure, by tracking exposures received, and identifying and controlling radiation sources. Exposure tracking systems inform personnel and their supervisors of exposures received and are used to plan radiological work. Administrative Control Levels (ACLs) and exposure tracking systems are management tools to help ensure that individual and collective exposures are minimized. Managers in all departments, as well as all workers, are responsible for controlling and minimizing external radiation exposures.

Internal radiation exposure control is accomplished by establishing ACLs, identifying and controlling sources or potential sources of airborne radioactivity, maximizing the use of engineered controls, applying respiratory protection where appropriate, and monitoring workers for internal radioactivity.

See Section 7.10 for discussions of expected occupational radiation exposures.

7.6.1 ADMINISTRATIVE LIMITS

The site's objective is to maintain personnel radiation exposure well below regulatory limits. To accomplish this objective, challenging ACLs are established at levels below the regulatory limits to administratively control and help reduce individual and collective radiation dose. These control levels are multi-tiered, and increasing levels of authority are required to approve higher ACLs (Ref. 7).

7.6.1.1 Administrative Control Provisions

WSRC Procedure Manual 5Q establishes practices for the conduct of radiological control activities at SRS (Ref. 7). The radiation safety program is further documented and implemented in procedure manuals and operating procedures specific for each activity. An operating procedure is defined as a detailed, stepwise instruction formally issued for a frequently performed job. When written to cover work in an RA, CA, etc., a procedure must include

instructions for radiation and contamination control or reference appropriate radiological controls procedures.

Routine and nonroutine site operations are covered by written procedures. Whenever an operation involves potential radiological hazards, RC personnel assist the responsible site department or subgroup in preparing the work procedure.

7.6.1.2 Administrative Control Levels

A DOE ACL of 2,000 mrem/year per person is established for all DOE activities. Approval by the appropriate DOE Secretarial Official or designee shall be required prior to allowing a person to exceed 2,000 mrem/year (Ref. 7).

WSRC establishes an SRS ACL for an individual based on evaluation of historical and projected radiation exposures, workload, and mission. This control level is reevaluated annually but must be maintained more restrictively than the DOE ACL. No person is allowed to exceed the SRS ACL without the prior approval of the WSRC President.

7.6.1.3 Radiological Worker Dose Limits

Dose limits set by DOE provide the upper bounds for exposure of operating personnel. A summary of annual dose limits for occupational workers is provided in Table 7.6-1 (Ref. 7).

Proposed use of the Planned Special Exposure, as allowed by the radiological protection S/RIDs, is applied only in extraordinary situations as specified in WSRC Procedure Manual 5Q (Ref. 7). Special authorization must be obtained for radiation workers who have received an unplanned dose in excess of the limits in Table 7.6-1 to return to work in Radiological Areas.

The manager of the DOE-Savannah River Operations Office must also review and approve resumption of operations of an SRS facility following an emergency or accidental exposure in excess of the limits in Table 7.6-1. The WSRC President must verify that the conditions under which the emergency or accidental exposures were received have been eliminated (Ref. 7).

A Special Control Level (SCL) for annual occupational exposure is established for each person with a lifetime occupational dose exceeding N roentgen equivalent man (rem), where N is the age of the person in years. The SCL shall not exceed 1 rem and should allow the person's lifetime occupational dose to approach N rem as additional occupational exposure is received (Ref. 7).

7.6.1.4 Embryo/Fetus Dose Limits

After a female radiation worker voluntarily notifies her employer in writing that she is pregnant, for the purpose of fetal/embryo dose protection, she is considered a declared pregnant worker. The employee's SRS organization must provide the option of a mutually agreeable assignment of

work tasks, without loss of pay or promotional opportunity, such that further occupational radiation exposure is unlikely.

For a declared pregnant worker who continues working as a radiation worker:

- The dose limit for the embryo/fetus from conception to birth (entire gestation period) is 500 mrem.
- Efforts are made to avoid exceeding 50 mrem per month to the pregnant worker.

If the dose to the embryo/fetus is determined to have already exceeded 500 mrem when a worker notifies her employer of her pregnancy, the worker cannot be assigned to tasks where additional occupational radiation exposure is likely during the remainder of the gestation period (Ref. 7).

7.6.1.5 Special Control Levels

Certain situations require lower individualized exposure control levels. SCLs are established with the advice of S&HO, Medical, Human Resources, and/or Legal, as appropriate. WSRC and other SRS organizations must be attentive to special circumstances of employees, such as those undergoing radiation therapy, and establish SCLs as appropriate (Ref. 10).

7.6.1.6 Planned Special Exposures

Certain employees have specialized skills important to facility and public safety, and for this and other reasons, it is recognized that unusual conditions can arise in which higher-than-normal doses can be justified. Under approved, well-justified, well-planned, well-controlled, highly infrequent and unusual conditions, operating management are permitted to allow exposure of specified individuals to doses exceeding the 5 rem/year limit. The planned special exposure does not apply to emergency conditions (Ref. 7).

The total dose from planned special exposures for an employee during any given year cannot exceed 5 rem. This is in addition to the 5 rem annual occupational exposure dose limit. Thus, apart from emergency situations, the maximum annual dose that an employee could receive would be 10 rem. An employee could receive no more than 25 rem of planned special exposures from DOE and non-DOE operations during his/her career. Every planned special exposure must be approved in advance by the DOE and requires the informed consent of the employee involved. Documentation of each planned special exposure is required to be recorded in the employee's occupational exposure file and provided to the employee.

The procedure for conducting a planned special exposure is contained in WSRC Procedure Manual 5Q (Ref. 7).

7.6.1.7 Emergency and Accidental Exposures

In extremely rare cases, emergency exposure to radiation may be necessary to rescue personnel or to protect major property. Dose limits and criteria to perform emergency actions are prescribed in WSRC Procedure Manual 5Q for volunteers performing rescue operation, protection of large population, or protection of major property (Ref. 7).

Under the guidance of WSRC-SCD-7, when the Emergency Operations Facility is activated, the WSRC Emergency Director (ED) is responsible for command and control of emergency response action during the emergency (Ref. 11). Therefore, the responsibility for these decisions shifts upward from the Facility Emergency Coordinator (FEC) to the WSRC ED.

The doses allowed would be in addition to those allowed for normal operating conditions. The FEC is not required to determine how much exposure a worker had already received during the current year.

Rescue action that might involve substantial personnel risk and emergency situation radiation exposures in excess of 10 rem must be performed only by a trained response group (Ref. 7, 8).

7.6.2 RADIOLOGICAL PRACTICES

7.6.2.1 Personnel Protective Equipment and Clothing

Clothing for protection against radioactive contamination includes such items as coveralls, hoods, shoe covers, rubber and cotton gloves, laboratory coats, and other specialized articles used for particular tasks. S&HO is responsible for ensuring that all clothing meets minimum standards and that the clothing provides the protection required.

Personal protective equipment and clothing are selected as prescribed by the controlling RWP. Training, qualification requirements, and procedures for donning and removal of personnel protective equipment and clothing are discussed in WSRC Procedure Manual 5Q1.1 and WSRC Procedure Manual 5Q1.2 (Ref. 8, 10). General guidelines for protective clothing selection and use are provided in WSRC Procedure Manual 5Q (Ref. 7).

All clothing is segregated and laundered depending on contamination level and type. Following laundering, the clothing is monitored and returned to the proper facility.

A variety of respiratory protective equipment can be used depending on the expected airborne contamination levels, work to be performed, and the facial features of the individual involved. Guidelines exist that specify the type of respiratory device to be used. Refer to Chapter 8 of this SAR for a discussion of the respiratory protection program.

7.6.2.2 Radiological Work Permits

The RWP is the primary administrative mechanism used to establish radiological controls for intended work activities. The RWP informs workers of area radiological conditions and entry requirements, and provides a mechanism to relate worker exposure to specific work activities (Ref. 7). The RWP program is documented in WSRC Procedure Manual 5Q1.1 (Ref. 8).

Two types of RWPs are currently implemented at SRS. The first type is the general RWP (commonly known as the standing RWP). The general RWP is used to control routine or repetitive activities, such as surveillance tours, inspections, or minor work activities in areas with well-characterized and stable radiological conditions, when work does not involve elevated or complex radiological conditions (e.g., entry into an RBA, visitor access, facility observations, routine RC surveillance activities, and operator rounds). The general RWP remains in effect for a maximum of 1 year.

The second type is the job-specific RWP, which is used to control nonroutine operations and complex work activities, work in areas with changing radiological conditions, or routine and repetitive activities when work involves elevated or complex radiological conditions (e.g., process system line breaks, jobs performed via special written procedures, and any job that has required a pre-job ALARA review). The RWP remains in effect only for the duration of the job (Ref. 8).

The RWP includes the following information (Ref. 7):

- Description of work
- Work area radiological conditions
- Dosimetry requirements
- Pre-job briefing requirements, as applicable
- Training requirements for entry
- Protective clothing and respiratory protection requirements
- RC coverage requirements and stay time controls, as applicable
- Limiting radiological conditions that may void the RWP
- Special dose or contamination reduction considerations
- Special personnel frisking considerations
- Technical work document number, as applicable
- Unique identifying number
- Date of issue and expiration
- Authorizing signatures

Radiological surveys are routinely reviewed to evaluate adequacy of RWP requirements. RWPs are updated if radiological conditions change to the extent that protective requirements need modification.

RWPs are posted or made available at the access point to the applicable radiological work area. Workers must sign that they have read, understand, and will comply with the RWP requirements prior to initial entry to the area and after any revision to the RWP (Ref. 7). In addition, before entering the work area, a worker must ensure he/she has completed the appropriate training, whole-body count, and has submitted the proper bioassay samples (Ref. 8).

Alternative formal mechanisms, such as written procedures or test authorizations, may be used in lieu of an RWP as the administrative control over radiological work activities. If an alternative mechanism is used, it must meet the requirements set forth in the WSRC Procedure Manual 5Q and be approved by the RCO Facility Manager (Ref. 7).

7.6.2.3 Planning of Radiological Work

WSRC Procedure Manual 5Q establishes practices for the conduct of radiological control activities at SRS (Ref. 7). Site personnel, including FLSs and S&HO personnel, are given the responsibility and authority to ensure that work involving radiological protection is performed correctly. Technical work documents and RWPs are used to specify conduct of radiological work activities. Technical work documents encompass documents such as procedures, work packages, and job or test authorizations used to control hands-on work with radioactive materials. The RWP is the primary administrative mechanism used for planning and controlling radiological work and for informing the worker of the radiological conditions and entry requirements. Refer to Section 7.6.2.2 for additional information regarding RWPs (Ref. 7, 8).

Technical requirements for the conduct of work incorporate radiological protection control criteria to ensure safety and maintain radiation exposures ALARA. To accomplish this, the design and planning processes include radiological considerations in the early planning stages. WSRC Procedure Manual 5Q contains a checklist to be used for reducing occupational radiation exposure (Ref. 7).

The following are examples of established trigger levels that require formal radiological review of the nonroutine or complex work activities involved (Ref. 7):

- Estimated individual or collective dose greater than pre-established values
- Predicted airborne radioactivity concentrations in excess of pre-established values
- Work area removable contamination greater than 100 times the values in WSRC Procedure Manual 5Q
- Entry into areas where dose rates exceed 1 rem/hour
- Potential radioactive releases to the environment

Tasks with the potential to exceed the trigger levels must undergo a formal, documented radiological or pre-job ALARA review. The purpose of conducting ALARA reviews is to analyze the proposed work and determine what actions can be taken to minimize worker radiation doses. Such actions include pre-job decontamination, flushing of process lines, and additional temporary or permanent shielding (Ref. 4).

For review and planning of major tasks involving an estimated cumulative exposure greater than 10 person-rem, a detailed and documented cost-benefit evaluation is performed by S&HO HPT (Ref. 8).

In addition to the above ALARA review, pre-job briefings must be held prior to conducting work that is anticipated to exceed the trigger levels. At a minimum, the pre-job briefing includes the following (Ref. 7):

- Scope of work to be performed
- Radiological conditions of the workplace
- Procedural and RWP requirements
- Special radiological control requirements
- Radiologically limiting conditions, such as contamination or radiation levels that may void the RWP
- RC Hold Points
- Communications and coordination with other groups
- Provisions for housekeeping and final cleanup
- Emergency response provision to abnormal and/or degrading conditions

Pre-job briefings are conducted and documented by the cognizant work supervisor.

7.6.2.4 Infrequent or First-Time Activities

For activities with significant dose implications (exceeding pre-established trigger levels) that are infrequently conducted (less than annually) or that represent first-time operations, planning includes the following (Ref. 7):

- A formal radiological or ALARA review
- Senior management review directed toward anticipation of concerns and emphasis and specification of protective measures
- Review and approval by the cognizant A/RAC
- Enhanced line and S&HO management oversight during the initiation and conduct of the work

7.6.2.5 Review of Work in Progress

As part of their normal work review, work supervisors periodically review ongoing jobs to ensure that prescribed radiological controls are being implemented. S&HO personnel are required to conduct frequent walkdowns of the workplace to review the adequacy of radiological work practices, posting, and area controls. For jobs in which a pre-job dose estimate was made, S&HO, in conjunction with line management, must periodically monitor collective dose accumulation and compare it with the pre-job estimate. Differences must be reviewed to identify causes and assess the need for corrective actions (Ref. 7).

7.6.2.6 Stoppage of Radiological Work

RCTs, their supervisors, line management, and any other designated personnel have the authority and responsibility to stop radiological work activities for any of the following reasons (Ref. 12):

- Inadequate radiological controls
- Radiological controls not being implemented
- RC Hold Point not being satisfied

Once radiological work has been stopped, it cannot be resumed until proper radiological control has been reestablished. Resumption of radiological work requires the approval of the line manager responsible for the work and the S&HO Manager (Ref. 7).

7.6.2.7 Conduct of Critiques

During the conduct of radiological work and the handling of radioactive materials, abnormal events may occur that could indicate a weakness or area of programmatic breakdown of radiological controls. Prompt, consistent gathering of facts related to such events is required to satisfy reporting and investigation requirements and to formulate corrective actions to prevent recurrence. In addition, successful performance or completion of unique activities is evaluated to identify and incorporate appropriate lessons learned.

Critiques are meetings of the personnel knowledgeable about an event (either a success or an abnormal event) to document a chronological listing of the facts. The purpose of the critique is not to assign blame, but to establish and record the facts. Critiques of abnormal events are conducted per the requirements in WSRC Procedure Manual 5Q and WSRC Procedure Manual 2S (Ref. 7, 13).

7.6.2.8 Post-Job Review

Following completion of radiological work, a post-job ALARA review may be required. The following are radiological circumstances that would require this review (Ref. 7):

- All jobs requiring a pre-job ALARA review
- When the pre-job estimated cumulative dose is less than 1.0 person-rem and the actual cumulative dose exceeds 1.0 person-rem
- When the actual cumulative dose exceeds the pre-job estimated cumulative dose by greater than 25 percent and the total cumulative dose is at least 0.5 person-rem
- After completion of nonroutine radiological work

Lessons Learned are available from post-job reviews and reports of past events onsite and at other facilities. S&HO, in conjunction with line management, evaluates Lessons Learned, provides prompt distribution, and incorporates the lessons into the radiological control program, the radiological training program, and related operations (Ref. 7).

7.6.2.9 Radiological Area Boundaries, Posting and Controls

Requirements for controlling personnel exposure to radiation, airborne radioactivity, and surface contamination are established in WSRC Procedure Manual 5Q (Ref. 7). Any area where radiation and/or radioactive contamination levels are above specified values must have access controls commensurate with the level of the hazard. These controls may include signs and barricades, control devices on entrances, locks, alarms, direct surveillance, or administrative controls. In order to make employees aware of radiation and contamination conditions, it is required that signs be clearly posted to identify those areas that are controlled to manage potential exposures and those areas where radiation levels exceed certain values. Containers of radioactive material and radioactive items are required to be properly labeled to provide information needed for purposes of radiation protection and the prevention of inadvertent transfer to locations outside of radiological areas (Ref. 7).

Figure 7.5-1 illustrates typical control areas and required training.

IDENTIFICATION OF BOUNDARIES

Posting Requirements

Radiological posting is used to alert personnel to the presence of radiation and radioactive materials and to aid them in minimizing exposures and preventing the spread of contamination. In order to limit the spread of contamination, the following are examples of general posting requirements (Ref. 7):

- Signs containing the standard radiation symbol colored magenta on a yellow

background. Lettering is either magenta (preferred color) or black. Where practicable, standardized signs are used.

- Radiological postings are displayed only to signify actual or potential radiological conditions. Signs used for training, such as "For Training Purposes Only," are clearly marked.
- Posted areas are maintained as small as practicable for efficiency.
- Postings are maintained in a legible condition and updated based upon the results of the most recent surveys.
- If more than one radiological condition (such as contamination and high radiation) exists in the same area, each condition is identified.
- In areas of ongoing work activities, the dose rate and contamination level or range of each is included on or in conjunction with each posting as applicable.
- Entrance points to areas of ongoing work activities controlled for radiological purposes state basic entry requirements, such as dosimetry, RWP, and respirator required.
- Rope, tape, chain, and similar barriers used to designate the boundaries of posted areas are yellow and magenta in color.
- Physical barriers are placed so that they are clearly visible from all directions and at various elevations. They are not easily walked over or under, except at identified access points. These barriers are set up such that they do not impede the intended use of emergency exits or evacuation routes.
- Posting of doors is such that the postings remain visible when doors are open or closed.
- A radiological posting that signifies the presence of an intermittent radiological condition includes a statement specifying when the radiation is present, such as "RADIATION AREA WHEN RED LIGHT IS ON."

Controlled Area

A Controlled Area is any area to which access is managed in order to protect individuals from exposure to radiation and/or radioactive materials. Individuals who enter only the Controlled Area without entering radiological areas are not expected to receive a total effective dose equivalent of more than 100 mrem in a year (Ref. 8).

Radiological Buffer Area

RBAs are established within Controlled Areas. RBAs provide a second boundary to minimize the spread of contamination and to limit doses to general employees who have not been trained as radiological workers (Ref. 7). WSRC Procedure Manual 5Q1.1 provides facility-specific

radiation and contamination guides, instructions, and protective clothing and equipment requirements for an RBA at SRS facilities (Ref. 8).

The perimeter of an RBA is posted on all accessible sides to prevent inadvertent intrusion by personnel. A sign indicating that all personnel must monitor prior to exiting the RBA is prominently displayed at each exit. A personnel monitoring device is located at all egress points of RBAs that enclose contamination or airborne radioactivity areas. Where installation of personnel monitoring devices is not feasible, instructions to monitor at a nearby location are posted, or RCTs may monitor personnel with portable survey instruments (Ref. 8).

RADIOLOGICAL AREAS

A radiological area is an area within an RBA in which an individual can be exposed to increased individual hazards greater than those in the "buffer" portion of an RBA. Radiological areas identify the type of radiological hazard present in an area. The following sections discuss various radiological areas at SRS.

Radiation Area

An RA is any area within an RBA in which an individual can receive a dose equivalent greater than 5 mrem, but less than or equal to 100 mrem in 1 hour at 30 centimeter (cm) from the source of radiation, or any surface through which the radiation penetrates. A perimeter boundary must be established to alert personnel to the presence of external radiation. The perimeter of the RA is posted in accordance with the posting requirements (Ref. 7).

High Radiation Area

A High Radiation Area (HRA) is any area within an RBA in which an individual can receive a dose equivalent greater than 100 mrem in 1 hour at 30 cm, but less than or equal to 500 radiation absorbed dose (rad) in 1 hour at 100 cm from the source of radiation, or the surface through which the radiation penetrates. A perimeter boundary must be established to alert personnel to the presence of external radiation. This can include locked or guarded entrances or other positive access control features. The perimeter of the HRA is posted in accordance with the posting requirements (Ref. 7).

Very High Radiation Area

A Very High Radiation Area (VHRA) is any area within an RBA in which an individual can receive a dose equivalent of 500 rad or greater in 1 hour at 100 cm from the radiation source, or the surface from which the radiation penetrates. A perimeter boundary must be established to prevent inadvertent intrusion by personnel. Access control for these areas must include one or more of the features listed in WSRC Procedure Manual 5Q (Ref. 7). Entry alarms or other

positive control devices may also be used. The posting of a VHRA is in accordance with the posting requirements (Ref. 8).

Airborne Radioactivity Area

An ARA is any area in which the airborne concentration of radioactive material exceeds 10 percent of the Derived Air Concentration (DAC). The DAC values for radionuclides are provided in the radiological protection S/RIDs. A perimeter boundary must be established to prevent inadvertent intrusion by personnel. The perimeter of the ARA is posted in accordance with the posting requirements. When establishing the perimeter boundary for these areas, it is important to remember that air currents can carry airborne radioactivity across open boundaries; therefore, the boundary must be positioned so that personnel located outside the posted area will not be exposed to an airborne concentration that is above established limits for airborne radionuclides (Ref. 8).

To minimize the potential for uptakes of radionuclides, S&HO personnel may also post additional areas as ARAs (regardless of actual levels of airborne concentration) if there is a significant potential that radioactive contaminants in the areas may become airborne in sufficient quantities to exceed 10 percent of the DAC (Ref. 8).

Contamination Area and High Contamination Area

CAs and HCAs are areas in which removable or total radioactivity exceeds levels prescribed in WSRC Procedure Manual 5Q (Ref. 7). A perimeter boundary must be established to prevent inadvertent intrusion by personnel. The perimeters of the CAs and HCAs are posted in accordance with the posting requirements.

Each entrance/exit to a CA or HCA has a step-off pad and appropriate containers for depositing used protective clothing and contaminated waste. A personnel monitoring device is located at or near each exit from a CA or HCA. Where installation of personnel monitoring devices is not feasible, instructions to monitor at a nearby location are posted (Ref. 8).

An RWP is required for entry into an HCA.

Radioactive Material Area

Areas where sealed, well-contained, or low levels of radioactive materials are used, handled, or stored, but which would otherwise not require posting, are posted: "CAUTION, RADIOACTIVE MATERIAL AREA." RMAs are generally located within Controlled Areas. Posting for RMAs is not required when the radioactive material is inside a CA or ARA (Ref. 7).

7.6.2.10 Entry and Exit Control

ENTRY AND EXIT REQUIREMENTS

Access to radiologically posted areas is controlled to minimize radiation exposure, the spread of radiological contamination, and personnel contamination. Typical entry requirements for radiologically posted areas are appropriate training, personnel dosimetry, and worker's signature on the RWP. Entry and exit requirements for controlled and radiological access are provided in WSRC Procedure Manual 5Q (Ref. 7).

Physical controls are used to prevent inadvertent or unauthorized access to HRAs and VHRAs. Typical control features are described in WSRC Procedure Manual 5Q (Ref. 7). Weekly inspections of the physical access controls to HRAs and VHRAs are conducted by the facility custodian to verify controls are adequate to prevent unauthorized entry. Physical access controls over HRAs and VHRAs are established in a way that does not prevent a person from leaving the area (Ref. 7). See Section 7.6.2.9 for additional information on HRAs and VHRAs.

Requirements for entry into RMAs, where whole-body dose rates exceed 5 mrem/hour or removable contamination levels exceed those given in WSRC Procedure Manual 5Q, are equivalent to those for entry into RAs and CAs (Ref. 7).

All personnel and equipment leaving CAs, HCAs, or ARAs must be monitored for surface contamination. Contamination levels higher than the limiting values are not allowed outside of radiological areas except in the case of fixed contamination under prescribed conditions. Special control devices such as step-off pads and contamination monitoring equipment are discussed in WSRC Procedure Manual 5Q (Ref. 7).

Visitor Entry Requirements

Requirements for visitors have been established for entry into Controlled Areas at SRS and are described in WSRC Procedure Manual 5Q (Ref. 7).

7.6.2.11 Permanent Shielding

Permanent shielding used to control radiation exposure is discussed in Section 7.4.3. See facility-specific SARs for more information on permanent shielding specific to the facility, as appropriate.

7.6.2.12 Contamination Control

Contamination control minimizes the potential for worker internal exposure and the spread of contamination. Contamination should be controlled at the source, and areas that become contaminated should be promptly decontaminated. Contamination levels caused by ongoing

work shall be monitored and maintained ALARA. Reducing the size and scope of existing CAs in a facility generally improves productivity.

Personnel should not be exposed unnecessarily to contamination and airborne radioactivity. Use of engineering and administrative controls should be evaluated before allowing personnel to work with or without respiratory protection. When engineering and administrative controls have been applied and the potential for airborne radioactivity still exists, respiratory protection should be used to limit internal exposures. The selection of respiratory protection equipment includes considerations of worker safety, comfort and efficiency. Positive pressure respiratory protection devices are recommended whenever practical to alleviate fatigue and increase comfort. When feasible, work should be planned to avoid the routine use of respiratory protection devices. (Ref. 4)

In addition, eating, drinking, chewing, and smoking are prohibited inside CAs, HCAs or ARAs to minimize the chance of ingesting contaminated materials. Under certain circumstances, such as high risk of heat stress, drinking may be permitted inside these areas (Ref. 7).

7.6.3 DOSIMETRY

To ensure that the radiation and contamination control programs are adequately protecting both occupational workers and visitors, a dosimetry program has been established at SRS. The S&HO Department has responsibility for providing a primary means of measuring external radiation exposure and maintaining a permanent radiation history file for employees at SRS. All activities associated with the External Dosimetry Program (EDP) are maintained in compliance with DOE requirements.

The SRS External Dosimetry Technical Basis Manual provides the technical basis for the EDP, including measures to ensure the validity and quality of external dosimetry results (Ref. 14). Through this comprehensive program, the performance of radiological control measures is evaluated, thus helping to ensure that administrative dose control limits are not exceeded.

7.6.3.1 External Dosimetry

At SRS, several types of radiation dosimeters are used to determine dose equivalent from external radiation exposures. Most dosimetry used at SRS is based on the property of some materials to become thermoluminescent following exposure to ionizing radiation. By recording the amount of light emitted as the materials are heated, the amount and type of radiation to which the dosimeter was exposed can be estimated (Ref. 14).

PERSONNEL THERMOLUMINESCENT DOSIMETER BADGE

A whole-body Thermoluminescent Dosimeter (TLD) must be worn by all personnel entering RBAs or handling radioactive materials. This dosimeter is the primary device used to measure beta-gamma radiation exposure. Additional details on the use of the TLDs, such as control and

issuance, return, or lost or damaged TLDs, are provided in WSRC Procedure Manual 5Q1.2 (Ref. 10).

PERSONNEL NEUTRON BADGE

A Thermoluminescent Neutron Dosimeter (TLND) is required if personnel are likely to exceed 100 mrem from neutrons during the calendar year. Additional details on the use of TLNDs, such as control and issuance, handling and storage, or lost or damaged TLNDs, are provided in WSRC Procedure Manual 5Q1.2 (Ref. 10).

SPECIAL DOSIMETRY FOR EXTREMITIES

The S&HO Department is responsible for determining the radiation exposure rates for each type of radiation present at worksites. Extremity radiation dose rates are determined prior to the start of work where such rates may be the time-limiting factor.

Extremity TLDs, or finger rings, or wristbands are needed if the extremities are expected to receive more exposure than the whole body. Hands are normally the extremity of interest, but under some circumstances the forearm, feet, and ankles may also be of concern. Positioning of extremity dosimeters is specified by the S&HO Department to ensure that the TLD is facing the exposure source. When required, use of extremity dosimeters should be specified in the RWPs (Ref. 6). Additional details on the use of extremity TLDs, such as control and issuance, handling and storage, or lost or damaged extremity TLDs, are provided in the WSRC Procedure Manual 5Q1.2 (Ref. 10).

SELF READING POCKET DOSIMETER

The Self Reading Pocket Dosimeter (SRD) or equivalent shall be worn by personnel when entering RAs where exposure could exceed 10 percent of an ACL from external radiation in one work day, when entering/working in HRAs or VHRAs, or when required by an RWP. Routinely, SRDs are worn under plastic suits, unless dose readings during the job are necessary, and on the inside of protective clothing other than plastic suits unless designated by S&HO. The SRD is useful for personnel who need to maintain a day-to-day cumulative record of their exposure and can be used as an aid in controlling radiation exposure during work. Daily, each worker records his/her estimated dose on an Employee Radiation Dose Record (Ref. 3). Additional details on the use of SRDs, such as control and issuance, handling and storage, or lost or damaged SRDs, are provided in WSRC Procedure Manual 5Q1.2 (Ref. 10).

7.6.3.2 Internal Dosimetry

Internal dosimetry at SRS is accomplished by in-vivo and in-vitro bioassays and subsequent dose assessment. Personnel whose routine duties may involve exposure to surface or airborne contamination or to radionuclides readily absorbed through the skin are considered for an

appropriate bioassay program as described below. The basis for the methods and frequency of these bioassay programs is documented in the SRS Internal Dosimetry Technical Basis Manual (Ref. 15).

CRITERIA FOR REQUIRING BIOASSAYS

Personnel who enter RBAs must participate in a bioassay program when they are likely to receive intakes resulting in a committed effective dose equivalent of 100 mrem or more. Personnel assigned to a bioassay program must submit urine samples and receive whole-body and/or chest counts at the prescribed frequency (Ref. 8). Personnel are required to participate in a follow-up bioassay monitoring program if their bioassay results indicate an intake in the current year with a committed effective dose equivalent of 100 mrem or more (Ref. 7).

Bioassay programs are described in the following sections.

Routine Sampling

Routine bioassay programs consist of both in-vivo and in-vitro sampling. Adherence to bioassay schedules is essential to maintaining the integrity of the bioassay program. The routine sampling program consists of the following (Ref. 8):

- Performing whole-body counts and chest counts
- Collecting urine samples and analyzing for facility-specific radionuclides
- Baselining employees if required
- Receiving closeout bioassay samples from employees whose movement from one facility to another requires a change in their bioassay schedule
- Receiving appropriate termination samples and/or in-vivo counts from individuals ending their affiliation with SRS

Non-Routine Sampling

When jobs with the potential for unknown radiological conditions to occur or unusual radionuclides to be present are undertaken, a non-routine, job-specific bioassay program should be considered. In such cases, an in-vitro sample and/or in-vivo count may be required before beginning work and again when work is completed. Such a sampling program is at the discretion of S&HO supervision and is noted on the RWP for the task (Ref. 8).

Special bioassay programs must be performed if an intake of radioactive materials is suspected. Examples of cases when an intake should be suspected are described in WSRC Procedure Manual 5Q1.1 (Ref. 8). In the event that radioactive material is detected by in-vivo or in-vitro bioassay, a follow-up bioassay monitoring program is conducted.

Visitor Bioassay

Visitors may be required to participate in in-vivo and in-vitro bioassay programs in conjunction with a visit to an RBA. The host organization is responsible for arranging for the appropriate samples and other entry requirements for their visitors. Prior to such a visit, S&HO supervision should be notified (Ref. 8).

ROUTINE BIOASSAY PROGRAM ASSIGNMENT

Routine in-vivo and in-vitro bioassay programs are designed to verify that workers who enter RBAs and have the potential to receive intakes resulting in a committed effective dose equivalent of 100 mrem or more in a year have not received intakes of radioactive material due to an undetected incident. Personnel who wear respiratory protection or who work in posted CAs or ARAs must be sampled for the radionuclide to which they are potentially exposed, either through the routine sampling program or the nonroutine, job-specific sampling program.

The type of work performed can be broken down into four distinct categories with respect to risk of intake (Ref. 8):

- Category I (highest probability of intake)
- Category II (lower probability of intake)
- Category III (little risk of intake)
- Category IV (negligible risk of intake)

The higher probability of intake requires more frequent sampling rates. See the SRS Internal Dosimetry Technical Basis Manual for more information on internal dosimetry and bioassay sampling (Ref. 15).

7.6.3.3 Combining Internal and External Dosimetry Results

The effective dose equivalent is the sum of the weighted dose equivalent to all significantly irradiated organs. The annual effective dose equivalent to an individual is determined by adding the effective dose equivalent from both external and internal irradiation received in a calendar year. This information is provided to DOE in the Annual Radiation Dose Summary (Radiation Exposure Information and Reporting System [REIRS] report) (Ref. 15).

The exposure of the whole body to penetrating radiation is estimated by combining the following (Ref. 7):

- Radiation dose determined from TLD readings; TLDs are worn by all personnel who have access to areas that are potentially subject to exposure to ionizing radiation in excess of 100 mrem per year
- Neutron radiation dose as determined by a TLND

- Dose received from radionuclide uptakes, as determined by internal dosimetry

7.6.3.4 External and Internal Dosimetry Accreditation

The radiological protection S/RIDs specify the requirements for accreditation of personnel external dosimetry monitoring programs by the DOE Laboratory Accreditation Program (DOELAP) (Ref. 1). DOELAP involves a rigorous dosimetry testing program as well as a comprehensive onsite assessment of the dosimetry processing facility, QA, technical capability, and documentation.

WSRC Procedure Manual 5Q2.1 specifies the method for preparing and submitting TLDs for DOELAP performance testing, processing and reporting TLD results to the DOELAP Testing Laboratory, and conducting blind spike tests of TLDs and finger rings to test the overall TLD processing system performance (Ref. 16).

The SRS dosimetry program is required to be re-accredited at the DOE prescribed frequency (Ref. 16).

There are currently no specific requirements for accreditation of the internal dosimetry program, however a DOELAP Bioassay Pilot based on American National Standards Institute (ANSI) N13.30 is underway. The technical basis for the methods and frequency of the bioassay programs is documented in the SRS Internal Dosimetry Technical Basis Manual (Ref. 15). The requirements and recommendations of the DOE Radiological Control Manual, the DOE Internal Dosimetry Implementation Guide, and ANSI N13.30 (draft) are addressed in the SRS Internal Dosimetry Technical Basis Manual (Ref. 15).

7.6.3.5 Accident Dosimetry

A Criticality Neutron Dosimeter (CND) is used at SRS to assess neutron doses from prompt neutrons. CNDs are required to be worn (between the neck and waist) by personnel assigned to facilities that handle and store fissionable materials such as uranium and plutonium in quantities that would require the installation of Nuclear Incident Monitors. In the event of a criticality incident, the CNDs and any other dosimetry are collected at the rally point by S&HO (Ref. 14).

In the event of a major radiological accident or highly unusual exposure condition, SRS has implemented the requirement for all personnel to have a Teflon lithium fluoride TLD disc and an indium activation foil incorporated between the laminated layers of the security photobadge. The Teflon TLD disc is processed only in the event that an individual is suspected of having received an exposure in excess of 10 rad. The indium foil is used to verify whether an individual may have received an excessive exposure to neutrons (Ref. 14). Additional details on the use of CNDs, such as control and issuance, handling and storage, or lost or damaged CNDs, are provided in WSRC Procedure Manual 5Q and WSRC Procedure Manual 5Q1.2 (Ref. 7, 10).

7.6.3.6 Reports

WSRC Procedure Manual 5Q2.1 describes the process for preparing and distributing reports concerning radiation exposures (Ref. 16). It also describes a method by which personnel radiation exposure information may be released by the S&HO Department (Ref. 16). In addition, WSRC Procedure Manual 5Q requires the establishment of a radiological records management program (Ref. 7). This program ensures that auditable records and reports are controlled through the stages of creation, distribution, use, arrangement, storage, retrieval, media conversion (if applicable) and disposition (Ref. 7).

Radiation dose records are maintained for all WSRC, federal, and subcontractor employees who are part of the personnel dosimetry program at SRS. Radiation dose records contain information sufficient to identify each person, including social security or employee number. External dose records include the following (Ref. 7):

- Extremity, skin, eye and whole-body dose results measured with personnel dosimeters, including all multiple dosimeter badging results
- Evaluations resulting from anomalous dose results such as unexpected high or low doses
- Dose reconstructions from lost, damaged, or contaminated dosimeters
- Evaluations of non-uniform radiation doses

In accordance with WSRC Procedure Manual 5Q, the Annual Radiation Exposure Report must be submitted to DOE for the preceding calendar year for DOE and DOE contractor radiation workers and for non-employee occupational workers at SRS (Ref. 1, 7).

For visitors entering an area where radiation monitoring is required, the following records are maintained (Ref. 7):

- Documentation of completion of Radiological Orientation
- Radiation dose records, including zero dose

In accordance with the radiological protection S/RIDs, all external and internal radiation exposures recorded for visitors during the period of their visit to SRS must be reported to the visitor's employer (or to the visitor if he or she has no employer) within a period of 30 days after the date of the visit or within 30 days after the visitor's exposure has been determined, whichever is later (Ref. 1).

All recorded external and internal radiation measurements (including zero results) for DOE Headquarters employees while at SRS must be reported to the System Safety Development Center, EG&G Idaho, Inc., on DOE Form F5480.7 within 30 days after the individual's date of visit or within 30 days after their exposure has been assessed, whichever is later (Ref. 12).

Personnel who are monitored by the personnel dosimetry program are provided an annual report of their dose (Ref. 10). In accordance with the radiological protection S/RIDS, current radiation exposure information must be made available to all employees upon request.

In accordance with the radiological protection S/RIDs, all recorded external and internal radiation exposures compiled for employees during their employment at SRS must be reported to terminating employees (Ref. 1). This report is issued within 90 days of the last day of employment (Ref. 17).

An annual internal exposure report for all monitored individuals is required per the radiological protection S/RIDs (Ref. 1). DOE Form 5480.7 must be completed for any uptake of radioactive material occurring during the reporting year that independently, or when added to a current burden, is estimated to result in a dose commitment to the critical organ in excess of 50 percent of the pertinent annual dose equivalent limits set forth in the radiological protection S/RIDs (Ref. 1).

To support preparation of the DOE REIRS report, HPT Internal Dosimetry is responsible for generating the computer file containing all of the internal exposure data (Ref. 18). The REIRS report is an annual radiation exposure report for DOE and DOE contractor radiation workers and for non-employee occupational workers at SRS per Code of Federal Regulations (CFR) 10 CFR 835 (Ref. 19).

7.6.4 RESPIRATORY PROTECTION

It is WSRC policy to protect employees from exposure to airborne radioactive contaminants by using facilities and equipment with physical barriers and other safeguards incorporated into their design. This is the preferred method of protection. When engineering controls are not feasible, or while they are being initiated, protection is provided through a combination of administrative controls and approved respiratory devices.

The manager of the Industrial Hygiene Programs Section of the OS&HT Department has overall responsibility and authority for the respiratory protection program (Ref. 20). An evaluation of the program is performed annually (Ref. 21).

Additional information on the respiratory protection program is addressed in Chapter 8 of this SAR. Industrial Hygiene record keeping and the occupational medical program are addressed in Sections 8.9 and 8.6.3, respectively.

7.7 RADIOLOGICAL MONITORING

Typical radiological control monitoring includes Area Radiation Monitors (ARMs), Continuous Air Monitors (CAMs), air sampling, personnel contamination monitoring, and ventilation monitoring. WSRC Procedure Manual 5Q1.7 provides additional information on specific types of monitors and systems (Ref. 22).

Each facility that processes/handles radioactive particulate byproduct or enriched material must have a Facility Annual Review of Monitoring Systems (FARMS). The FARMS is a joint venture between Facility Management and S&HO personnel. The FARMS considers the criteria for the protection of radiation and non-radiation workers, as discussed below (Ref. 10).

Radiological environmental monitoring consists of two major activities: effluent monitoring and environmental surveillance. Radiological effluent monitoring is a shared responsibility between Facility Management, S&HO, Savannah River Technology Center (SRTC), and Environmental Monitoring Section (EMS) personnel. However, Facility Management retains overall responsibility and ownership of a facility's radiological effluents. Environmental surveillance is the responsibility of EMS and SRTC.

The SRS Environmental Monitoring (EM) Program contains detailed descriptions of the existing activities, procedures, practices, and programs that implement the EM criteria and requirements set forth in the SRS EM Plan (Ref. 23, 24).

7.7.1 RADIOLOGICAL CONTROL MONITORING AND SURVEYS

7.7.1.1 General Requirements

Workplace monitoring provides a control mechanism to detect and quantify external radiation and radioactive contamination levels, enables measures to be taken to prevent unanticipated and unplanned exposures, and contributes to maintaining actual exposures ALARA. The monitoring must be routine and sufficient to control potential sources of radiation and radioactivity, and to demonstrate compliance with the radiation protection program. Determining the frequencies and locations of workplace monitoring is the responsibility of the site and must be commensurate with the actual work and exposure situations. Radiological surveys are recorded on a Radiation Survey Logsheet, which is used for determining personnel stay time, area postings, and other radiological work planning, as well as historical documentation. WSRC Procedure Manual 5Q1.2 provides instructions and techniques for documenting and performing radiation and contamination surveys on a Radiation Survey Logsheet (Ref. 10).

Instruments used to perform radiation surveys must be response-checked daily or prior to operation if used less frequently. Assessments of radiological conditions include a sufficient number of survey points to characterize the radiation and radioactive material present and to verify boundaries. Surveys are performed before, during, and at the completion of work that has the potential for causing changes in levels of radiation and radioactivity as well as routinely on

predetermined schedules. Survey frequencies are established based on potential radiological conditions, probability of change in conditions, and area occupancy factors (Ref. 7).

Monitoring results are reviewed by the S&HO supervisor. The review must ensure that all required surveys have been performed and that the documentation is accurate and complete. Monitoring results are made available to line management and used in support of pre- and post-job evaluations, ALARA preplanning, contamination control, and management of radiological control operations (Ref. 7).

7.7.1.2 Radiation Surveys

Performance of radiation surveys includes dose rate measurements of the general area, dose rates at a distance of 30 cm from a source or surface of interest to evaluate potential whole-body exposures, and dose rates on contact with potential sources of radiation where there is a potential for hands-on work (Ref. 7).

Surveys are conducted whenever operations are being performed that might result in personnel being exposed to small intense beams of radiation, such as those generated by shielded x-ray devices or due to removal or alteration of shielding (Ref. 7).

7.7.1.3 Contamination Surveys

Contamination surveys are conducted on a routine basis in affected areas. Frequencies for conducting routine contamination surveys are specified in WSRC Procedure Manual 5Q (Ref. 7).

Potentially radioactive materials in CAs, HCAs, or ARAs are surveyed prior to release. Contamination surveys on materials, equipment, and portable facilities for controlled release from an RA or RBA are conducted as specified in WSRC Procedure Manual 5Q1.1 (Ref. 8).

7.7.1.4 Area Radiation Monitors

ARMs are 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 entering these areas. The need for and placement of ARMs are documented and assessed when changes to facilities, systems, or equipment occur (Ref. 10). Where an ARM is incorporated into a safety interlock system, the circuitry must be such that a failure of the monitor either prevents entry into the area or prevents operation of the radiation producing device.

If installed instrumentation is removed from service for maintenance or calibration, an equivalent radiation monitoring program is maintained consistent with the potential for unexpected increases in radiation dose rates.

ARMs are calibrated annually and tested at least quarterly to verify audible alarm system operability and audibility under ambient working conditions and operability of visual alarms, as appropriate (Ref. 7).

7.7.1.5 Airborne Radioactivity Monitoring

Air monitoring equipment is used in situations where airborne radioactivity levels can fluctuate, and early detection of airborne radioactivity could prevent or minimize inhalation of radioactivity by personnel. Air monitoring equipment includes portable and fixed air sampling equipment and CAMs (Ref. 7).

Air sampling equipment is used to measure air concentrations to which persons are exposed. Preliminary assessments of air samples utilizing field survey techniques are performed promptly upon removal. Air sample results are evaluated as quickly as practicable to determine the need for respiratory protection, area evacuation (if necessary), worker intake and worker relief from respirator use (Ref. 7).

CAMs have alarm capabilities and sufficient sensitivity to alert personnel that immediate action is necessary in order to minimize or terminate inhalation exposures (Ref. 7). The proper operation of CAMs is verified annually by calibration, daily by performance of operational checks, and weekly by checking for instrument response with a check source or with ambient levels of radon and thoron daughters (Ref. 22).

7.7.2 RADIOLOGICAL ENVIRONMENTAL MONITORING

Radiological environmental monitoring consists of two major activities: effluent monitoring and environmental surveillance. The environmental and radiological protection S/RIDs requires that an EM Plan be prepared for each DOE site (Ref. 1, 24).

The purpose of the SRS EM Plan is to define the criteria, regulations, and guideline requirements with which SRS will comply (Ref. 24). These criteria and requirements are applicable to environmental monitoring activities performed in support of the SRS EM program (Ref. 23). They are not applicable to monitoring activities utilized exclusively for process control monitoring or radiological control monitoring.

The EM program requirements documented in the SRS EM Plan incorporate all applicable requirements of DOE guidance document DOE/EH-0173T (Ref. 23, 24, 25).

The SRS EM program serves two main purposes: it shows compliance with federal, state, and local regulations, as well as with DOE Orders, and it monitors any effects of SRS operations on the environment, both onsite and offsite (Ref. 23).

7.7.2.1 Radiological Effluent Monitoring - General Requirements

Radiological effluent monitoring results are a major component in the determination of compliance with applicable dose standards. DOE Order 5400.5 and proposed Rule 10 CFR 834 set annual dose standards to members of the public resulting from routine DOE operations at 100 mrem through all exposure pathways, 10 mrem from airborne releases, and 4 mrem from the drinking water pathway (Ref. 26, 27). Compliance with dose standards is determined by the Environmental Dosimetry Group of SRTC and is documented in the SRS Annual Environmental Report, which is issued to the general public (Ref. 24, 26).

In addition to this, SRS management is committed to and responsible for maintaining radiation exposures to the general public and releases of radioactive materials to the environment at ALARA levels. As part of the SRS environmental program, annual dose-based Environmental ALARA Release Guides are established for each operating area and for the site. Monthly and year-to-date release totals are compared to these guides in the SRS Monthly Radioactive Releases Reports. The SRS Environmental ALARA program is described in WSRC Procedure Manual 3Q (Ref. 28).

DOE Order 5400.5 and Proposed Rule 10 CFR 834 also establish Derived Concentration Guides (DCGs) for radionuclides in air and water (Ref. 26, 27). DCGs, which are applicable at the point of discharge from the conduit to the environment, are used as follows:

- As reference concentrations for conducting environmental protection programs
- For dose comparisons
- As screening values for considering Best Available Technology (BAT) treatment of liquid effluents

Annual average concentrations of radionuclides in effluents are compared to the DCGs in the SRS Annual Environmental Report. In addition to this, for radioactive liquid effluents, EMS compares the monthly concentrations and 12-month average concentrations against the DCGs. This comparison is documented in the SRS Monthly Radioactive Releases Reports. If, at any liquid effluent point, the sum of the fractional DCG values (based on consecutive 12-month average concentrations) for all radionuclides (except tritium) detectable in the effluent exceeds 1.0, then a BAT process would be initiated per WSRC Procedure Manual 3Q (Ref. 28).

7.7.2.2 Radiological Effluent Monitoring – Liquid Effluents

In addition to the DOE Order 5400.5 and proposed Rule 10 CFR 834 dose standards, the U.S. Environmental Protection Agency (EPA) drinking water standards for radionuclides in 40 CFR 141, apply at the water treatment plants serving Beaufort and Jasper counties in South Carolina and Port Wentworth, Georgia (Ref. 26, 27, 29). Compliance with the EPA drinking water standards is documented in the SRS Annual Environmental Report.

Each process area liquid effluent discharge point that releases or has the potential to release radioactive materials is sampled routinely and analyzed for radioactivity. Site streams are also sampled upgradient and downgradient of seepage basins and solid waste disposal facilities to calculate the amount of radioactivity migrating from them.

Liquid effluents are sampled continuously at, or very near, their points of discharge to the receiving streams. Three primary systems are used: paddlewheel samplers, Brailsford pumps, and Isco samplers. EMS personnel normally collect the liquid effluent samples weekly and transport them to the EMS laboratory for analysis.

EMS provides most of the necessary radioanalytical laboratory functions to conduct the site liquid effluent monitoring program. Specific low-level analysis for selected radioisotopes is performed by the SRTC Environmental Technology Section (ETS).

Liquid effluent flow rates generally are determined by one of four methods: U.S. Geological Survey flow stations, stream velocity measurements, Isco sampler flow meters, or pump capacity calculations. Effluent flow rates are used by EMS to determine the total radioactivity released.

A complete description of the sampling and analytical procedures used for radiological liquid effluent monitoring is presented in the SRS EM Program (Ref. 23).

7.7.2.3 Radiological Effluent Monitoring - Airborne Effluents

In addition to the dose standards in DOE Order 5400.5 and proposed Rule 10 CFR 834, radiological airborne releases are regulated by 40 CFR 61, Subpart H, National Emission Standard for Hazardous Air Pollutants (NESHAP) - Radiological (Ref. 26, 27, 30). The SRS NESHAP radionuclide program incorporates sampling, monitoring, and dose assessment practices that meet or exceed the EPA requirements. Compliance with the NESHAP dose standards (10 mrem per year) is documented in the SRS Annual Environmental Report.

Process area stack discharge points that release or have the potential to release radioactive materials are continuously monitored by applicable on-line monitoring and/or sampling systems. For both routine and nonroutine operations, the reactor facilities and tritium facilities use real-time instrumentation to determine instantaneous and cumulative atmospheric releases of tritium and noble gas radioisotopes. All other monitored radionuclides are sampled using filter papers, charcoal filters, or other air effluent sampling media.

Filter paper samples that are used to monitor routine releases of radioactive particulates are collected daily or weekly and screened for radioactivity by S&HO personnel. Charcoal canisters (used to monitor radioiodines) are collected weekly. On a weekly basis, S&HO personnel routinely transfer the charcoal canisters and composited filter paper samples to EMS personnel for radioanalysis.

EMS provides most of the necessary radioanalytical laboratory functions to conduct the site airborne effluent monitoring program. Specific low-level analysis for selected radioisotopes is performed by ETS.

For facilities with the potential to have unplanned releases, or unplanned increases in emission levels of radioactive particulates and/or radioiodines, additional on-line instrumentation is installed to signal the need for corrective actions and to provide real-time monitoring of emissions following an accident.

Stack flows generally are determined with hot-wire anemometers, Pitot tubes, or fan capacity calculations. Sample line flow rates usually are determined with in-line rotameters or hot-wire anemometers. Flow rates are used to determine the total quantity of radioactivity released.

A complete description of the sampling and analytical procedures used for radiological airborne effluent monitoring is presented in the SRS EM Program (Ref. 23).

7.7.2.4 Radiological Environmental Surveillance - General Requirements

The environmental protection S/RIDS mandate the establishment of and present the general requirements for an environmental surveillance program at DOE sites (Ref. 1, 26, 27, 31). Further specific program elements are detailed in DOE/EH-0173T (Ref. 25).

Other regulations impact the implementation and conduct of portions of the radiological surveillance program. These include EPA's Resource Conservation and Recovery Act and the Comprehensive Environmental Response, Compensation, and Liability Act, which describe requirements for environmental surveillance samples to be used for waste site characterization studies (Ref. 32).

The radiological surveillance program conducted at SRS is designed to survey and quantify any effects that routine and nonroutine onsite operations may have on the site, the surrounding area, or people living in the vicinity of SRS (the onsite and offsite environment). The program is conducted to meet the following criteria:

- Verify compliance with environmental commitments made by the site in environmental impact statements, environmental assessments, and other such documents.
- Characterize and define trends in the biological environment.
- Establish environmental baselines of environmental quality.
- Continually assess pollution abatement programs.
- Identify and quantify new or existing environmental problems.

To accomplish these goals, routine surveillance of all radiation exposure pathways (ingestion, inhalation, immersion, and submersion) is performed on all environmental media that may lead to a measurable annual dose at the site boundary. Included in this surveillance are analyses of the

atmosphere, surface waters, drinking water, rainwater, sediment and soil, vegetation, food products, and wildlife. Also, extensive monitoring of ambient gamma radiation levels is performed onsite, at the site boundary, and in population centers (surrounding communities).

A complete description of the program rationale and design, as well as the sampling locations, and sampling and analytical procedures used for radiological environmental surveillance, is presented in the SRS EM Program (Ref. 23).

7.7.3 ASSOCIATED RECORDS/REPORTS

Records generated as part of the radiological control monitoring program are maintained in accordance with procedures and the department Records Inventory and Disposition Schedule (RIDS) (Ref. 3, 10).

For the SRS EM program, numerous records and reports are generated to document environmental monitoring activities and to demonstrate compliance with applicable regulations. In addition, records and reports are used to notify the proper officials of unusual or unforeseen environmental occurrences, to maintain an accurate and continuous record of the effects of SRS operations on the environment, and to communicate results of environmental monitoring activities and compliance programs to DOE, other government agencies, and the general public.

The environmental protection S/RIDs contain requirements for reporting effluent monitoring and environmental surveillance activities (Ref. 27).

A complete description and listing of the records and reports generated in support of the radiological environmental monitoring program is presented in the SRS EM Program (Ref. 23).

7.7.4 METEOROLOGICAL DATA COLLECTION/EVALUATION

The facility-specific SARs contain information on the location of weather monitoring stations, instrumentation and alarms, and equipment surveillance. Chapter 1 of this SAR provides additional information on site characteristics.

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7.8 RADIOLOGICAL PROTECTION INSTRUMENTATION

7.8.1 CRITERIA FOR SELECTION OF EQUIPMENT AND INSTRUMENTATION

Selection of radiation instrumentation for a specific facility is dependent on the radionuclides handled in that facility. Fixed instruments, such as ARMs and CAMs, are required by 10 CFR 835 to characterize the workplace (Ref. 19). The selection of specific types of instrumentation and their initial placement are the responsibility of HPT. Information on facility layout, location of materials (and their estimated quantities), and design airflow patterns is used to determine initial placement of the instrumentation. Once the facility is operational and actual conditions can be measured, these locations are reviewed for adequacy.

An instrument procurement is initiated when the need for a given type of instrument is identified by new monitoring requirements or by the necessity of replacing obsolete or damaged instruments. Selection of commercially available instruments with all or most of the desired capabilities is made from vendor literature or documented design articles. Sample instruments are then obtained from vendors for onsite evaluation and testing.

A purchase requisition is initiated with purchase specifications based on the technical requirements. Once a piece of equipment is purchased, vendor-furnished documentation is placed in the appropriate files and supplied to the user. Operating and calibration instructions are prepared and incorporated into the appropriate procedure manuals.

Prior to initial operation, each instrument is examined and tested by S&HO and instrument maintenance groups to ensure that all specifications have been met. Equipment and sources used for calibration and testing of new instruments are controlled and calibrated to determine if the instrument meets purchase specifications. Traceability to the National Institute of Standards and Technology (NIST) is provided as required. All records applicable to the purchased RC instruments are retained in project files, purchase requisition files, and/or instrument evaluation files of the S&HO Department.

The criteria for selection, examination, and testing of radiological protection equipment and instrumentation are discussed in WSRC Procedure Manual 5Q1.2 and WSRC Procedure Manual Q2 (Ref. 10, 33).

7.8.2 CONTROL OF THE CALIBRATION PROCESS

Personnel in the S&HO instrument calibration facility are responsible for storage, maintenance, calibration, and distribution of radiation survey instruments, direct reading and electronic dosimeters, and for procurement of this equipment. This facility contains NIST-traceable radiation calibration standards, remote control apparatus for safe source manipulation, instruments to determine air-flow rate, shielded rooms, and related support equipment. The S&HO Department operates the instrument calibration facility.

After calibration and before distribution, instruments are labeled with an expiration date by which they must be returned for recalibration. Calibrated instruments are distributed to all areas of the site from the facility. Instruments in use in the field are routinely checked for proper response using standard radiation sources.

Calibration of fixed radiological protection instrumentation is the responsibility of Central Services Works Engineering (CSWE). Fixed instrumentation includes personnel contamination monitors, portal monitors, area radiation monitors, and CAMs. Initial calibration is performed when the equipment is installed. The length of time between calibrations varies depending on the instrument but does not exceed one year. CSWE personnel perform preventive maintenance and repair on fixed instrumentation. Maintenance of fixed instrumentation is performed in each area. Portable instrument repair is performed by CSWE personnel in S&HO instrument calibration facility.

Additional details pertaining to the control and calibration of portable radiation monitoring equipment are provided in the WSRC Procedure Manual Q2 (Ref. 33).

7.8.3 QUALITY ASSURANCE FOR CALIBRATION AND MAINTENANCE

S&HO selects, evaluates, and approves all radiological instrumentation (both portable and fixed) used at SRS for radiological protection (Ref. 34, 35). Radiological instruments are used only to measure the radiation for which their calibrations are valid. The radiological protection S/RIDs mandate the requirements for radiological instrumentation calibration (Ref. 1). Calibrations must use NIST traceable standards.

Calibration procedures are developed by S&HO for each radiological instrument type and include frequency of calibration, precalibration requirements, primary calibration requirements, periodic performance test requirements, calibration record requirements, and maintenance requirements.

In unusual and limited situations, it may be necessary to use an instrument under conditions that vary significantly from those for which the instrument is designed. Special calibrations are performed for use of instrumentation outside manufacturer's specifications. The instruments are adjusted, calibrated, and labeled to identify the special conditions and used only under the special conditions for which it was calibrated. These special conditions are discussed in WSRC Procedure Manual 5Q (Ref. 7).

Instruments bear a label or tag with the date of calibration and date recalibration is due. HPT must provide immediate notification to the appropriate S&HO office of the out-of-calibration condition of any instrument not source-checked in the field that has "as found" readings, indicating it may have been used while out of calibration. The S&HO office reviews surveys performed with the instrument while it was out of calibration (Ref. 3).

A program for preventive and corrective maintenance of radiological instrumentation has been established and documented. Preventive and corrective maintenance is performed using

components and procedural recommendations at least as stringent as those specified by the instrument's manufacturer. Radiological instruments undergo calibration prior to use following any preventive or corrective maintenance or any adjustment that voids the previous calibration (Ref. 7).

Refer to Chapter 14 of this SAR for additional discussion of QA.

7.8.4 TYPES OF DETECTORS AND MONITORS

See the facility-specific SARs for a summary of the quantity, sensitivity, and range of instrumentation in use.

7.8.5 CONTAMINATION CONTROL EQUIPMENT AND FACILITIES

See the facility-specific SARs and Section 7.7 of this SAR for more information.

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7.9 RADIOLOGICAL PROTECTION RECORD KEEPING

WSRC Procedure Manual 5Q and WSRC Procedure Manual Q1-1 contain the prescribed practices for preparing and retaining radiologically related records (Ref. 3, 7). These records provide employees and management with knowledge of radiological exposures and are needed to demonstrate the effectiveness of the overall program. The workforce and management are required to use records to document radiological safety afforded to personnel at SRS. Records of radiological programs may be required to support worker health studies and future disputes or claims. Therefore, these records must be of high quality, readily retrievable and managed for the prescribed retention period. Records are handled such that personal privacy is protected (Ref. 7).

7.9.1 INVENTORY, RETENTION AND DISPOSITION OF RECORDS

WSRC Procedure Manual 5Q requires the establishment of a radiological records management program (Ref. 7). This program ensures that auditable records and reports are controlled through the stages of creation, distribution, use, arrangement, storage, retrieval, media conversion (if applicable) and disposition. The S&HO Department is responsible for the inventory, retention, and disposition of radiological records.

Where radiological services (e.g., dosimetry and laboratory analyses) are purchased, an agreement is required regarding creation and disposition of records in accordance with WSRC Procedure Manual Q1-1 and WSRC Procedure Manual 1Q (Ref. 3, 35). Specification, maintenance, and final disposition of records from vendors for radiological services is the responsibility of S&HO (Ref. 7).

Records are categorized as Permanent, Lifetime, or Nonpermanent Records to ensure assignment of proper retention times. Specific retention periods are specified in the RIDS. The RIDS contains, by category, a list of all records, their appropriate retention period, and appropriate disposition instructions. Additions, deletions, or other changes to the RIDS must be sent to the Records Management Coordinator. Revisions are reviewed and approved by Document and Records Administration.

7.9.2 REPORTS

Refer to Section 7.6.3.6 for discussions of radiological reporting.

7.9.3 MAINTENANCE OF PROCEDURES

The site radiation protection program is primarily documented in S&HO manuals and procedures. WSRC Procedure Manual Q1-1 establishes the responsibilities and requirements for preparation, review, approval, revision, cancellation, and administration of S&HO Department procedures (Ref. 3). S&HO procedures are reviewed during each use and updated as necessary. In addition, all S&HO procedures are formally reviewed in their entirety at least every 2 years. In

conducting the periodic review, individual reviewers consider, as a minimum, the following criteria (Ref. 3):

- QA requirements
- Process, equipment, and building changes
- Organizational changes
- Improvements in basic plans and procedures
- Clarity of presentation of plans and procedures
- DOE requirements

WSRC Procedure Manual Q1-1 also prescribes the responsibilities and requirements for controlled revisions to WSRC Procedure Manual 5Q (Ref. 3, 6). In accordance with Management Requirement and Procedure 3.26, "Management of Company-Level Policies and Procedures," a periodic review of the WSRC Procedure Manual 5Q must be conducted at least every two years (Ref. 7, 36). Management Standards is responsible for establishing the periodic review and coordinating the processing of these reviews with the S&HO Department (Ref. 3).

7.10 OCCUPATIONAL RADIATION EXPOSURE

The facility-specific SARs provide a summary of projected annual exposures to facility workers from radiological hazards based on historical facility or related operations data. Identification of the methods used in the projected exposures and a comparison of the projected exposures with allowable limits are also included. See Section 7.6.3.6 for information on radiation exposure reports and records.

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7.11 REFERENCES

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3. Safety and Health Administrative Procedures Manual. WSRC Procedure Manual Q1-1, Westinghouse Savannah River Company, Aiken, SC, January 1999.

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15. Savannah River Site Internal Dosimetry Technical Basis Manual. WSRC-IM-90-139, Rev. 6, Westinghouse Savannah River Company, Aiken, SC, December 1997.

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19. Occupational Radiation Protection. Code of Federal Regulations, 10 CFR 835, Office of the Federal Register, Washington, DC, November 1998.

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21. Industrial Hygiene Manual. WSRC Procedure Manual 4Q, Westinghouse Savannah River Company, Aiken, SC, November 1998.

22. Safety and Health Radiation and Air Monitoring Systems Procedures. WSRC Procedure Manual 5Q1.7, Westinghouse Savannah River Company, Aiken, SC, January 1999.

23. SRS EM Program. WSRC 3Q1-2, Volume 1, Section 1100, Westinghouse Savannah River Company, Aiken, SC, February 1998.

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29. National Primary Drinking Water Regulations. Code of Federal Regulations, 40 CFR 141, Office of the Federal Register, Washington, DC, July 1992.

30. National Emission Standard for Hazardous Air Pollutants (NESHAP). Code of Federal Regulations, 40 CFR 61, Subpart H, Office of the Federal Register, Washington, DC, 1989.

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Table 7.6-1 Summary of Dose Limits

TYPE OF EXPOSURE	ANNUAL LIMIT (rem)
Radiological Worker: Whole body (internal + external)	5
Radiological Worker: Lens of eye	15
Radiological Worker: Extremity (hands and arms below the elbow; feet and legs below the knees)	50
Radiological Worker: Any organ or tissue (other than lens of eye) and skin	50
Declared Pregnant Worker: Embryo/Fetus	0.5 per gestation period
Minors and Students (under age 18): Whole body (internal + external)	0.1
Visitors* and public: Whole body (internal + external)	0.1

* Applies to visitors who have not completed training in accordance with Articles 632 or 633 or have not met the special considerations of Article 657 of Reference 7.

Notes:

1. Internal dose to the whole body shall be calculated as committed effective dose equivalent. The committed effective dose equivalent is the resulting dose committed to the whole body from internally deposited radionuclides over a 50-year period after intake. See the radiological protection S/RIDS for the weighting factors to be used in converting organ dose equivalent to effective dose equivalent for the whole body dose (Ref. 1).

2. Background, therapeutic and diagnostic medical exposures shall not be included in either personnel radiation dose records or assessment of dose against the limits in this table.

3. See Chapter 2, Appendix 2C of WSRC Procedure Manual 5Q for guidance on non-uniform exposure of the skin (Ref. 7).

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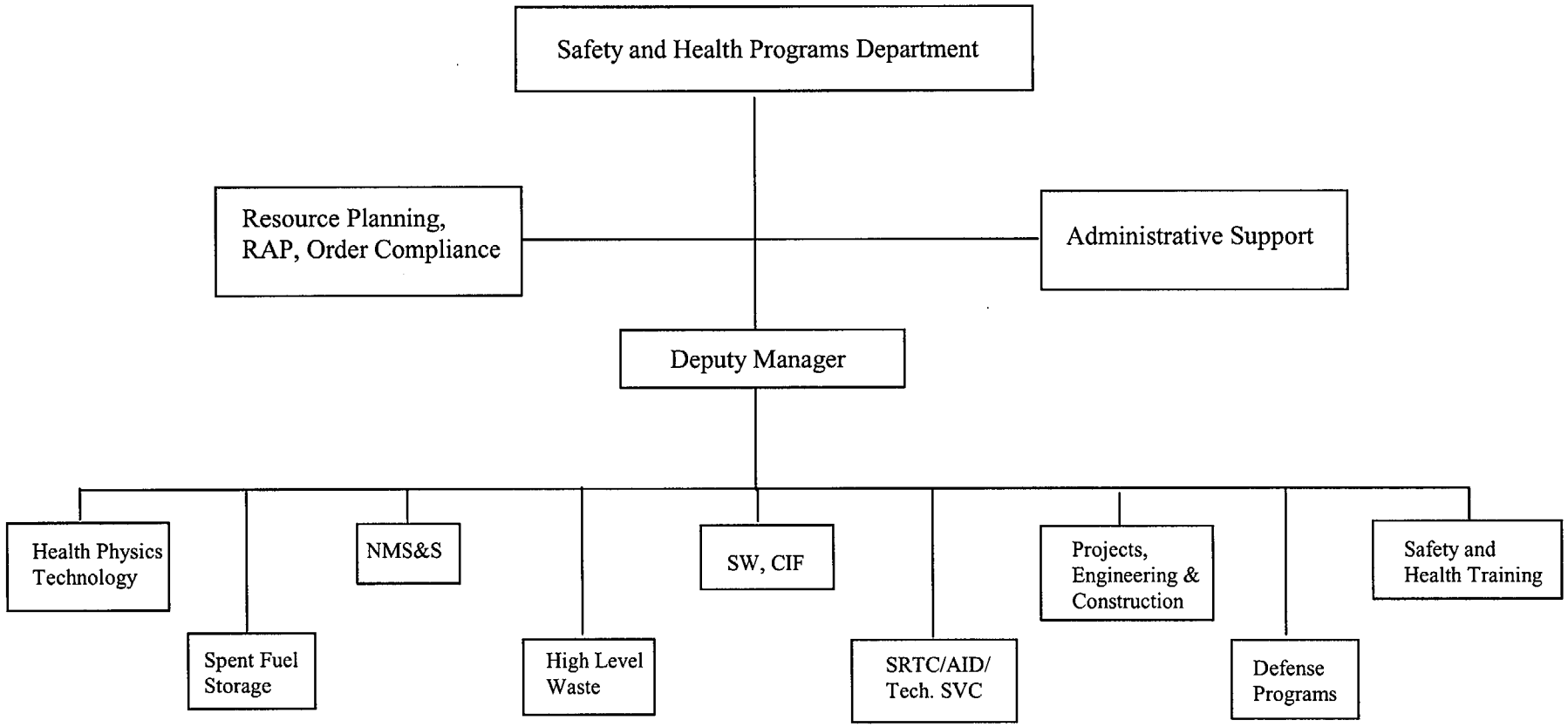
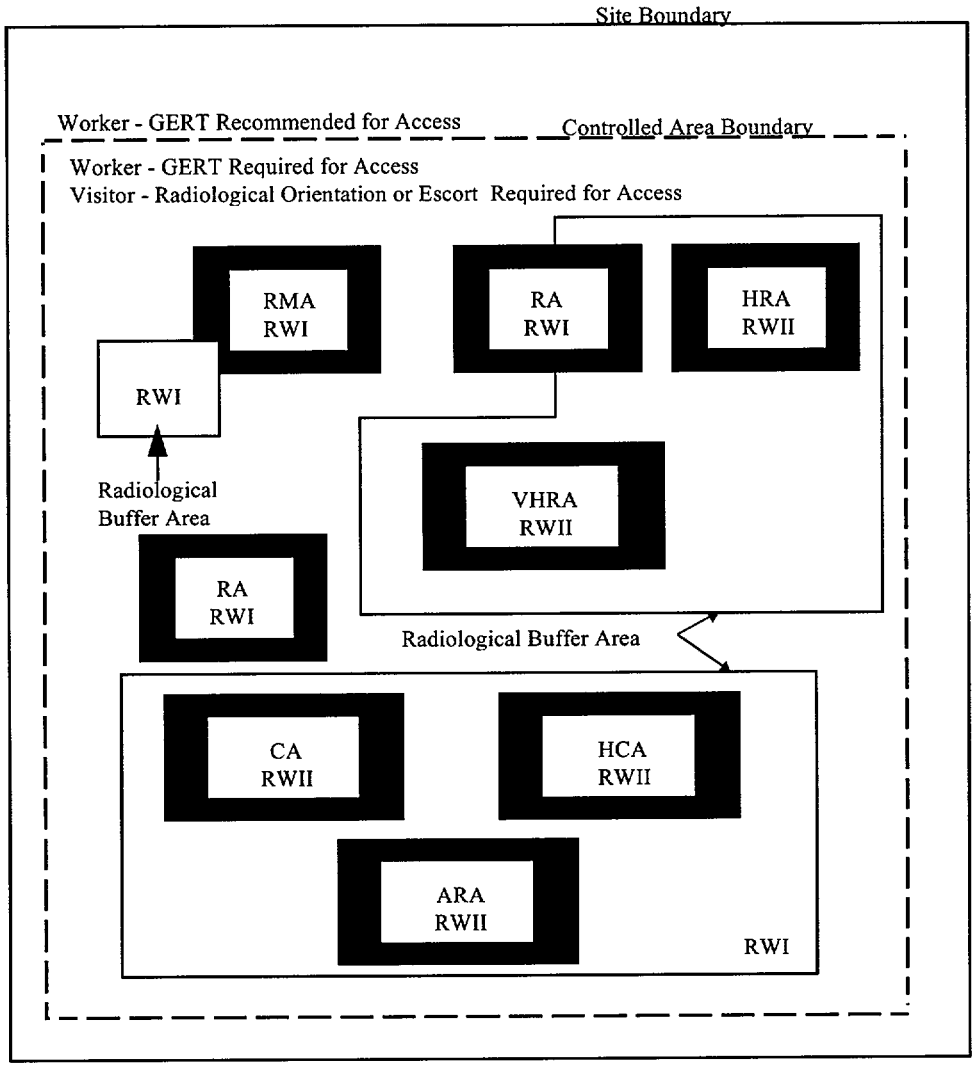


Figure 7.3-1 Typical Safety and Health Organization



Legend:

GERT - General Employee Radiological Training	HRA - High Radiation Area
RWI - Radiological Worker I	VHRA - Very High Radiation Area
RWII - Radiological Worker II	CA - Contamination Area
RMA - Radioactive Material Area	HCA - High Contamination Area
RA - Radiation Area	ARA - Airborne Radioactivity Area

Figure 7.5-1 Typical Control Areas and Required Training

GENERIC SAFETY ANALYSIS REPORT

CHAPTER 8

HAZARDOUS MATERIAL PROTECTION

September 1999

Westinghouse Savannah River Company
Aiken, SC 29803

SAFETY • RESPONSIBILITY • SECURITY
SRS
SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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